HOLTEC
INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

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

September 30, 2008

Reference:
1. USNRC Docket No. 71-9261 (HI-STAR 100), TAC L24029
2. Holtec Project 5014
3. Letter from K. Hardin (USNRC) to T. Morin (Holtec), dated April 15, 2008
4. Letter from K. Hardin (USNRC) to S. Anton (Holtec), dated June 15, 2007
5. Holtec Letter 5014631, dated August 3, 2007
6. Holtec Letter 5014605, dated October 5, 2006
7. Holtec Letter 5014637, dated October 5, 2007
8. Holtec Letter 5014635, dated September 27, 2007

Subject: Response to 2nd Request for Additional Information (RAI) on License Amendment Request (LAR) 9261-5 to HI-STAR 100 Certificate of Compliance (CoC) No. 9261

Via letter (Reference 3), the SFST requested additional information on the proposed amendment (Reference 6) to the HI-STAR 100 Certificate of Compliance (CoC). The license amendment request originally submitted (Reference 6) was modified (Reference 5) in response to NRC Staff's initial request for additional information (Reference 4). Additionally Holtec proprietary reports were sent to the NRC (References 7 and 8) to support the proposed license amendment request.

The responses to the request for additional information (RAI) are provided in Attachment 1. Additional changes to the proposed HI-STAR 100 CoC and Technical Specifications (TS) resulting from the responses to the RAI are provided in Attachment 2. Changes made to the Safety Analysis Report (SAR) text are provided in Attachment 3 with a List of Effective Pages preceding the SAR changes. Justifications for these changes are provided in the response to the RAI that initiated the change. Attachments 4 and 5 contain modified Summary of Proposed Changes and Licensing Drawing Changes, respectively. To aid in the review, Attachment 6 identifies the specific changes to the CoC/TS and SAR text as a result of this RAI response.

NMSS01

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

Holtec proprietary reports are provided to support the responses under separate cover in Holtec letter 5014667.

The following attachments are provided with this letter:

Attachment 1: Written Responses to NRC Request for Additional Information.

Attachment 2: Revised Proposed CoC and Technical Specification Changes. All text changes are marked by vertical bars in the right margin, additions are in italics and deletions in strikeout.

Attachment 3: Proposed Revised SAR Sections. All changed sections are provided and are labeled as either "Proposed Rev. 13" (initial request), "Proposed Rev. 13a" (changes due to RAI response on August 3, 2007), or "Proposed Rev 13b" (changes due to this RAI response) in the footer. All text changes are marked by vertical bars in the right margin. Deletions are in strike-out and insertions are in italics.

The following SAR Sections and Supplements are provided along with a List of Effective Pages:

- Chapter 1: 1.0, 1.1, 1.2, 1.3, 1.4 (including revised drawings), 1.5, 1.6, 1.I
- Chapter 2: 2.1, 2.3, 2.4, 2.6, 2.7, 2.A, 2.I
- Chapter 3: 3.1, 3.2, 3.3, 3.4, 3.6, 3.7, 3.I
- Chapter 4: 4.0, 4.1, 4.2, 4.3, 4.4, 4.I
- Chapter 5: 5.1, 5.2, 5.4, 5.5, 5.I
- Chapter 6: 6.1, 6.2, 6.3, 6.4, 6.I
- Chapter 7: 7.0, 7.1, 7.2, 7.3, 7.4, 7.5, 7.I
- Chapter 8: 8.0, 8.1, 8.2, 8.3, 8.I

Attachment 4: Modified Summary of Proposed Changes reflecting all the changes requested in the LAR.

Attachment 5: Modified Licensing Drawing Changes reflecting all the changes requested in the LAR.

HOLTEC
INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

Attachment 6: List of CoC/TS and SAR text changes specific to second RAI response.

Sincerely,

Tammy Morin
Acting Licensing Manager,
Project Manager, LAR 9261-5
Holtec International

cc: USNRC Document Control Desk, hardcopy only
Ms. Kimberly Hardin, Sr. Project Manager, NRC/NMSS/SFST, 10 copies
Mr. Eric Benner, Licensing Branch Chief, NRC/NMSS/SFST, Letter only
Mr. Larry Pulley, PSE&G, Letter only via email
Group 1, Holtec, Letter only via email

LAR 9261-5

ATTACHMENT 1
to
Holtec Letter 5014666

RAI Responses

Attachment 1 to Holtec Letter 5014666
Holtec Responses to NRC RAI

Technical Specifications:

TS-1. Verify the number of 6x6D assemblies containing rods with maximum initial enrichments of 5.5 wt.%.

The applicant indicates that only two of these 6x6D assemblies may be loaded in the HI-STAR HB (see note 14 of Table A.3 in Appendix A to the CoC). However, Table 3.1-2 of the Humboldt Bay (HB) ISFSI FSAR indicates there are four such assemblies. Table A.3 of Appendix A of the CoC and Table 1.I.4 of the SAR should be updated as necessary.

This information is needed to confirm compliance with 10 CFR 71.33(b).

Holtec Response:
Holtec confirms that there are only two fuel assemblies at Humboldt Bay with a single high power test rod. According to the fuel data document supplied to Holtec there were initially four assemblies (HE41, HE42, HE43, and HE44) with a single high power test rod with 5.5% enrichment. However, the same document stated that the high power test rod was removed from fuel assemblies HE41 and HE43 and shipped off-site. Therefore only two assemblies containing the high power test rod with 5.5% enrichment remain at the Humboldt Bay site, HE42 and HE44, and require transportation in the HI-STAR HB System. This response also provides justification for the response to RAI 6-4.

TS-2. Justify the proposed limit of 1 kg of stainless steel for stainless steel clad fuel debris in a HB damaged fuel container (DFC).

The applicant has proposed to limit the amount of stainless steel in a HB DFC to 1 kg. The basis for this limit is not clear. Further, the practicality of ensuring compliance with this proposed limit is also not clear. The applicant should establish a limit that is based upon such considerations as the amount of stainless steel clad fuel debris known/estimated to be at the HB plant and the ability to ensure compliance with the limit as well as consistency with the limits given in the HB ISFSI license and technical specifications. Analyses in the SAR should support whatever limit is established for this material.

This information is needed to confirm compliance with 10 CFR 71.33(b) and 71.35.

Holtec Response: According to Humboldt Bay fuel records there are no more than 337 linear inches of fuel fragments remaining at the Humboldt Bay Power Plant site. These fuel fragments may contain stainless steel or zircaloy cladding. As a bounding approach, it is assumed that all of the clad is stainless steel. With the known cladding dimensions, the amount of stainless steel is calculated to be 1.25 kg. Based on this, a maximum amount of 1.5 kg is specified for each cask. Note 3 of Table A.1 Section VI of the TS, has *been updated to specify this limit as well as Note 4 in SAR Table 1.I.7. Supplement 5.I has also been updated to reflect this limit. Compliance can be demonstrated based on the fuel records that were used to calculate the value.*

Chapter 2 Structural Review

NOTE: Holtec proprietary Report, HI-2073743, "Benchmark the LS-DYNA Impact Response Prediction Model for the HI-STAR Transportation Package Using the AL-STAR Impact Limiter Test Data," presents the LS-DYNA approach to modeling the free-drop rigid body impact response for the HI-STAR family of transportation packages, including the HI-STAR HB. The staff evaluated the report and documents for the issues for which the attached proprietary additional information is needed to complete the review listed as questions 2-1 through 2-11. Recognizing that long lead-time may potentially be needed to resolve the issues, the staff will review other justifiable methods, if proposed, for determining rigid-body decelerations for the HI-STAR HB cask. As discussed in the review below in questions 2-1 through 2-4, this includes, but is not limited to, model similitude analyses for applying the Appendix 2.A differential equation method, which is based on the impact limiter drop tests of a ¼-scale HI-STAR 100 package, to the HI-STAR HB with identical impact limiter configurations except for aluminum section crush strengths.

Holtec Response: Holtec has reviewed the structural issues raised by the staff concerning Holtec proprietary report HI-2073743, "Benchmark the LS-DYNA Impact Response Prediction Model for the HI-STAR Transportation Package Using the AL-STAR Impact Limiter Test Data". Upon weighing the issues, we have decided that the most prudent path forward is to qualify the HI-STAR HB transport package using the differential equation method as defined in Appendix 2.A of the HI-STAR 100 SAR. The LS-DYNA benchmarking report (HI-2073743) is no longer used or referenced in the SAR. With that in mind, the responses to RAI questions 2-1 through 2-5 are provided below.

Since the LS-DYNA approach to modeling the free-drop rigid body impact response is no longer used for the HI-STAR HB System, the additional proprietary information requested by the Staff to complete its review of Holtec proprietary report HI-2073743 is not required. Therefore, no responses are provided to proprietary RAI questions 2-1 through 2-11.

2-1 Refer to Drawing 5014-C1765, Sheet 3 of 7 (Revisions 13 and 13a). Perform an analysis by using the Appendix 2.A differential equation method or other justifiable means to support the Note 3 design change statement, "As an option for HI-STAR 100, aluminum section Types 2 and 5 may be replaced in full by 1,420 psi (NOM.) uni-directional material."

Drawing 5014-C1765, Sheet 6 of 7 (Revision 12), depicts the previously approved crush strengths of 700 psi (uni-directional) and 2300 psi (cross core) for the aluminum section Types 2 and 5, respectively. Sheet 3 of 7 (Revision 12) depicts that, for the top impact limiter, six (6) Type 5 and ten (10) Type 2 interior sections are used. As displayed in Sheet 7 of 7 (Revision 12), all sixteen (16) aluminum honeycomb sections are of Type 2 construction for the bottom impact limiter. The staff notes that the proposed optional strength of 1,420 psi for the Type 2 sections is much higher than the previously approved 700 psi. As such, the optional crush strength may potentially result in higher cask decelerations, which must be shown to remain bounded by the design bases, including the 60 g for the bottom end HAC drop test.

This information is needed to confirm whether the package design and contents complies with 10 CFR 71.35(a) requirements.

Holtec Response: Note 3 on Drawing 5014-C1765 (Sheet 3 of 7) does not represent a design change as the note was previously approved as part of HI-STAR 100 SAR Revision 12. In Proposed SAR Revision 13 and 13a, an editorial change was made to the note to clarify that it applies only to the HI-STAR 100 System (not the HI-STAR HB System).

The optional construction described in Note 3 is also supported by the analysis presented in Appendix 2.A of the latest approved SAR. See response to RAI 2-2 for further details.

2-2 Refer to Drawing 5014-C1765, Sheet 6 of 7 (Revision 13a). Perform an analysis by using the Appendix 2.A differential equation method or other justifiable means to support the proposed design changes of revising the previously approved (Revision 12) and recently documented/proposed (Revisions 13, 13a) crush strengths for all Types 1, 2, 3, 4, and 5 aluminum sections for the HI-STAR 100 transportation package.

The staff notes the design changes for the aluminum section crush strengths as follows.

Section Type	Proposed Strength (psi) Rev. 13a	Appr'd/Docum'd Strength (psi) Revs. 12, 13
Type 1 (outer)	780	700
Type 1 (inner)	1,940	1,700
Type 2	780	700
Type 3	2,500	2,300
Type 4A	2,500	2,300
Type 4B	1,230	1,100
Type 5	2,500	2,300

It's unclear what evaluations were performed to justify the crush strength design changes to ensure that the resulting cask free-drop cask decelerations remain to be bounded by the previously established design bases for HI-STAR 100, including the 60 g for all HAC drop orientations.

This information is needed to confirm whether the package design and contents complies with 10 CFR 71.35(a) requirements.

Holtec Response: The changes to the crush strengths for all Types 1, 2, 3, 4, and 5 aluminum sections on Drawing 5014-C1765 (Sheet 6 of 7) in Proposed SAR Revision 13a are made to more precisely align the drawing with the actual crush strengths used in the *impact limiter drop tests of the ¼-scale HI-STAR 100 package and in the benchmark analysis presented in Appendix 2.A.*

Up to and including the latest approved SAR (Revision 12), the crush strengths reported in Chapter 1 (Drawing 5014-C1765, Sheet 6 of 7) were nominal design values chosen prior to procuring the material for the ¼-scale drop tests. The actual crush strengths of the material received from Hexcel Corporation for the ¼-scale impact limiters were slightly different from the nominal values because of material availability and manufacturing tolerances. The analytical results, which are reported in Appendix 2.A of SAR Revision 12, were obtained using the actual crush strengths for the as-received aluminum honeycomb sections (input values based on average crush strength for each type as determined from HI-981891, Impact Limiter Drop Test Report – Second Series, Rev. 3). The following table presents the crush strength data used to obtain the results in Appendix 2.A, which utilizes the differential equation method.

Table 1: CRUSH STRENGTH DATA FOR ¼ SCALE TEST IMPACT LIMITERS AND ANALYTICAL SIMULATION OF DROP TESTS

ALUMINUM HONEYCOMB SECTION	*CRUSH STRENGTH (PSI)*
Type 1 (outer)	*783*
Type 1 (inner)	*1943*
Type 2	*783**
Type 3	*2583*
Type 4A	*2583*
Type 4B	*1231*
Type 5	*2583**

** Top impact limiter has six (6) Type 5 and ten (10) Type 2 interior sections all of the same size. The weighted average of these sixteen sections is 1458 psi. This value is used as input for the analytical model of the top impact limiter, and it is the basis for Note 3 on drawing 5014-C1765 (Sheet 3 of 7). Bottom impact limiter has sixteen (16) Type 2 interior sections and zero Type 5 sections.*

Based on the parametric studies reported in Appendix 2.A of the HI-STAR 100 SAR, the acceptable range for the aluminum honeycomb crush strength, for each section type, is between 82% and 100% of the value in Table 1 above. In other words, the crush strengths in Table 1 represent the maximum acceptable values for each section type, and the minimum acceptable values are obtained by multiplying the crush strengths in Table 1 by a factor of 0.82. The crush strengths specified on drawing 5014-C1765 (Sheet 6 of 7) in Revision 12 of the HI-STAR 100 SAR fall within the acceptable range (i.e., 82% to 100%). However, in SAR Revision 12 the drawing does not indicate the maximum and minimum acceptable values for crush strength. In order to improve the drawing and clearly establish the design basis limits for the aluminum honeycomb material, drawing 5014-C1765 has been updated to specify the material crush strengths in Table 1 above as the maximum acceptable values for each section type. The minimum acceptable values *have also been indicated on the latest revised drawing, which is included with this RAI response submittal.*

2-3 Refer to the August 3, 2007, response to the Request for Additional Information, Q 1-1. Verify the statement made on the optional aluminum section crush strengths, "[A]s part of the LAR, for the generic HI-STAR 100, a uniform arrangement of 1,455 psi crush strength material is added as an option....a similar option has been added to the licensing drawing for HI-STAR HB to use a material with uniform crush strength of 690 psi as seen on Drawing 5014-C1765 (Revision 13a), Sheet 3, Note 4."

The subject drawing identifies the optional crush strength of 1,420 psi, contrary to the 1,455 psi cited above, for the generic HI-STAR 100. Also, Note 4 is added to the Revision 13a drawing, Revision 13a, which identifies the optional crush strength of 800 psi, in lieu of 690 psi, for HI-STAR HB. The optional crush strengths noted in the drawings are different from those discussed in the RAI response.

This information is needed to confirm compliance with 10 CFR 71.7(a).

Holtec Response: As noted by the staff, there are inconsistencies between the August 3, 2007 response to RAI Q 1-1 and Notes 3 and 4 on Drawing 5014-C1765 (Sheet 6 of 7) in Proposed SAR Revision 13a. While the statements made in the response to RAI Q 1-1 are correct, the optional crush strengths identified in Notes 3 and 4 on Drawing 5014-C1765 are not. Notes 3 and 4 have been updated to accurately identify the optional crush strengths for the HI-STAR 100 and HI-STAR HB top impact limiter as 1458 psi (see Table 1 of RAI Response 2-2) and 694 psi, respectively. Note that previously, in the response to RAI Q 1-1, the optional crush strengths were rounded down to the nearest 5 psi increment (i.e., 1455 psi and 690 psi). Additional clarifications on the notes have also been made in the revision submitted with this response.

2-4 Considering the impact limiter crush strengths tabulated in Drawing 501-C1764, Sheet 1 of 7 (Revisions 13, 13a), provide calculations to show that the most damaging rigid body cask decelerations for the HI-STAR HB remain to be bounded by the design basis of 60 g for the HI-STAR 100 subject to the HAC drop tests.

The staff agrees with the Holtec assertion that HI-STAR 100 structural analysis results bound HI-STAR HB results for the same cask decelerations of 60 g. Hence, should the Appendix 2.A differential equation method be used for determining rigid body decelerations for the HI-STAR HB cask, justification must also be provided to demonstrate that the dynamic multipliers, Zs, which were originally determined for HI-STAR 100, are appropriately reduced for the HI-STAR HB application.

This information is needed to confirm whether the package design and contents complies with 10 CFR 71.7(a) and 71.35(a).

Holtec Response: The differential equation method has been applied to the HI-STAR HB impact limiters, and a table of results has been incorporated into Supplement 2.I. This table replaces the previous table providing results using LS-DYNA. A supporting calculation package has also been created and is submitted to the Staff as part of the revised submittal. This new calculation provides back-up documentation of the tabular results reported in Supplement 2.I. The values for HI-STAR HB impact limiter crush strength used in the calculations are consistent with the version of drawing 5014-C1765 (Sheet 1 of 7) submitted with this RAI response. The results show that the most damaging rigid body cask deceleration for the HI-STAR HB is bounded by the design basis 60 g deceleration for the HI-STAR 100 cask. Finally, all references to LS-DYNA results are removed from Supplement 2.I.

The Hexcel manufacturer's catalog states that dynamic crush strengths are a function of initial impact velocity only; there is no information suggesting that the "Z" factors in the differential equation method are a function of crush material density. Therefore, the calculation results for the HI-STAR HB impact limiters use the same "Z" factor function of velocity that was used for the HI-STAR 100.

2-5 Justify that the spent fuel inspection performed by PG&E at the Humboldt Bay ISFSI, adequately characterizes fuel defects and damage to the degree needed to ensure that fuel condition for transportation is known and that the fuel will not reconfigure during transportation.

10 CFR 71.33(b)(3) requires that the chemical and physical form of the content be specified. 10 CFR 71.55(d)(1) and (2) requires that under normal conditions of transport that: 1) the contents will be subcritical, and 2) the geometric form of the packaging content would not be substantially altered. Furthermore, 10 CFR 71.55(b) requires the system to stay subcritical under the most reactive credible configuration and with the cask fully moderated. Based on the lack of reactor records at Humboldt Bay indicating that the fuel assemblies were intact and the inability of the exterior four sided visual examination performed by PG&E to determine if the internal rods of the assembly are grossly beached or have had a weakening of the cladding due to an interaction of the pool water and the UO_2 material, there is inadequate assurance that the condition of the fuel is known and that the cited requirements can be satisfied without appropriate measures taken to load the fuel as damaged per the proposed SAR definition.

A revised definition of an undamaged fuel assembly, for example: all the exterior rods in the assembly visually being shown to be intact while the interior rods being of unknown condition, as allowed by the latitude in ISG-1 Rev 2, may satisfy regulatory requirements with commensurate supporting analyses. For example, a criticality analysis would consist of addressing where assemblies with intact outer rods could potentially have multiple damaged fuel rods in the inner fuel rod positions. Additionally, a revised shielding analysis would need to demonstrate that this condition is bounded.

This information is needed to satisfy the criteria of 10 CFR 71.33 and 71.55.

Holtec Response: To account for the limited inspection of the fuel, an additional fuel classification of "undamaged" fuel has been defined for the HB fuel in the SAR, and corresponding analyses or evaluations have been added to qualify this fuel to be loaded in place of intact fuel. The definition recognizes the fact that the fuel rods on the inside of the assemblies are present, but that the cladding of these rods is of unknown condition. The criticality analyses were updated, since the undamaged fuel could potentially show a higher maximum k_{eff} than the intact fuel. The shielding evaluation has also been updated to show that the condition is already bounded by the current evaluations, since it assumed a compaction of all assemblies in the basket. Other evaluations were also reviewed and it was determined that the effect of this condition is negligible.

Chapter 3 Thermal Review

3-1 Remove statements from the application that the thermal properties of Holtite-A won't change over time, or explain the changes and any resulting impact it would have on the Hi-Star 100 thermal analysis.

Staff expects that the performance of the periodic thermal test will continue to be included in the technical specifications unless or until additional testing described as follows have been performed. Holtite-A is a polymer and, as such, is typically susceptible to heat and radiation degradation. There is no direct correlation between the testing documented in the reports submitted by Holtec (see the referenced reports below) and providing assurance that the thermal conductivity of Holtite-A does not change with time. The physical properties monitored during the Holtec testing are not directly related to thermal conductivity. In addition, the tested samples should be exposed first to thermal aging and then to irradiation aging, to measure the combined effect, rather than aging the samples independently and only looking at the singular effect of aging (i.e., either radiation or thermal). As a result, the staff requests that Holtec perform testing to determine the time dependent effect of radiation and heat on the thermal conductivity of Holtite-A.

Reference reports:
1) "Holtite-A: Results of Pre- and Post- Irradiation Tests and Measurements," HI-2002420, Rev. 1, dated 4/8/03.
2) "Holtite-A Development History and Thermal Performance Data," HI-2002396, Rev. 3, dated 4/10/03.

This information is needed to confirm whether the package design and contents complies with 10 CFR 71.33.

Holtec Response:
Holtec will reinstate the requirements for a periodic thermal test in SAR Section 8.2.5 as was previously required and described in CoC Condition 6.(b)(8).

Chapter 6 Criticality Review

6-1 Revise Note 16 on Drawing Number 4103, Sheet 1, Revision 3, to ensure that, for two pieces, the combined free space at the top, bottom, and gap between the poison plate segments totals to no more than ¼ inch to assure proper control of the potential size of the gap between the two plate segments.

Section 6.4.13 states that during the manufacture of the spent nuclear fuel (SNF) basket, a maximum gap of ¼ inch is permitted between neutron absorbing plates.

This information is needed to confirm compliance with 10 CFR 71.55 and 10 CFR 71.59.

Holtec Response: It has been confirmed by a review of QA records that all of the fuel baskets manufactured for use at Humboldt Bay have full length, full width Metamic panels; therefore Holtec considers Note 16, referring to two-piece panel construction, as well as Note 17, which addresses reductions in panel width, unnecessary by now. These notes are therefore removed from licensing drawing 4103. Accordingly, the analysis presented in subsection 6.4.13 has also been removed from the SAR.

6-2 Use sufficiently conservative boron density assumptions to estimate K_{eff}. Provide additional justification supporting the conclusions presented in Section 6.4.14 which demonstrate that inhomogeneities of less than 8 cm in length in the neutron poison plates will not have an adverse effect on Keff.

It may be unreasonable to assume an area of a neutron poison plates that is deficient in boron is located adjacent to an area of a neutron absorber plates that exceeds the minimum required content. Similarly, it may be non-conservative to assume that each individual neutron absorber (plate or region) consists of alternating boron-rich and boron-poor regions. The latter assumption may be considered to be valid, however, if it can be demonstrated that the configuration is the most conservative.

Additionally, the analysis presented in Section 6.4.14 assumes that boron-poor regions of the neutron poison plates have no less than 80% of the minimum boron content. It is unclear why a boron-poor region in a METAMIC plate could not contain less than 80% of the minimum required boron, taking thinning into account.

This information is needed to determine compliance with 10 CFR 71.55(d)(1) and 10 CFR 71.55(e).

Holtec Response: The QA records for the panels used in the MPC-HB baskets show that all panels meet the minimum requirements. Subsection 6.1.14 has therefore been removed.

6-3 Revise Note 10 on Drawing Number 4103, Sheet 1, Revision 3, to explicitly prohibit poison plates with manufacturing damage and (or including) weld-related or any other damage greater than the equivalent of a 1" diameter hole in each panel. Provide a note consistent with Note 16 of this same drawing.

This information is needed to confirm compliance with 10 CFR 71.55 and 71.59.

Holtec Response: This note is consistent with the note on the approved generic fuel basket drawings (3925 Rev. 5, 3926 Rev. 5, 3927 Rev. 6, and 3928 Rev. 5). Nevertheless, Holtec has added the further clarification to explicitly prohibit plates with damage greater than the equivalent of a 1" diameter hole.

6-4 If it is possible to load more than the specified number of fuel rods (2) with an initial maximum rod enrichment greater than that stated in Table A.3 (≤ 4.0 wt.%), revise the criticality analyses to address loading more than two of these fuel rods as either intact or damaged rods. (See question TS-1.)

This information is needed to confirm compliance with 10 CFR 71.55 and 71.59.

Holtec Response: Please refer to the Holtec Response to TS-1 which confirms that there are only two high power test rods remaining at the HBPP.

Chapter 7 Package Operations

7-1 Revise the descriptions of package operations in Chapter 7 of the SAR as follows:

a. Revise Step #2 of Section 7.1.3.1 to have the receipt inspection of the MPC and removal of road dirt/debris and any foreign material performed prior to the other activities described in this step. The activities should be described in the sequence in which they are to be performed; the current text does not do this.
b. Include language in Step #1 of Section 7.1.5 that states leak testing will also be done after de-tensioning one or more overpack lid bolts, the drain port, or the vent port plug to be consistent with the conditions for which testing is required in Step #6 of Section 7.1.4.
c. Change the ALARA note on page 7.2-1 to discuss the HI-STAR 100 and not the HI-STAR 60.
d. Remove Section 7.I.1.1.2 from the SAR. This section for operations with the HI-STAR HB refers to Section 7.1.3.2 (dry loading), which the applicant deleted in response to the staff's previous RAI.
e. Add the description that pocket trunnions, if present and not in use, are plugged to meet 10 CFR 71.87(h) in Section 7.1.5.

This information is needed to confirm compliance with 10 CFR 71.87.

Holtec Response:
The modifications requested in this RAI have been incorporated into the proposed SAR, Revision 13b.

Chapter 8 Acceptance Tests and Maintenance Program

8-1 Revise the acceptance tests for the neutron shielding in Section 8.1.5.4 of the proposed amendment to include the shielding effectiveness tests in Section 8.1.5.2 of the currently approved SAR.

The current amendment proposes to use the radiological surveys performed in Chapter 7, "Operating Procedures" prior to transport to constitute acceptance tests for the neutron shielding material. This proposal does not meet the purpose of acceptance tests, as the pre-shipment radiological surveys only ensure that the Part 71 dose rate limits are met for a particular shipment. Acceptance tests verify that the as fabricated neutron shielding performs, for approved contents, as designed. This verification involves comparison of dose rate measurements for a given contents with the values calculated for the same contents and would be performed prior to the first shipment. The shielding effectiveness tests described in Section 8.1.5.2 of the currently approved SAR fulfill the purpose of acceptance tests. Further guidance regarding shielding acceptance tests is contained in NUREG/CR-3854, "Fabrication Criteria for Shipping Containers."

This information is needed to confirm compliance with 10 CFR 71.85(c).

Holtec Response:
Section 8.1.5.4 of the proposed SAR, Revision 13b, has been revised to include the shielding effectiveness tests described in Section 8.1.5.2 of the currently approved SAR.

8-2 Revise Section 8.2.4 of the "Maintenance Program" for shielding to include Condition 6.(b)(6) of the currently approved CoC for periodic verification of the neutron shield integrity.

The current amendment proposes to use the pre- and post-transport radiological surveys, conducted as part of Chapter 7, "Package Operations," to demonstrate continued shield integrity and efficacy. However, these Chapter 7 radiological surveys only ensure the package meets the Part 71 dose rate limits for a particular shipment. The maintenance tests should verify that the neutron shield performs as designed for approved contents, which verification involves comparison of dose rate measurements for given contents with values calculated for the same contents. This verification also accounts for potential degradation of the Holtite-A neutron shield during the service life of the package. The tests in the currently approved CoC Condition 6.(b)(6) fulfill the purpose of maintenance tests for shielding.

This information is needed to confirm that the maintenance program is adequate to assure that packaging effectiveness is maintained throughout the packaging's service life to ensure continuing compliance with 10 CFR Part 71, Subpart E.

Holtec Response:
Section 8.2.4 of the proposed SAR, Revision 13b, was revised to contain Condition 6.(b)(6) of the current CoC for periodic verification of the neutron shield integrity.

LAR 9261-5

ATTACHMENT 2
to
Holtec Letter 5014666

PROPOSED CoC CHANGES -Updated

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION	c. DOCKET	d. PACKAGE IDENTIFICATION	PAGE		PAGES
9261	~~5~~*TBD*	71-9261	USA/9261/B(U)F-~~85~~ *96*	1	OF	10

2. PREAMBLE

a. This certificate is issued to certify that the package (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 any 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 Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)* Revision ~~12~~ *TBD*, dated ~~October 9, 2006~~*TBD*.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below. *The HI-STAR 100 System includes the HI-STAR 100 Version HB (also referred to as the HI-STAR HB).*

Multi-Purpose Canister

There are ~~six~~ *seven* Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, ~~and~~ MPC-68F*, and the MPC-HB*. All MPCs are designed to have identical exterior dimensions, except ~~those~~ *1)* MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design*; and 2) MPC-HBs custom-designed for the Humboldt Bay plant, which are approximately 6.3 feet shorter than the generic Holtec MPC designs.* ~~A single overpack design is provided that is capable of containing each type of MPC.~~ The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to 32 intact PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel

5.(a) (2) Description (continued)

assemblies. ~~BWR fuel debris may be shipped only in the MPC-68F.~~ *The MPC-HB is designed to contain up to 80 Humboldt Bay BWR fuel assemblies.*

~~PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24EF.~~

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. *The outer diameter of the Humboldt Bay MPCs is the same as the generic MPC, but the height is approximately 6.3 feet shorter than the generic MPC design. The Humboldt Bay MPCs are transported in a shorter version of the HI-STAR overpack, designated HI-STAR HB.* The fuel basket designs vary based on the MPC model. ~~For the HI-STAR 100 System transporting fuel debris in a MPC-68F or MPC-24EF, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a welded enclosure constructed entirely of a stainless steel alloy.~~

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision ~~12~~ *TBD*:

(a) HI-STAR 100 Overpack Drawing 3913, Sheets 1-9, Rev. ~~7~~*9*

5.(a) (3) Drawings (continued)

(b)	MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. ~~14~~*16*
(c)	MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 5
(d)	MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev. 5
(e)	MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 5
(f)	HI-STAR 100 Impact Limiter ~~CoC No. 9261, Appendix B~~	Drawing C1765, Sheet~~s~~ 1, *Rev. 4;* ~~and~~ *Sheet* 2, Rev. ~~2~~*3*; Sheet 3, Rev. ~~1~~*4*, Sheet 4, Rev. ~~2~~*4*; ~~and~~ Sheet~~s~~ 5, *Rev. 2; Sheet* ~~7~~ *6*, Rev. ~~1~~*3; and Sheet 7, Rev. 1*
(g)	HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. ~~1~~*2*
(h)	Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i)	Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j)	Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k)	Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 7
(l)	~~HI-STAR 100 MPC-32~~ *MPC-32 Fuel Basket Assembly*	Drawing 3927, Sheets 1-4, Rev. 6
(m)	*HI-STAR HB Overpack*	*Drawing 4082, Sheets 1-7, Rev. 3*
(n)	*MPC-HB Enclosure Vessel*	*Drawing 4102 Sheets 1-4, Rev. 1*
(o)	*MPC-HB Fuel Basket*	*Drawing 4103 Sheets 1-3 Rev. 5*
(p)	*Damaged Fuel Container HB*	*Drawing 4113 Sheets 1-2 Rev. 1*

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(i) below are authorized for transportation.

5.(b)(1) Contents (continued)

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by *a* review of records, greater than pinhole leaks or hairline cracks, ~~missing~~ *empty* fuel rod *location*s that are not ~~replaced~~ *filled* with dummy fuel rods, missing structural components such as grid spacers, ~~assemblies~~ whose structural integrity ha~~ve~~s been impaired *such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations*, or ~~those~~ that cannot be handled by normal means. Fuel assemblies ~~which~~ *that* cannot be handled by normal means due to fuel cladding damage are considered ~~fuel debris~~*FUEL DEBRIS*.

Damaged Fuel Containers (or Canisters)(DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10, 1.2.11, *and 1.I.1* of the HI-STAR 100 System SAR, Rev. ~~12~~*TBD*.

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, *including containers and structures supporting these parts*. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s). Trojan fuel assemblies not loaded into DFCs or FFCs are classified as intact assemblies.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TDPs), and Rod Cluster Control Assemblies (RCCAs).

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

5.(b)(1)(b) Definitions (continued)

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System SAR, Rev. ~~12~~*TBD*.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. ~~12~~*TBD*.

Undamaged fuel assemblies *are fuel assemblies where all the exterior rods in the assembly are visually inspected and shown to be intact. The interior rods of the assembly are in place; however the cladding of these rods is of* ***unknown*** *condition. This definition only applies to Humboldt Bay fuel assembly array/class 6x6D and 7x7C.*

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the damaged fuel assemblies or the intact fuel assemblies.

5.(b)(1) Contents (continued)

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

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

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR and the values shall be compared against limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC-32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.1 of the SAR.

5.(c) Criticality Safety Index (CSI)= 0.0

6. ~~For operating controls and procedures, i~~ *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 *provided in Chapter 7 of the HI-STAR SAR.*

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

~~(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.~~

~~(3) The package must satisfy the following leak testing requirements:~~

~~(a) All overpack containment boundary seals shall be leak tested to show a total leak rate of not greater than 4.3 x 10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15 x 10^{-6} atm cm^3/sec (helium) and shall be performed:~~

~~(i) within the 12-month period prior to each shipment;~~
~~(ii) after detensioning one or more overpack lid bolts, drain port, or the vent port plug; and~~
~~(iii) after each seal replacement.~~

(b) ~~Within 30 days before each shipment, all overpack containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1 x 10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.(a)(3)(a) above.~~

(c) ~~Each overpack containment boundary seal must be replaced after each use of the seal.~~

(4) ~~The relief devices on the neutron shield vessel shall be replaced every 5 years.~~

(5) ~~MPC-68F and MPC-24EF shall be leak tested prior to shipment to show a leak rate of no greater than 5 x 10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5 x 10^{-6} atm cm^3/sec (helium).~~

(6) ~~MPCs deployed at an ISFSI under 10 CFR Part 72 prior to transportation may be dried using the vacuum drying method or the Forced Helium Dehydration (FHD) method. MPCs placed directly into transportation service under 10 CFR 71 without first being deployed at an ISFSI must be dried using the FHD method. Water and residual moisture shall be removed from the MPC in accordance with the following specifications:~~

~~For those MPCs vacuum dried:~~

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

(b) ~~The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.~~

~~For those MPCs dried using the FHD System:~~

(a) ~~Following bulk moisture removal, the temperature of the gas exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes~~

(7) ~~Following drying, the MPC shall be backfilled with 99.995% minimum purity helium: > 0 psig and ≤ 44.8 psig at a reference temperature of 70 °F~~

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

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

(b) ~~The overpack annulus shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.~~

(9) ~~Following vacuum drying, the overpack shall be backfilled with helium to ≥ 10 psig and ≤ 14 psig.~~

(10) ~~The following fasteners shall be tightened to the torque values specified below:~~

~~Fastener~~	~~Torque (ft-lbs)~~
~~Overpack Closure Plate Bolts~~	~~2895 ± 90~~
~~Overpack Vent and Drain Port Plugs~~	~~45 +5/-2~~
~~Top Impact Limiter Attachment Bolts~~	~~256 +10/-0~~
~~Bottom Impact Limiter Attachment Bolts~~	~~1500 +45/-0~~

~~(11)~~ ~~Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.~~

~~(12)~~ ~~Appropriate monitoring for combustible gas concentration shall be performed prior to, and during MPC lid welding and weld cutting operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding and weld cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.~~

(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 *provided in Chapter 8 of the HI-STAR SAR.*

~~(1)~~ ~~The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.~~

~~(2)~~ ~~The MPC shall be pressure tested in accordance with ASME Section III, Subsection NB, Article NB-6110 and applicable sub-articles. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of the design pressure. The minimum hydrostatic test pressure shall be 125 psig. If pneumatic testing is used, the MPC shall be pressure tested to 120% of the design pressure. The minimum pneumatic test pressure shall be 120 psig.~~

~~(3)~~ ~~The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.~~

~~(4)~~ ~~The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection results, including all relevant indications shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.~~

~~(5)~~ ~~The radial neutron shield shall have a minimum thickness of 4.3 inches. 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 prior to each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.~~

~~(7)~~ ~~The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 5.(a)(3) of this Certificate of Compliance. 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 within 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-24E, MPC-24EF, and MPC-68, and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques.~~

~~(10)~~ ~~Flux trap sizes:~~

~~(a)~~ ~~The minimum flux trap size for the MPC-24 is 1.09 inches.~~

~~(b)~~ ~~The minimum flux trap sizes for the generic MPC-24E and MPC-24EF are 0.776 inch for cells 3, 6, 19, and 22; and 1.076 inch for the remaining cells.~~

~~(c)~~ ~~The minimum flux trap sizes for the Trojan MPC-24E and MPC-24EF are 0.526 inch for cells 3, 6, 19, and 22; and 1.076 inch for the remaining cells.~~

~~(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 14.5-1997.~~

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000

pounds, *except for the HI-STAR HB, where the gross weight shall not exceed 187,200 pounds.*

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

9. The personnel barrier shall be installed at all times while transporting a loaded overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

11. Revision ~~4~~ *5* of this certificate may be used until ~~October 12, 2007~~ *TBD*.

12. Expiration Date: March 31, 2009

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision ~~12~~*TBD*, dated ~~October 9, 2006~~*TBD*.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

TBD, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: ~~October 12, 2006~~*TBD*

APPENDIX A
CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 5*TBD*

MODEL NO. HI-STAR 100 SYSTEM

INDEX TO APPENDIX A

Page:	Table:	Description:
Page A-1 to A-~~19~~ *21*	Table A.1	Fuel Assembly Limits
Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3 with or without Zircaloy channels.
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6X6C, 7x7A, or 8x8A.
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-9		MPC-68F: 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.

INDEX TO APPENDIX A

Page:	Table:	Description:
A-10	Table A. 1 (Cont'd)	MPC-68F: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-11		MPC-68F: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-12		MPC-68F: 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.
A-13		MPC-68F: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters.
A-15		MPC-24E: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-16		MPC-24E: Trojan plant damaged fuel assemblies.
A-17		MPC-24EF: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-18		MPC-24EF: Trojan plant damaged fuel assemblies.
A-19		MPC-24EF: Trojan plant Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris.
A-20 to A-21		MPC-32: Uranium Oxide, PWR intact fuel assemblies in array class 15x15 D, E, F, and H and 17x17 A, B, and C listed in Table A.2.
A-22 to A-23		*MPC-HB:Uranium oxide, INTACT and/or UNDAMAGED FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C.*
A-2~~0~~4 to A-2~~3~~7	Table A.2	PWR Fuel Assembly Characteristics
A-2~~4~~*8* to A-~~28~~*33*	Table A.3	BWR Fuel Assembly Characteristics

INDEX TO APPENDIX A

Page:	Table:	Description:
A-~~29~~*34*	Table A.4	Fuel Assembly Cooling, Average Bumup, and ~~Minimum~~Initial Enrichment MPC-24/24E/24EF PWR Fuel with Zircaloy Clad and With Non-Zircaloy In-Core Grid Spacers.
A-~~29~~*34*	Table A.5	Fuel Assembly Cooling, Average Bumup, and ~~Minimum~~Initial Enrichment MPC-24/24E/24EF PWR Fuel with Zircaloy and with Zircaloy In-Core Grid Spacers.
A-3~~0~~*5*	Table A.6	Fuel Assembly Cooling, Average Burnup, and ~~Minimum~~Initial Enrichment MPC-24/24E/24EF PWR Fuel with Stainless Steel Clad.
A-3~~0~~*5*	Table A.7	Fuel Assembly Cooling, Average Bumup, and ~~Minimum~~Initial Enrichment MPC-68.
A-3~~1~~*6*	Table A.8	Trojan Plant Fuel Assembly Cooling, Average Burnup, and ~~Minimum~~Initial Enrichment Limits.
A-3~~1~~*6*	Table A.9	Trojan Plant Non-Fuel Hardware and Neutron Source Cooling and Burnup Limits.
A-3~~4~~7	Table A.10	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-32 PWR fuel with Zircaloy Clad and Non-Zircaloy In-Core Grid Spacers
A-3~~4~~7	Table A.11	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-32 PWR Fuel with Zircaloy Clad and Non-Zircaloy Grid Spacers
A-3~~5~~8	Table A.12	Fuel Assembly Maximum Enrichment and Minimum Burnup Requirement for Transportation in MPC-32
A-3~~6~~9	Table A.13	Loading Configurations for the MPC-32
A-3~~2~~9		References

Table A.1 (Page 1 of ~~19~~*21*)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

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

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (Page 2 of ~~19~~*21*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, *except assembly classes 6x6D and 7x7C,* with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.~~8~~*45* wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	≤272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat ≤183.5 Watts.
ii SS Clad:	≤83 Watts
f. Fuel assembly length:	≤ 176.2 inches (nominal design)
g. Fuel assembly width:	≤ 5.85 inches (nominal design)
h. Fuel assembly weight:	≤ 700 lbs, including channels

Table A.1 (Page 3 of ~~19~~*21*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.~~8~~*45* wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 4 of ~~19~~*21*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 5 of ~~19~~*21*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 6 of ~~19~~*21*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 10) and meeting the following specifications:

a.	Cladding type:	ZR
b.	Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c.	Number of rods per Thoria Rod Canister:	$\leq$ 18
d.	Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e.	Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f.	Initial heavy metal weight:	$\leq$ 27 kg/canister
g.	Fuel cladding O.D.:	$\geq$ 0.412 inches
h.	Fuel cladding I.D.:	$\leq$ 0.362 inches
i.	Fuel pellet O.D.:	$\leq$ 0.358 inches
j.	Active fuel length:	$\leq$ 111 inches
k.	Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table A.1 (Page 7 of ~~19~~*21*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum 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~~*45* wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

Table A.1 (Page 8 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.~~8~~*45* wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 9 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original 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~~*45* wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 10 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 11 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 550 lbs, including channels

Table A.1 (Page 12 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original 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 in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels

Table A.1 (Page 13 of ~~19~~*21*)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 10) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1 (Page 14 of ~~19~~*21*)
Fuel Assembly Limits

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

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 15 of ~~19~~*21*)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1 (Page 16 of ~~19~~*21*)

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class.
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A.1 (Page 18 of ~~19~~*21*)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 19 of ~~19~~*21*)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum initial enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 19 of ~~19~~*21*)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding Type:	ZR
b. Maximum initial enrichment	As specified in Table A.2 for the applicable fuel assembly array/class
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference):	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly	≤ 625 Watts
f. Fuel assembly length	≤ 176.8 inches (nominal design)
g. Fuel assembly width	≤ 8.54 inches (nominal design)
h. Fuel assembly weight	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A.1 (Page 19 of ~~19~~*21*)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32 (continued)

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies

C. Fuel assebliės shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.1 (Page 20 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-HB

A. Allowable Contents

1. Uranium oxide, INTACT and/or UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C and the following specifications:

a. Cladding type:	*ZR*
b. Maximum planar-average initial enrichment:	*As specified in Table A.3 for the applicable fuel assembly array/class.*
c. Initial maximum rod enrichment:	*As specified in Table A.3 for the applicable fuel assembly array/class.*
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	*An assembly post-irradiation cooling time ≥ 29 years, an average burnup ≤ 23,000 MWD/MTU, and a minimum initial enrichment ≥ 2.09 wt% ^{235}U.*
e. Fuel assembly length:	*≤ 96.91 inches (nominal design)*
f. Fuel assembly width:	*≤ 4.70 inches (nominal design)*
g. Fuel assembly weight:	*≤ 400 lbs, including channels and DFC*
h. Decay heat per assembly	*≤ 50 W*
i. Decay heat per SFSC	*≤ 2000 W*

Table A.1 (Page 21 of 21)
Fuel Assembly Limits

VI. MPC MODEL: MPC-HB (continued)

B. Quantity per MPC-HB: Up to 80 fuel assemblies

C. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS must be stored in a DAMAGED FUEL CONTAINER. Allowable Loading Configurations: Up to 28 DAMAGED FUEL ASSEMBLIES/FUEL DEBRIS, in DAMAGED FUEL CONTAINERS, can be stored in the peripheral fuel storage locations as shown in SAR Figure 6.I.3, or up to 40 DAMAGED FUEL ASSEMBLIES/FUEL DEBRIS, in DAMAGED FUEL CONTAINERS, can be stored in a checkerboard pattern as shown in SAR Figure 6.I.4. The remaining fuel storage locations may be filled with INTACT and/or UNDAMAGED FUEL ASSEMBLIES meeting the above applicable specifications, or with INTACT and/or UNDAMAGED FUEL ASSEMBLIES stored in DAMAGED FUEL CONTAINERS.

NOTE 1: Fuel assemblies with channels may be stored in any fuel cell location.

NOTE 2: The total quantity of damaged fuel or fuel debris permitted in a single DAMAGED FUEL CONTAINER is limited to the equivalent weight and special nuclear material quantity of one intact assembly.

NOTE 3: FUEL DEBRIS includes material in the form of loose debris consisting of zirconium clad pellets, stainless steel clad pellets, unclad pellets or rod segments up to a maximum of one equivalent fuel assembly. A maximum of 1.5 kg of stainless steel clad is allowed per cask.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

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

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

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

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

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

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

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

2. ZR Designates cladding material made of Zirconium or Zirconium alloys.

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

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

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

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

Table A.3 (Page 1 of ~~5~~*6*)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

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

Table A.3 (Page 2 of ~~5~~*6*)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

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

Table A.3 (Page 3 of ~~5~~*6*)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9B**	**9x9C**	**9x9D**	**9x9E (Note 13)**	**9x9F (Note 13)**	**9x9G**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of ~~5~~6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	***6x6D***	***7x7C***
Clad Material (Note 2)	*ZR*	*ZR*
Design Initial U (kg/assy.) (Note 3)	*≤ 78*	*≤ 78*
Maximum planar-average initial enrichment (wt.% ^{235}U)	*≤ 2.6*	*≤ 2.6*
Initial Maximum Rod Enrichment (wt.% ^{235}U)	*≤ 4.0*	*≤ 5.5*
No. of Fuel Rod Locations	*36*	*49*
Fuel Clad O.D. (in.)	*≥ 0.5585*	*≥ 0.486*
Fuel Clad I.D. (in.)	*≤ 0.505*	*≤ 0.426*
Fuel Pellet Dia. (in.)	*≤ 0.488*	*≤ 0.411*
Fuel Rod Pitch (in.)	*≤ 0.740*	*≤ 0.631*
Active Fuel Length (in.)	*≤ 80*	*≤ 80*
No. of Water Rods (Note 11)	*0*	*0*
Water Rod Thickness (in.)	*N/A*	*N/A*
Channel Thickness (in.)	*≤ 0.060*	*≤ 0.060*

Table A.3 (Page ~~5~~*6* of ~~5~~*6*)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

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

2. ZR designates cladding material made from Zirconium or Zirconium alloys.

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

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

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

6. Square, replacing nine fuel rods.

7. Variable

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

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

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

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

12. This assembly is known as "QUAD+" and 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~~INITIAL ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND ~~MINIMUM~~INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 11	≤ 39,500	≥ 3.2
≥ 14	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND ~~MINIMUM~~INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Initial Enrichment (wt. % U-235)**
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND ~~MINIMUM~~INITIAL ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Initial Enrichment (wt. % U-235)**
≥ 5	*≤ 10,000*	*≥ 0.7*
≥ 7	*≤ 20,000*	*≥ 1.35*
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

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

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

NOTES:

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

Table A.9

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

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

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP,AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (Wt % U-235)
≥ 12	≤ 24,500	≥ 2.3
≥ 14	≤ 29,500	≥ 2.6
≥ 16	≤ 34,500	≥ 2.9
≥ 19	≤ 39,500	≥ 3.2
≥ 20	≤ 42,500	≥ 3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (Wt % U-235)
≥ 8	≤ 24,500	≥ 2.3
≥ 9	≤ 29,500	≥ 2.6
≥ 12	≤ 34,500	≥ 2.9
≥ 14	≤ 39,500	≥ 3.2
≥ 19	≤ 44,500	≥ 3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	Configuration (Note 2)	Maximum Enrichment (wt.% U-235)	Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)
15x15D, E, F, and H	A	4.65	$B=(1.6733)*E^3-(18.72)*E^2+(80.5967)*E-88.3$
	B	4.38	$B=(2.175)*E^3-(23.355)*E^2+(94.77)*E-99.95$
	C	4.48	$B=(1.9517)*E^3-(21.45)*E^2+(89.1783)*E-94.6$
	D	4.45	$B=(1.93)*E^3-(21.095)*E^2+(87.785)*E-93.06$
17x17A, B, C	A	4.49	$B=(1.08)*E^3-(12.25)*E^2+(60.13)*E-70.86$
	B	4.04	$B=(1.1)*E^3-(11.56)*E^2+(56.6)*E-62.59$
	C	4.28	$B=(1.36)*E^3-(14.83)*E^2+(62.27)*E-72.93$
	D	4.16	$B=(1.4917)*E^3-(16.26)*E^2+(72.9883)*E-79.7$

NOTES:

1. E= Initial enrichment (e.g., for 4.05 wt.%, E=4.05).

2. See Table A.13

3. Fuel assemblies must be cooled 5 years or more.

A.13

LOADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (interms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision ~~12~~ *TBD*, dated ~~October 9, 2006~~*TBD*.