May 8, 2009

Ms. Tammy Morin, Acting Licensing Manager
Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9261, REVISION NO. 7, FOR THE HI-STAR 100 SYSTEM

Dear Ms. Morin:

As requested by your application dated October 5, 2006, as supplemented June 29, July 27, August 3, September 27, October 5, and December 18, 2007; January 9, March 19, and September 30, 2008; and February 27, 2009, enclosed is Certificate of Compliance No. 9261, Revision No. 7, for the Model No. HI-STAR 100 System. This certificate supersedes, in its entirety, Certificate of Compliance No. 9261, Revision No. 6, dated February 13, 2009. Changes made to the enclosed certificate are indicated by vertical lines in the margin.

Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471. This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471. Registered users may request, by letter, to remove their names from the Registered Users List.

If you have any questions regarding this certificate, please contact me or Kim Hardin of my staff at (301) 492-3339.

Sincerely,

/RA/

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9261
TAC No. L24029

Enclosures: 1. Certificate of Compliance No. 9261, Rev. No. 7
2. Safety Evaluation Report
3. Registered Users

cc w/encls 1 & 2: R. Boyle, Department of Transportation
James M. Shuler, Department of Energy
Registered Users

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OFC	SFST	E	SFST	E	SFST	E	SFST	E
NAME	KHardin		MDeBose		RParkhill		DTarantino	
DATE	4/28/09		5/6/09		5/5/09		5/1/09	
OFC	SFST	C	SFST		SFST		SFST	
NAME	DTang		MCall		NJordan		LCampbell	
DATE	**4/29/09**		5/6/09		5/6/09		5/6/09	
OFC	SFST	E	SFST	E	SFST	E	SFST	E
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DATE	5/7/09		5/7/09		5/8/09			

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SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR 100 Package
Certificate of Compliance No. 9261
Revision No. 7

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SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR HB
Certificate of Compliance No. 9261
Revision No. 7

SUMMARY

By application dated October 5, 2006, as supplemented June 29, July 27, August 3, September 27, October 5, and December 18, 2007; January 9, March 19, and September 30, 2008; and February 27, 2009, Holtec International (Holtec or the applicant) requested a revision to the 10 CFR Part 71 Certificate of Compliance (CoC) No. 9261 for the Model No. HI-STAR 100 system. The spent fuel cask that is the subject of this amendment request is contained in a site specific license granted by NRC to PG&E for storage of specific fuel at the Humboldt Bay Power Station (HB) site (Docket No. 72-27). Additional supporting changes also requested include incorporating Metamic as an approved neutron absorber and updating the cask identification to B(U)F-96 in accordance with 10 CFR 71.19(e). The staff informed Holtec of the acceptance of the transportation application for technical review by letter dated November 9, 2006.

Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes, per its evaluation described in this Safety Evaluation Report (SER), that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

The following sections summarize the applicant's change requests with respect to the packaging and its contents.

1.1 Packaging

With respect to the packaging, the applicant is proposing to:

- add the HI-STAR 100 Version HB (HI-STAR HB) for use at HB,
- change the Safety Analysis Report (SAR) and licensing drawings to add Metamic as a neutron absorber for use in the HI-STAR HB, and
- change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC and Chapter 1 of the SAR.

A new shorter version of HI-STAR 100 was designed specifically for use at HB because the fuel assemblies are shorter than typical Boiling Water Reactor (BWR) fuel assemblies. The design includes a HI-STAR HB overpack, the multiple-purpose canister for HB (MPC-HB), and the impact limiters. The shorter design results in a reduced gross weight. Additionally, a HB specific Damaged Fuel Container (DFC) has been designed.

For the addition of Metamic as a neutron absorber for the HI-STAR HB, the drawings specified Metamic and the minimum B-10 loading, Metamic was added to SAR text, a description of the

material was added, an updated criticality analysis was provided, and the acceptance testing requirements were provided in SAR Chapter 8.

To support the request to change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC, staff reviewed the nineteen issues considered in the rulemaking process that resulted in the revised 10 CFR Part 71 dated January 26, 2004. The staff evaluated the applicant's request, as described below.

- Issue 1, Changing Part 71 to the International Systems of Units (SI) Only.

  This proposal was not adopted in the final rule, and therefore no changes are needed in the package application or the CoC to conform to the new rule.

- Issue 2, Radionuclide Exemption Values.

  The final rule adopted radionuclide activity concentration values and consignments activity limits in TS-R-1 for the exemption from regulatory requirements for the shipment or carriage of certain radioactive low-level materials. In addition, the final rule adopted an exemption from regulatory requirements for certain natural material and ores containing naturally occurring radionuclides. Based on the design purpose of the package and the allowed contents specified in the certificate, this change is not applicable to the HI-STAR 100 package. Thus, no changes are needed to conform to the new rule.

- Issue 3, Revision of A_1 and A_2.

  The final rule adopted changes in the A_1 and A_2 values from TS-R-1, with the exception of two radionuclides. The A_1 and A_2 values were modified in TS-R-1 based on refined modeling of possible doses from radionuclides, and the NRC agreed that incorporating the latest in dosimetric modeling would improve transportation regulations. The applicant provided an updated containment analysis in Chapter 4 of the application incorporating the revised A_2 values, which are for radioactive material in normal form. In general, the A_2 values for radionuclides important to the containment requirements for spent fuel shipments were increased. Although the calculated maximum allowable leakage rates were increased, the applicant retained the maximum allowable leakage rate that is demonstrated for the cask through leak testing (4.3 E-6 atm cm^3/sec He). The staff agrees that the package meets the containment requirements of 10 CFR 71.51 considering the changes in the A_1 and A_2 values in Appendix A, Table A-1, of the revised 10 CFR Part 71.

- Issue 4, Uranium Hexafluoride (UF_6) Package Requirements.

  These changes are not applicable, since the package is not authorized for the transport of uranium hexafluoride. Therefore, no changes are needed to conform to the new rule.

- Issue 5, Criticality Safety Index (CSI).

  The final rule adopted the CSI requirement from TS-R-1. The applicant revised Chapters 1 and 6 to clearly distinguish between the CSI and the Transport Index (TI).

- Issue 6, Type C Packages and Low Dispersible Material.

  This proposal was not adopted for the final rule. Thus, no changes are necessary.

- Issue 7, Deep Immersion Test.

  The final rule adopted an extension of the previous version of 10 CFR 71.61 from packages for irradiated fuel to any Type B package containing activity greater than 10^5 A_2. Because the Model No. HI-STAR 100 package is designed to transport irradiated fuel, the applicant already complies with the deep immersion requirements of 10 CFR 71.61. Thus, no changes are necessary to conform to the new rule.

- Issue 8, Grandfathering Previously Approved Packages.

  The final rule adopted a process for allowing continued use, for specific periods of time, of a previously approved packaging design without demonstrating compliance to the final rule. The applicant has decided in accordance with 10 CFR 71.19(e) to submit information demonstrating compliance with the final rule. Thus, grandfathering the design of the package is not necessary.

- Issue 9, Changes to Various Definitions.

  The final rule adopted several revised and new definitions. These changes were adopted to provide clarity to Part 71. No change is necessary to conform to the new rule.

- Issue 10, Crush Test for Fissile Material Packages.

  The revised 10 CFR 71.73 expanded the applicability of the crush test to fissile material packages. The crush test is required for packages with a mass not greater than 500 kilograms (1100 pounds). Since the Model No. HI-STAR 100 package has a mass greater than this, the crush test is not applicable. Therefore, no change is necessary to conform to the new rule.

- Issue 11, Fissile Material Package Design for Transport by Aircraft.

  The final rule adopted a new section, Section 71.55(f), which addresses packaging design requirements for packages transporting fissile material by air. The package is not authorized for shipment by air, and this requirement is not applicable to the Model No. HI-STAR 100 package.

- Issue 12, Special Package Authorizations.

  The final rule adopted provisions for special package authorization that will apply only in limited circumstances and only to one-time shipments of large components. This provision is not applicable to the Model No. HI-STAR 100 package. Thus, no change is necessary to conform to the new rule.

- Issue 13, Expansion of Part 71 Quality Assurance (QA) Requirements to Certificate Holders.

The final rule expanded the scope of Part 71 QA requirements to apply to any person holding or applying for a CoC. QA requirements apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. The applicant must meet the QA program requirements of 10 CFR 71.101(a), (b), and (c). No change is needed to conform to the new rule.

- Issue 14, Adoption of the American Society of Mechanical Engineers (ASME) code.

  This proposal was not adopted in the final rule. Thus, no change is needed to conform to the new rule.

- Issue 15, Change Authority for Dual-Purpose Package Certificate Holders.

  This proposal was not adopted for the final rule. Thus, no change is necessary to conform to the new rule.

- Issue 16, Fissile Material Exemptions and General License Provisions.

  The final rule adopted various revisions to the fissile material exemptions and the general license provisions in Part 71 to facilitate effective and efficient regulation of the transport of small quantities of fissile material. The criticality safety of the Model No. HI-STAR 100 package does not rely on limiting fissile materials to exempt or generally licensed quantities. Chapter 6 of the package application demonstrates criticality safety of the package with the authorized fissile contents. Therefore, no change is necessary to conform to the new rule.

- Issue 17, Double Containment of Plutonium.

  The final rule removed the requirement that packages with plutonium in excess of 0.74 terabecquerel (20 curies) have a second separate inner container. Holtec revised the package application to remove references and discussions related to the second inner container requirement. Additionally, the requirement for helium leak testing has been removed, since the MPC is no longer a containment boundary. Further, the CoC has been revised to delete the limits based on previous double containment requirement in the following condition:

  Condition 5(a)(2), deleted the words “BWR fuel debris may be shipped only in the MPC-68F;” “PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24-EF;” and “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.”

- Issue 18, Contamination Limits as Applied to Spent Fuel and High Level Waste Packages.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 19, Modification of Events Reporting Requirements.

  The final rule adopted modified reporting requirements. While the final rule is applicable to the package, no change is needed to either the CoC or the package application to conform to the new rule.

"-96" Conclusion

Based on the statements and representations in the application, the staff concludes that the design has been adequately described and meets the requirements of the revised 10 CFR Part 71. Thus, the staff agrees that including the designation "-96" in the identification number is warranted. To allow time to modify the packaging markings to include the "-96" designation in the package identification number, the certificate has been conditioned to allow use of packagings marked with the "-85" designation for a period of approximately one year. After May 31, 2009, the packaging must be marked with the package identification number including the "-96" designation.

1.2 Contents

The applicant has requested the following additions or changes to the contents to:

- add the HB fuel as authorized contents,
- change the definition of Damaged Fuel Assembly, and
- add a definition of Undamaged Fuel Assembly.

Changes were made to incorporate the fuel specific to the HB plant into the CoC.

The damaged fuel definition is changed to:

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

The undamaged fuel definition added is:

> 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.

These changes were made for consistency with the corresponding definitions in the HI-STORM Storage CoC (72-1014).

1.3 Generic Changes

The applicant has proposed the following generic changes that are not specific to the HI-STAR HB:

- delete the contents of CoC Section 6.(a) and replace with a direct reference to SAR Chapter 7,
- in the revised Chapter 7, a change was made to the closure plate bolt torque-from 2895 ft.-lbs to 2000 ft.-lbs,
- delete the contents of CoC Section 6.(b) and replace with a direct reference to SAR Chapter 8,
- in the revised Chapter 8, dimensional and B-10 loading requirements that are already specified on the licensing drawings are not repeated.
- change the minimum enrichment from 1.8 wt% ^{235}U to 1.45 wt% ^{235}U in CoC Appendix A, Sections II.A.1.d.i, II.A.2.d, III.A.1.d, III.A.2.d, and III.A.3.d,
- add two assembly cooling times, burnups, and initial enrichments in CoC Appendix A, Table A.7,
- change drawing to allow the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The modification of CoC Conditions 6.(a) and 6.(b) to reference SAR Chapters 7 and 8 were made to remove unnecessary duplication and details from the CoC and SAR and provide consistency with other 10 CFR Part 71 CoCs. The structural evaluations in SAR Chapter 2 have been revised to support the closure plate bolt torque.

The changes to cooling times, burnups, and initial enrichments are in response to client needs. SAR Chapter 5 was updated to reflect these changes.

The structural evaluations were updated to show that the Safety Factors remain acceptable for both Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) for the allowance of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

There were other SAR changes that reflected a need to update the SAR and did not require a CoC change.

1.4 Drawings

The applicant has requested approval of changes to Drawing Nos. 3913, 3923, 3930, and 5014-C1765 and added Drawing Nos. 4082, 4102, 4103, and 4113. Drawing 3927 was updated to change its title.

The changes to Drawing No. 3913 consist of specifying all necessary details of the buttress plate attachment holes in one callout on sheet 5 rather than spread throughout the drawing. The tolerance on the attachment bolt circle was corrected from +/- 1/8” on Sheet 5 to +/-0.06”. The drawing changes corrected an editorial error in detailing the closure plate bolt threads as UNC. The drawing now reflects the actual threads (UN) that are used. The drawing also now reflects the allowance of the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The changes to Drawing No. 3923 include replacing the thru hole in the upper fuel spacer plate with a threaded hole; making the lower plate optional for PWR fuel with control components; changes to eliminate the potential for an undersized condition in the shell, baseplate, and lid assembly; change to the optional lid diameter for the MPC-68 as justified in the structural evaluation; delete the requirements for a secondary containment on plutonium shipments (as allowed by the 10 CFR Part 71 rule change in 2004); and eliminate some redundancy.

The changes to Drawing No. 5014-C1765 include changes to update the current licensing drawing to allow for the use of the HI-STAR HB impact limiters as supported in the structural evaluation and editorial changes.

Drawing Nos. 4082, 4102, 4103, and 4113 were added to include the needed packaging specifications for HB fuel.

The staff reviewed the revised set of licensing drawings and finds that the information on the drawings provides an adequate basis for its evaluation against 10 CFR Part 71 requirements. The information on the drawings is consistent with the package as described and evaluated in the SAR.

2.0 STRUCTURAL

Supplement 2.I to the application provides a structural evaluation of the HI-STAR HB package, a shortened version of HI-STAR 100. The organization of the supplement mirrors the format and content of Chapter 2 of the application for the approved HI-STAR 100 package, except it only contains material directly pertinent to the HI-STAR HB.

The staff reviewed the application to revise the Model No. HI-STAR 100 package structural design and evaluation to assess whether the package will remain within the allowable values or criteria for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 2 (Structural Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

2.1 Structural Design

2.1.1 Design Feature Changes

The HI-STAR HB package consists of three principal structural components: (1) the HI-STAR HB overpack, (2) the MPC-HB, and (3) the impact limiters.

The HI-STAR HB overpack, including materials, configurations of inner and intermediate shells, top and bottom forgings, and closure lids, is structurally identical to those of the HI-STAR 100

except for the shorter overall length, lower package weight, reduced strength of impact limiter crush materials, and smaller-diameter threads on lifting trunnions. Other exceptions, which are determined to have insignificant adverse effects on the structural performance, include the optional use of the stacked SA350 LF3 ring forgings to replace the 2.5-inch thick inner shell and the decreased number and length of enclosure shell radial gussets, which connect the outer and intermediate shells to form the cavity for the placement of Holtite-A neutron shield material.

The MPC-HB, which consists of an enclosure vessel (EV) and a fuel basket, is configured to hold up to 80 Humboldt Bay fuel assemblies. Except for shorter length, the structural features of the EV are identical to those with the other HI-STAR 100 MPCs, including materials, shell diameters, and dimensions of base plate, lid, and penetrations. The MPC-HB fuel basket, as a honeycombed structural weldment, is similar in construction to other approved HI-STAR 100 MPC basket assemblies. It is held inside the EV cavity against the angle and bar spacers welded to the EV shell.

The impact limiters of the HI-STAR HB package maintain identical design in form and dimensions to that of the HI-STAR 100 demonstrated structurally adequate for the 60-g design basis deceleration limits. The crush strengths of the aluminum honeycomb material are reduced, however, to accommodate lighter weight of the HI-STAR HB to ensure that the maximum cask decelerations remain to be bounded by the 60-g design basis limits for the 30-ft, free-drop, hypothetical accident conditions.

2.1.2 Design Criteria and Performance Overview

Section 2.I.0 of the supplement notes that the applicable codes, standards, and design criteria, including the design basis maximum cask decelerations of 60 g for the HI-STAR HB, are identical to those for the HI-STAR 100.

For overpack structural components other than the enclosure shell, the application argued that the reduced length and weight of the HI-STAR HB ensure that all stress-based evaluations performed on the HI-STAR 100 produced lower bound safety factors compared to those that would have been calculated for the HI-STAR HB. Sections 2.I.6.1 and 2.I.6.3 evaluate the modified enclosure shell for the heat and reduced external pressure conditions, respectively. The evaluation concluded and the staff agrees that there exist large stress safety factors against an internal pressure developing from off-gassing of the neutron shield material. The staff also concurs that the change associated with the radial gussets will have minor effects on the global response of the overpack subject to a lateral drop.

Recognizing that the MPC-HB has been demonstrated capable of withstanding a side impact deceleration of 60 g for the Part 72 license for the Humboldt Bay ISFSI, the application concluded, and the staff agrees that no new analyses of the MPC-HB are required as long as other design and test conditions, such as the heat and cold pressure/temperature design bases, remain unchanged from those for the HI-STAR 100 enclosure vessel.

On the basis of the above, the staff focuses its review primarily on the impact limiter crush strengths modification for ensuring that maximum cask decelerations remain to be bounded by the design bases.

2.2 Weights and Centers of Gravity

Table 2.I.2.1 of the supplement lists weights of the impact limiters at 26,000 lbs, the loaded MPC-HB at 59,000 lbs, the combined weight at 161,200 lbs for the overpack plus loaded MPC-HB, and the total package weight at 187,200 lbs. The center of gravity of the loaded package is 61.4 inches above the base of the cask.

2.3 Mechanical Properties of Materials

This Materials Evaluation Report is part of an NRC certification review of Holtec's Model No. HI-STAR 100 as a spent nuclear fuel transportation package under requirements specified in accordance with 10 CFR Part 71.

The changes proposed are changes that introduce the HB version of the HI-STAR system (HI-STAR HB and MPC-HB), generic changes that are necessary to support the HI-STAR HB (Metamic neutron absorber), but also apply to the other HI-STAR versions and minor changes not directly related to the HI-STAR HB. These minor changes do not affect the materials currently approved in the HI-STAR 100 System. Fuel assemblies for HB are shorter than typical BWR fuel assemblies; therefore, a shorter version of HI-STAR 100 was designed specifically for use at HB (i.e., one time transportation for use at HB only). The shorter design results in a reduced gross weight. Additionally, a HB specific DFC has been designed and fuel class arrays 6x6D and 7x7C have been characterized and analyzed for the HI-STAR HB.

The staff's materials evaluation of the proposed FSAR revisions is based on whether the applicant meets the applicable requirements of 10 CFR Part 71 for packaging and transportation of radioactive material. The evaluation focused on a brief review of the previously approved HI-STAR 100 System and the specific material modifications requested in the application. The objectives of this material review are to ensure adequate material properties exist.

2.3.1 Description

The following is a discussion of the HI-STAR 100 System (generic), the HI-STAR HB design, materials, and the changes proposed that introduce the HB version of the HI-STAR 100 system (HI-STAR HB and MPC-HB). Table 1.3.3 of the application, "Materials and Components of the HI-STAR 100 System," lists the specific material specifications. The HI-STAR 100 System was briefly reviewed for material properties generic to all components followed by a material review of specific material changes proposed that introduce the HB version HI-STAR HB and MPC-HB. No new determination on the adequacy of material properties was made unless it was used as the basis for the proposed revisions.

The HI-STAR 100 System in general consists of three primary components as follows: the multi-purpose canister (MPC), the overpack (HI-STAR) assembly and a set of impact limiters. The overpack confines the MPC and provides the containment boundary for transport conditions. The MPC is a hermetically sealed, welded cylinder with flat ends and an internal honeycomb fuel basket for storing and shipping spent nuclear fuel (SNF) within the overpack containment boundary. A set of impact limiters attached at both ends of the overpack provide energy absorption capability for NCT and HAC.

There are seven MPC models, all are designed to have similar exterior dimensions, one exception is custom designed for the HB plant (MPC-HB), which is approximately 6.3 feet

shorter than the generic Holtec MPC designs. Each corresponding fuel basket design varies based on the MPC model.

2.3.2 HI-STAR 100 Overpack (generic):

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel.

The overpack containment boundary is formed by an inner shell welded to a bottom plate and to a heavy top flange with a bolted closure plate (all SA 350-LF3, 3½% nickel alloy steel with good impact toughness for use at low temperatures). The closure plate is machined with two concentric grooves for seals (Alloy X750 – NiCrFe).

The outer surface of the overpack inner shell is reinforced with intermediate shells (SA 516-70, carbon steel for moderate and low temperature service with improved notch toughness) of gamma shielding that are installed to ensure a permanent state of contact between adjacent layers. These layers provide additional strength to the overpack to resist puncture or penetration. Radial channels (SA 515-70, carbon steel intended for intermediate or high temperature service) are vertically welded to the outside surface of the outermost intermediate shell at equal intervals around the circumference acting as fins improving heat conduction to the overpack outer enclosure shell surface and as cavities for retaining and protecting the neutron shielding (Holtite-A). An enclosure shell (SA 515-70) is formed by welding panels between each of the radial channels forming additional cavities. These panels together with the exterior flats of the radial channels form the overpack outer enclosure shell. The Holtite-A is placed into each of the radial cavity segments formed, the outermost intermediate shell, and the enclosure shell panels. Pressure relief devices (rupture disks) are positioned in a recessed area on top of the outer enclosure to relieve internal pressure that may develop as a result of a fire accident and subsequent off-gassing of the neutron shield material. A layer of silicone sponge is positioned within each radial channel acting as thermal expansion foam to compress as the neutron shield expands in the axial direction.

The exposed steel surfaces, except seal seating surfaces, of the overpack and the intermediate shell layers are coated with Thermaline 450 and Carboline 890 to prevent corrosion based on expected service conditions. The inner cavity of the overpack is coated with a material appropriate to its high temperatures and the exterior of the overpack is coated with a material appropriate for fuel pool operations and environmental exposure. The coating applied to the intermediate shells acts as a surface preservative and is not exposed to the fuel pool or ambient environment.

Lifting trunnions are manufactured from a high strength alloy (SB 637 - NiCrFe) and installed in threaded openings to the overpack top flange for lifting and rotating the cask body between vertical and horizontal positions. Pocket trunnions were eliminated from the original design and are no longer considered qualified tie-down devices. For transportation, the HI-STAR 100 System is engineered to be mounted on a transport frame secured to the transporter bed.

2.3.3 Multi-Purpose Canisters (generic):

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, a canister shell, a lid with vent/drain ports and cover plates, and a closure ring (all from Alloy X). Generic MPCs are interchangeable, which have identical exterior dimensions. MPC baskets are formed from an array of plates (Alloy X) welded to each other, such that a honeycomb structure is created.

A series of basket supports (Alloy X) are welded to the inside of the shell position and support the MPC fuel basket. Optional aluminum (Alloy 1100) heat conduction elements are installed in some early production models in the peripheral area formed by the basket, the MPC shell, and the basket supports. A refined thermal analysis has allowed making this aluminum design feature “optional.”

The MPC lid (Alloy X) is a circular plate (fabricated from one piece, or two pieces - split top and bottom) that is edge-welded to the MPC shell. Only the top piece is analyzed as part of the enclosure vessel pressure boundary if the two piece lid design is employed. The bottom piece acts primarily as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. The MPC lid is equipped with vent and drain ports that are used to remove moisture and gas from the MPC and backfill the MPC with a specified pressure of inert helium gas.

Holtec states that the free volume of the MPC and the annulus between the external surface of the MPC and the inside surface of the overpack containment boundary are filled with 99.995% pure helium gas during fuel loading operations. Following MPC drying operations, the MPC is backfilled with a pre-determined amount of helium gas. The helium backfill ensures adequate heat transfer and provides an inert atmosphere for fuel cladding integrity. The helium gas also provides conductive heat transfer across any gaps between the metal surfaces inside the MPC and in the annulus between the MPC and the overpack containment boundary. Metal conduction transfers the heat throughout the MPC fuel basket, through the MPC aluminum heat conduction elements (if installed) and shell, through the overpack inner steel shell, intermediate steel shells, steel radial connectors, and finally, to the outer neutron shield enclosure steel shell.

The closure ring (Alloy X) is a circular ring edge-welded to the MPC shell and MPC lid. The lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the threaded holes in the MPC lid during transfer from the storage only HI-STORM 100 System to the HI-STAR 100 overpack for transportation. Threaded insert plugs (Alloy X) are installed to provide shielding when the threaded holes are not in use.

MPCs are designed to store and transport intact fuel assemblies, damaged fuel assemblies, and fuel classified as fuel debris. Intact SNF can be placed directly into the MPC. Damaged SNF and fuel debris must be placed into a Holtec DFC for transportation inside the MPC and the HI-STAR 100 overpack.

HB damaged fuel and fuel debris will be transported in the MPC-HB.

All MPCs are constructed entirely from stainless steel alloy materials except for the neutron absorber and aluminum vent and drain cap seal washers with no carbon steel parts. All structural components in a HI-STAR 100 MPC will be fabricated of Alloy X. For the MPC design and analysis, any steel part in an MPC may be fabricated from any of the acceptable Alloy X

materials listed as follows, except that all steel pieces comprising the MPC shell must be fabricated from the same Alloy X stainless steel type: Type 316, Type 316LN, Type 304 and Type 304LN. Holtec states that the Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable and bounding for the entire group for the analysis in question.

2.3.4 Impact Limiters (generic):

Once the HI-STAR 100 overpack is positioned and secured in the transport frame the overpack is fitted with (ALSTAR) aluminum honeycomb impact limiters, one at each end. Impact limiters ensure the inertia loadings during NCT and HAC are maintained below design levels.

2.3.5 Shielding (generic):

To minimize personnel exposure the HI-STAR 100 System is provided with shielding. Initial attenuation of gamma and neutron radiation emitted by the radioactive SNF is provided by the MPC fuel basket structure built from inter-welded plates (Alloy X) and Boral neutron poison panels with sheathing (Alloy X) attached to the fuel cell walls. The MPC canister shell, baseplate, and lid provide additional thicknesses of steel to further reduce gamma radiation and to a lesser extent, neutron radiation at the outer MPC surfaces.

Primary HI-STAR 100 shielding is located in the overpack and consists of neutron shielding (Holtite-A) and additional layers of steel for gamma shielding. Gamma shielding is provided by the overpack inner, intermediate and enclosure steel shells with additional axial shielding provided by both the bottom steel plate and top closure steel plate. Impact limiters provide an increase in gamma shielding and provide additional distance from the radiation source at the ends of the package during transport. A circular segment of neutron shielding is contained within each impact limiter to provide neutron attenuation.

Both Boral and Metamic are neutron absorber materials made of B_4C and Aluminum. Metamic and Boral are both approved materials for the HI-STAR HB only.

2.3.6 Holtite-A Neutron Shielding (Overpack, generic):

Holtec International states Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A may be specified with a B_4C content of up to 6.5 wt%, however, Holtite-A is specified with a nominal B_4C loading (finely dispersed powder form) of 1 wt% percent for the HI-STAR 100 System. The nominal specific gravity of Holtite-A is 1.68 g/cm^3 and is reduced for shielding analysis by 4% to 1.61 g/cm^3 to conservatively bound and account for any potential weight loss at the design temperature and any inability to reach theoretical density. The nominal weight concentration of hydrogen is 6.0%. However, all shielding analyses will conservatively assume 5.9% hydrogen by weight in the calculations.

2.3.7 Gamma Shielding Material (Overpack, generic):

Carbon steel is used in successive layers of plate stock form with each layer of the intermediate shells constructed from two halves. Both halves of the shell are sheared, beveled, and rolled to the required radii. The two halves of the second layer are wrapped around the first shell. Each shell half is positioned in its location and while applying pressure using a uniquely engineered fixture, the halves are tack welded. The second layer is made by joining the two halves using two longitudinal welds. Successive layers are installed by repeating this process. The welding of every successive shell provides a certain inter-layer contact.

2.3.8 Coolants (generic):

No coolants are used, however, helium is sealed within the MPC internal cavity. The annulus between the MPC outer surface and overpack containment boundary is purged and filled with helium gas. The overpack annulus is backfilled with helium gas for heat transfer and seal testing. Concentric (Alloy X750 – NiCrFe) metallic seals in the overpack closure plate prevent leakage of the helium gas from the annulus and provide the containment boundary for the release of radioactive materials.

2.3.9 Chemical and Galvanic Reactions (generic):

Holtec states that no plausible mechanism exists for significant chemical or galvanic reactions in the HI-STAR 100 System during loading operations. The MPC, filled with helium, provides a non-aqueous and inert environment. Corrosion is a long-term time-dependent occurrence. The inert gas environment in the MPC precludes the incidence of corrosion during transportation. Additionally, the only dissimilar material groups in the MPC are (1) the neutron absorber material and stainless steel and (2) aluminum and stainless steel. Neutron absorber materials and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from neutron absorber materials and stainless steel materials, with geometries similar to the HI-STAR 100 MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This service experience provides a basis to conclude that corrosion will not likely occur in these materials. Furthermore, aluminum rapidly passivates in an aqueous environment, leading to a thin ceramic (Al_2O_3) barrier, which renders the material essentially inert and corrosion-free over long periods of application.

The HI-STAR 100 overpack combines low-alloy and nickel alloy steels, carbon steels, neutron and gamma shielding, thermal expansion foam, and bolting materials. All of these materials have a history of non-galvanic behavior within close proximity of each other. The internal and external carbon steel surfaces of the overpack and closure plates are sandblasted and coated to preclude surface oxidation. The coating does not chemically react with borated water. Therefore, chemical or galvanic reactions involving the overpack materials are unlikely and are not expected. Furthermore, the interfacing seating surfaces of the closure plate metallic seals are clad with stainless steel to assure long-term sealing performance and to eliminate the potential for localized corrosion.

2.3.10 Design Code Applicability (generic):

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing code for the construction of the HI-STAR 100 System. The ASME Code is applied to each component consistent with the function of the structure, system, and components (SSCs) of the HI-STAR 100 System that are labeled Important to Safety (ITS). Some components perform multiple functions and in those cases, the most restrictive code is applied.

The HI-STAR overpack top flange, closure plate, inner shell, and bottom plate are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB, to the maximum extent practical. The remainder of the HI-STAR overpack steel structure is designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF, to the maximum extent practical.

2.3.11 Acceptance Tests and Maintenance Program (generic):

Weld examinations shall be performed in accordance with written and approved procedures, by qualified personnel. All results, including relevant indications, shall be made a permanent part of the quality records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

ASME Code Section III and Regulatory Guides 7.11 and 7.12 require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Charpy V-notch testing shall be performed on each plate or forging for the HI-STAR 100 Package containment boundary (overpack inner shell, bottom plate, top flange, and closure plate) in accordance with ASME Code Section III, Subsection NB, Article NB-2300. Weld material used in fabricating the containment boundary shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NB, Articles NB-2300 and NB-2430.

Non-containment portions of the overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NF, Articles NF-2300, and NE-2430. The non-containment materials to be tested include the intermediate shells, overpack port cover plates, and applicable weld materials.

2.3.12 HI-STAR 100 System, Version for HB:

Holtec states that the HI-STAR 100 System has been expanded to include options specific for use at PG&E's HB plant for dry storage and future transportation of SNF.

HB fuel has a cooling time of more than 25 years and relatively low burnup. Heat load and nuclear source terms of this fuel are substantially lower than the design basis fuel. Peak cladding temperatures and dose rates are below the regulatory limits with a significant margin. All major dimensions and features, such as diameter, wall thickness, flange design, top and bottom thicknesses, are maintained identical to the standard (generic) design.

The HI-STAR HB overpack is a heavy-walled, steel cylindrical vessel identical to the standard (generic) HI-STAR 100 overpack, except that the outer height is approximately 128 inches and the inner height is approximately 115 inches. The HI-STAR HB overpack does not contain radial channels vertically welded to the outside surface of the outermost intermediate shell as

installed on the HI-STAR 100 overpack (generic). The HI-STAR HB overpack utilizes neutron shielding (Holtite-A) placed in the annulus region between the multi-layered shells and enclosure shell without connecting ribs. This feature is unique to the "HB" version. The annular shield, a thick layer of a low conductivity material, Holtite-A, retards the lateral transmission of fire heat during hypothetical accidents, which minimizes the heating of the HI-STAR HB package internals and the stored fuel during fires.

MPC-HB is similar to the generic MPC-68F, except it is approximately 114 inches high. The MPC-HB is designed to transport up to 80 HB BWR SNF assemblies. Damaged HB fuel and fuel debris must be transported in the Holtec custom designed HB DFC.

Holtec considers that almost all of the HB fuel assemblies not classified as damaged are intact, however, the inspection records of the HB fuel assemblies precludes classifying the assemblies as intact fuel since the interior rods of the assembly are in an unknown condition. These rods are classified as undamaged and can perform all fuel specific and system related functions, even with possible breaches or defects.

Applicable design codes, standards and criteria for the HI-STAR HB are identical to HI-STAR 100 except that the internal surfaces of the intermediate shells will not be coated with a silicone encapsulate due to its lower heat loads.

Differences between the HI STAR HB and HI-STAR 100 are limited to: shorter overall length, lower package weight, reduced strength of impact limiter crush materials, smaller diameter threads on lifting trunnions, and MPC-HB neutron absorber material as follows:

Holtec states that Metamic is a neutron absorber material developed by the Reynolds Aluminum Company for spent fuel reactivity control in dry and wet storage applications. Metamic is requested to be used in the HI-STAR HB. Metallurgically, Metamic is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy (precipitation hardening, with magnesium and silicon as its major alloying elements) containing Type 1 ASTM C-750 boron carbide. Metamic is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. Typically, the average B_4C particle size is between 10 and 15 microns. High performance and reliability of Metamic is derived from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields uniform homogeneity. For the Metamic sheets used in the MPCs, the extruded form is rolled down into the required thickness.

Metamic has been subjected to an extensive array of qualification tests sponsored by the Electric Power Research Institute (EPRI) which evaluated the functional performance of the material at elevated temperatures (900°F) and radiation levels. Test results, documented in an EPRI report, indicate that Metamic maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state.

Time required to dehydrate a Metamic equipped MPC is expected to be less when compared to an MPC (generic) containing Boral, due to the absence of interconnected porosities. Analyses performed by Holtec show that the streaming due to particle size is virtually non-existent in Metamic. Metamic is a solid material, therefore, no capillary path through which spent fuel pool water can penetrate Metamic panels can chemically react with aluminum in the interior of the material to generate hydrogen. Chemical reaction of the outer surfaces of the Metamic neutron absorber panels that may occur with water to produce hydrogen transpires rapidly and reduces

to an insignificant amount in a short period of time. Nevertheless, combustible gas monitoring for Metamic equipped MPCs and purging or exhausting the space under the MPC lid during welding and cutting operations is required until sufficient field experience is gained that confirms that little or no hydrogen is released by Metamic during these operations.

Holtec states that each manufactured lot of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet specified requirements. Testing and installation shall be performed in accordance with written and approved procedures and/or standards and shall become part of the QA record. The material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances. Procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps or voids from occurring in the material. Neutron shield integrity shall be verified via measurements either at first use or with a check source over the entire surface of the neutron shield, including the impact limiters.

2.3.13 Conclusion

For the purposes of the this review, the staff did revisit previously approved material properties used in the original HI-STAR 100 Cask System application and no new determination on the adequacy of those material properties was made unless it was used as the basis for the proposed revision requested in this application.

Furthermore, staff finds that the HI-STAR HB is composed of materials with a service proven history of use. The staff concludes that the materials and manufacture of Metamic as stated in the application for use in the HI-STAR HB are sufficient. The staff's conclusion regarding the manufacture, qualification, and use of Metamic, for the purposes of this review, is applicable to the HI-STAR HB. Definitions for Fuel Debris, Damaged Fuel Assemblies and Undamaged Fuel Assemblies, added to address the HB fuel assembly limited inspection records for packaging and transportation, are acceptable based on structural, containment, and criticality reviews.

The staff determined that, based on review of the application and supplements, that all material properties used by the applicant for the Model No. HI-STAR 100 for use at HB continue to meet the requirements of 10 CFR Part 71.

2.4 General Standards for All Packages

Section 2.I.4 of the supplement notes that the HI-STAR HB is a shorter and lighter version of the HI-STAR 100, and the design features presented in Section 2.4 of the application continue to apply to the HI-STAR HB. Therefore, the staff finds that the HI-STAR HB package meets the 10 CFR 71.43 requirements for the general standards for all packages.

2.5 Lifting and Tie-Down Standards

Section 2.I.5.1 of the supplement considers a bounding lifting weight of 161,200 lbs of the HI-STAR HB to recalculate the governing section moment and stresses for the trunnions with a slightly smaller diameter at the threaded portion of the trunnion than that for the HI-STAR 100. Considering the NUREG-0612 load multipliers of 6 and 10 for the yield and ultimate section load capacities, respectively, the application determines that all safety margins are greater than 1.0. This meets the 10 CFR 71.45(a) provision, which requires that lifting devices be sized to resist three times the design load without reaching material yield strength.

Section 2.I.5.2 notes that the tie-down devices and the reaction loads in Section 2.5 of the application bound those for the HI-STAR HB, which is shorter and lighter than the HI-STAR 100. On this basis, the staff agrees with the applicant's conclusion that no new analysis needs to be performed for the tie-down devices for satisfying the requirements of 10 CFR 71.45 (b)(1).

Section 2.I.5.3 determines that the ultimate bearing capacity at the trunnion-to-top forging interface is greater than the trunnion load limit. The ultimate moment capacity at and beyond the interface is also greater than the trunnion moment limit. This demonstrates that the trunnion shank reaches ultimate structural capacities prior to the top forging reaching its corresponding ultimate load capacities. Therefore, the staff agrees with the applicant's conclusion that failure of the external shank of the lifting trunnion will not cause loss of any other structural or shielding function of the overpack. This satisfies the excessive load requirements of 10 CFR 71.45(a) and 10 CFR 71.45(b)(3).

2.6 Normal Conditions of Transportation

Heat and Free-Drop. Section 2.I.6.1 of the supplement notes that the operating temperatures for the Humboldt Bay fuel, which are at or below comparable temperatures for the HI-STAR 100 analyses, give relatively higher at-temperature stress allowables for the HI-STAR HB. It includes a bounding stress evaluation of the cask enclosure shell. On this basis, the applicant argued and the staff agrees that, for the same cask free-drop deceleration limits and design pressures/temperatures, all other stress analyses of the HI-STAR HB would have resulted in additional margins compared to those for the HI-STAR 100. Section 2.I.6.7 notes that, as part of the Section 2.I.7 impact limiter drop analyses, the cask decelerations for the 1-ft free drops for the HI-STAR HB are shown to be less than the HI-STAR 100 design basis limits. Hence, the applicant's evaluations satisfy the requirements of 10 CFR 71(c)(1) and 10 CFR 71(c)(7) for the heat condition and free drop tests, respectively.

Cold, Reduced External Pressure, Increased Internal Pressure, and Vibration. Section 2.I.6.3 of the supplement refers to the Section 2.I.6.1, "Bounding Evaluation of the Enclosure Shell," which is also applicable to the reduced external pressure condition. As a result, Sections 2.I.6.2 through 2.I.6.5 note and the staff agrees that no new analyses or other calculations need to be performed for the normal conditions of transport cold, reduced external pressure, increased external pressure, and vibration, respectively, to satisfy the requirements of 10 CFR 71(c)(2), (3), (4), and (5).

Water Spray, Corner Drop, and Compression. The HI-STAR HB is identical to the HI-STAR 100 in all respects except for the length of the overpack. As such, Sections 2.I.6.6, 2.I.6.8, and 2.I.6.9 note and the staff agrees that the respective 10 CFR 71(c)(6), (8), and (9) conditions of water spray, corner drop, and compression are not applicable to the HI-STAR HB package.

2.7 Hypothetical Accident Conditions

30-Ft Free Drop. Section 2.7 of the application provides an evaluation of 10 CFR 71.73 hypothetical accident conditions for the HI-STAR 100, which is being modified with limited design feature changes for the HI-STAR HB. Since the applicable design criteria and safety analyses for the HI-STAR 100 remain to be bounding for the HI-STAR HB, the staff focuses its review primarily on the Holtec approach of using analysis alone to establish the design basis cask decelerations associated with the impact limiter modifications.

Drawing No. C1765, sheet 2 of 7, of the application lists reduced nominal crush strengths of aluminum honeycomb materials for the five section types, which range from about 50% to 60% of the corresponding strengths of the HI-STAR 100. The configuration, including overall geometry and attachment to the overpack of the HI-STAR HB impact limiter, is identical to that of the HI-STAR 100, with the sole difference being the impact limiter crush material strengths.

There is no scale model drop testing of the HI-STAR HB. Contrary to the common practice of qualifying impact limiters of spent fuel transportation packages by drop tests, the applicant adapted the HI-STAR 100 differential equation method for evaluating the HI-STAR HB free-drop accidents only by analysis. Section 2.I.7.1 of the supplement provides a summary description of the method as used for the HI-STAR 100. Essentially, the method entails a combined analysis and testing approach in three steps: (1) the 1/8-scale quasi-static tests of the impact limiter to establish its load-deflection characteristics; (2) the ¼-scale cask drop tests to demonstrate that the experimentally obtained cask rigid body decelerations were below the design bases; and (3) the development of analytical models capable of correlating the observed and calculated results.

The HI-STAR 100 analytical models involve either a single second-order differential equation for end drops or a set of three differential equations for side and slap-down drops. There exist, in each equation, one or two resistive force terms each formulated as product of the impact limiter static deformation and dynamic multiplier, also called dynamic correlation function. To implement the method for the HI-STAR HB, given that all other relevant physical attributes of the cask system can be incorporated into the differential equations directly, the dynamic correlation function remains the only parameter that is updated, at each time integration step, as a linear function of the concomitant crush velocity of the impact limiter. In a September 30, 2008, letter the applicant addressed the staff RAI on appropriate selection of the dynamic multipliers for the HI-STAR HB without relying on scale model drop tests. As discussed in Section 2.I.7.1, Revision 13c, of the supplement, the applicant further clarified it by noting that, based on manufacturer's catalog, no information suggesting that the dynamic multipliers in the differential equation method are a function of crush material strength. Therefore, the staff has reasonable assurance to agree with the applicant that the dynamic multipliers originally determined for the HI-STAR 100 remain valid for the HI-STAR HB analytical method, as discussed below.

The numerical simulation analysis of the HI-STAR HB impact limiters uses the same dynamic multipliers as those for the HI-STAR 100, for which, analysis results correlate adequately with the drop test results. Compared to the HI-STAR 100 impact limiters, the staff notes that the HI-STAR HB aluminum honeycomb sections are identically configured, constrained, and supported by an essentially rigid backbone structure. This ensures development of similar load paths for the resistive forces in both the HI-STAR 100 and HI-STAR HB impact limiters for same dynamic

multipliers. Thus, the staff has reasonable assurance that the HI-STAR 100 differential equation method is an acceptable predictive tool for determining HI-STAR HB cask decelerations and impact limiter crush depths. The staff also agrees with the applicant's assessment that the HI-STAR HB impact limiters will continue to remain attached to the cask if the calculated maximum cask decelerations are below 60 g, given that the HI-STAR 100 impact limiters remained attached to the overpack during the scale model drop tests.

Table 2.I.7.1 of the supplement lists calculated maximum cask decelerations and impact limiter crush depths for the 30-ft free drop accidents. With nominal strengths of the crush material, the maximum decelerations are 56.5 g, 45.6 g, 34.8 g, 33.8 g, and 45.9 g for the top-end, bottom-end, side, C.G.-Over-Corner, and slapdown drops, respectively. Table 2.I.7.3 lists the crush strength sensitivity analysis results for ten drop cases. For a crush strength increase of 15% over the nominal, the maximum decelerations are 59 g, 38.5 g, and 49.2 g for the respective top-end, side, and slapdown drops, which are all below the design basis limit of 60 g. For the 30-ft side drop, for a decrease of crush strength by 15%, the calculated crush depth of 15.2 inch indicates that the impact limiters experience some material lockup. This results in a cask side-drop deceleration incrrease from the nominal 34.8 g rise to 43 g, as it would be. However, the impact limiters still provide acceptable protection for the cask subject to a maximum side-drop deceleration far less than the 60 g design basis limits.

On the basis of the review above, the staff concludes that the HI-STAR HB design basis decelerations are bounded by the 60 g used also for the HI-STAR 100 package evaluation. As such, all relevant Section 2.7.1 evaluations for the HI-STAR 100 remain to be applicable to the HI-STAR HB in demonstrating its structural capabilities for meeting the 10 CFR 71.73 (c)(1) free-drop requirements.

Puncture. Section 2.I.7.2 of the supplement notes and the staff agree that the structure at the puncture locations is unchanged from the HI-STAR 100. Hence, no new or modified calculations need be performed for the HI-STAR HB for meeting the puncture test requirements of 10 CFR 71.73(c)(3).

Thermal. Section 2.I.7.3 of the supplement notes the thermal evaluation of fire accident. The staff agrees with the applicant's assessment that no new or modified structural calculations need be performed to qualify the HI-STAR HB for the 10 CFR 71.73(c)(4) fire test requirements.

Immersion - Fissile Material and Immersion – All Packages. The staff evaluated the structural feature differences between the HI-STAR HB and HI-STAR 100 packages and concurs that no new or modified calculations need be performed to qualify the HI-STAR HB for the subject immersion requirements of 10 CFR 71.73(c)(5) and (6).

2.8 Fuel Rods

The Humboldt Bay fuel is shorter than the design basis fuel carried by the HI-STAR 100 and will, therefore, exhibit larger structural margins against the side- and end-drop accidents. Thus, the staff concurs with the applicant's conclusion that the performance of the HI-STAR HB fuel rods is bounded by that of the HI-STAR 100, and is, therefore, acceptable. Table 1.0.1 of the CoC defines undamaged fuel assemblies as those where all exterior rods in the assembly are visually inspected and shown to be intact even if the cladding of interior rods are of unknown condition. Section 2.I.9 of the supplement notes that the exterior fuel rods serve to confine the interior fuel rods, thereby preventing interior fuel rods from dislocating and falling to the bottom

of the fuel basket. This potential, but limited, reconfiguration of interior fuel rods of undamaged fuel assembly will result in negligible change of fuel mass and center of gravity height, which has insignificant effects on the structural response of the HI-STAR HB. Furthermore, as reviewed in Section 6.4 of this safety evaluation, reconfigured interior fuel rods are acceptable based on the conditions analyzed for criticality control. This justifies loading the “undamaged fuel assemblies” into the MPC-HB directly without placing them in the HI-STAR HB damaged fuel containers.

2.9 Review Findings

The staff reviewed the statements and representations in the application by considering the regulations, appropriate Regulatory Guides, applicable codes and standards, and acceptable engineering practices. The staff concludes that the structural design has been adequately described and evaluated for meeting the requirements of 10 CFR Part 71.

3.0 THERMAL

The staff reviewed the application to revise the Model No. HI-STAR 100 package thermal design and evaluation to assess whether the package temperatures will remain within their allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 (Thermal Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

3.1 Thermal Review Description

The purpose of this Revision is to facilitate the transport of HB reactor spent fuel; however, some of the changes are generic in nature and apply to the entire family of contents of the HI-STAR 100. Besides the fuel type change, other changes impacting the thermal evaluation are: the addition of Metamic as a neutron absorber material, the addition of a higher emissivity value for stainless steel plates, and the request to remove the thermal acceptance test along with the thermal periodic test.

3.2 Thermal Evaluation

The fuel from HB is low burnup fuel that has been sitting in the spent fuel pool and/or dry cask storage for many years, and as a consequence, the decay heat limit for this physically smaller package is only 2 kW for its associated 80 fuel assemblies. The current approved HI-STAR 100 thermal design limit for BWR fuel is 18.5 kW for 68 fuel assemblies.

Additionally, the applicant has chosen to add Metamic as an additional neutron absorber in the SAR. Since Metamic is made of the same materials as Boral, it exhibits the same thermal conductivity, even though the manufacturing process is different. As a consequence, the same values of thermal conductivity for Boral are used.

A higher value emissivity (0.587) for stainless steel plates was provided, but it was stated, for conservatism, that a lower value of 0.36 continues to be used in the thermal calculations. The staff was concerned that this higher emissivity value may not be conservative for the HAC fire. However, after reviewing the Holtec response, the staff agrees that the starting point for the temperature distribution within the cask would be lower, and as a consequence, the impact of

the 30 minute HAC fire would be alleviated by this and the thermal inertia of the massive transportation cask.

From the thermal analysis for HB, the NCT maximum temperature for the fuel cladding is 419°F and the overpack top plate is 129°F. These values are considerably less than the design basis HI-STAR 100 BWR fuel of 713°F for the fuel cladding and 162°F for the overpack closure plate. Also, since the decay heat load of the fuel for HB is significantly lower than the design basis heat load of the HI-STAR 100 and the HB overpack utilizes a heat shield, the HB thermal loading is bounded by the current design basis of the HI-STAR 100 for NCT and HAC.

At the request of the staff, a description of the heat shield (referred to in Table 3.I.5) was added to SAR supplement Section 3.I.1. The heat shield is another term for the neutron shield where the previous radial support channels for the outer enclosure shell of the neutron shield were replaced with gussets at the top and bottom of the shield - so that only neutron shield material would be at the radial centerline of the cask to further inhibit heat input to the fuel during the HAC fire.

Additionally, the applicant initially requested that the thermal acceptance test and the thermal periodic test be removed for all package configurations. Since the thermal acceptance test was limited to the first fabricated HI-STAR overpack, and that test was completed satisfactorily, the staff has no objection in removing this commitment. For the periodic thermal test, the staff is confident in the time dependent thermal performance of the HI-STAR 100 packaging materials except for the shielding material Holtite-A, and requested that its time dependent thermal characteristics be evaluated since Holtite-A is a polymer and, as such, is typically susceptible to heat and radiation degradation. In response to the staff's request HOLTEC submitted two reports entitled:

1) "Holtite-A: Results of Pre- and Post- Irradiation Tests and Measurements," HI-2002420, Rev. 1, dated 4/8/03, and

2) "Holtite-A Development History and Thermal Performance Data," HI-2002396, Rev. 3, dated 4/10/03.

The former report documents irradiation aging and performed physical observations; weight and density determinations; dimensional measurements; and neutron attenuation testing or chemical analyses on the Holtite-A samples. The latter report documents that the thermally aged samples of Holtite-A were free of large voids and gaps; visual examination confirmed material was stable (no warping, swelling, or cracking); and that weight loss results were less than 4%. Also, these tests were performed independently and did not evaluate the synergistic effects of dose and temperature. The staff did not find a direct correlation between these reports measurements/conclusions and providing assurance that the thermal conductivity of Holtite-A does not change with time. As a result, the staff requested Holtec to provide additional data of Holtite-A to determine the time dependent effect of radiation and heat on Holtite-A, or continue performing the periodic thermal test. Holtec chose to continue performing the periodic thermal test. This commitment is included in the CoC since the CoC requires that all procedures for acceptance testing and maintenance be developed from the provisions of SAR Chapter 8.

From a review of the proposed application of HB, the staff concludes that adequate justification has been presented to conclude that the material temperature limits of the HI-STAR HB transportation package have been satisfied considering the large design margins stemming

from the relatively lower decay heat load and the approximate 300°F difference between the HAC cladding temperature and its limit of 1058°F.

3.3 Conclusion

Based on the review of the application, the staff found reasonable assurance that the applicant has demonstrated that the HI-STAR HB package meets the thermal requirements for NCT and HAC as required by 10 CFR Part 71.

4.0 CONTAINMENT

The staff reviewed the application to revise the Model No. HI-STAR 100 package to verify that the package containment design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

4.1 Containment System Design

The purpose of this Revision request is to facilitate the transport of HB reactor spent fuel. Besides the fuel type change, other changes that could impact the containment evaluation are: the revised A_2 values from 10 CFR Part 71, the removal of the MPC as a secondary containment (including the commitment to leak test it), updated procedures to change the closure bolt torque to 2000 ft-lbs from 2895 ft-lbs, and the changed definition of “damaged fuel assembly.”

4.2 Containment Evaluation

The HB spent fuel is bounded by the design basis fuel for source term, and consequently, its reference leak rates are more than that of the design basis fuel and result in no impact on the previously approved HI-STAR 100 package. Furthermore, the reference leak rate calculations, presented in Table 4.1.1, “Summary of Containment Boundary Design Specifications,” of the SAR continue to justify a leakage rate acceptance criterion of 4.3 E-6 atm cm^3/sec, He. This leakage rate acceptance criterion has been verified to be valid for the HB fuel, as well as for the changes in the A_2 values.

Since the double containment requirement for plutonium shipments has been removed from the regulations (ref. 10 CFR 71.63 dated 1/26/2004), the requirement to have the MPC serve as a second containment boundary and be leak tested is no longer required and that information has been removed from the SAR.

The lessening of the closure bolt torque values has been reviewed in the structural section and staff agrees that these new torque values still provide an adequate closure force to ensure integrity of the containment boundary.

The change in the definition of a damaged fuel assembly has no effect upon the containment evaluation because it has no effect on the source term.

Also, the latest version of ANSI N14.5 (i.e., 1997) was referenced by the applicant at the request of staff.

4.3 Conclusion

Based on the statements and representations in the application, staff agrees that the applicant has shown that the use of the Model No. HI-STAR 100 for use at HB continues to meet the containment requirements of 10 CFR Part 71.

5.0 SHIELDING

The staff reviewed the application to revise the Model No. HI-STAR 100 package to verify that the shielding design has been described and evaluated under NCT and HAC, as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 5 (Shielding Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

5.1 Description of Shielding Design

5.1.1 Packaging Design Features

Staff reviewed the changes to the design features proposed in the amendment request. The following changes were proposed that affect the shielding design:

Authorization of Metamic as a neutron absorber material was added to the application for use in the HI-STAR HB. The shielding analysis takes credit for the presence of the absorber plates. The currently approved design uses Boral as the neutron absorber.

Addition of the HB overpack, MPC, and basket (and HB-specific DFC). The design for the HB system differs from the standard HI-STAR 100 system in a few parameters. A significant difference is the reduced axial height of the overpack and the MPC. The shielding materials are unchanged. An additional difference is the lack of radial channels welded to the outermost intermediate steel shell of the HB overpack; thus, the neutron shield is not penetrated by steel channels that will result in neutron streaming paths through the neutron shield. Additionally, the HB overpack neutron shield thickness is a minimum of 4 inches, while the minimum thickness is 4.3 inches for the standard overpack. The MPC-HB basket holds 80 HB fuel assemblies (versus the 68 assemblies of the standard BWR MPC baskets).

No other proposed changes affect the shielding design. The staff reviewed the licensing drawings and descriptions of the HI-STAR 100 system as modified in the proposed amendment and finds there is sufficient detail to perform a shielding evaluation. The staff reviewed the proposed changes and finds them to be described in sufficient detail to perform an evaluation of the shielding design.

5.1.2 Codes and Standards

The applicant continues to use the flux-to-dose conversion factors from ANSI 6.1.1-1977. The staff finds use of these conversion factors to be acceptable.

5.1.3 Summary Table of Maximum Radiation Levels

The summary dose rate tables for the MPC-24, MPC-32, and MPC-68 are provided in Section 5.5 of the amended SAR. Some of the dose rates for all three MPCs were modified due to a modification of the impact limiter model used in the analyses. Dose rates for the MPC-68 were modified to account for the proposed additional contents at dose locations where the proposed additional contents result in higher dose rates. The tables show maximum dose rates for the side, top, and base of the HI-STAR 100 to be below the 10 CFR Part 71 regulatory dose rate limits. Staff reviewed the dose rates presented in Section 5.5 of the SAR and finds they are consistent with those reported in the rest of the analysis and are the maximum dose rates.

5.2 Source Specification

The staff reviewed the specifications for the contents proposed in the amendment request. No changes were proposed to the allowable PWR contents. Proposed changes to the BWR contents include a decreased minimum allowable enrichment for intact and damaged 6x6A, 6x6C, 7x7A, and 8x8A assembly arrays/classes and fuel debris from these assembly classes (the proposed minimum enrichment is 1.45 wt.%) and two new maximum burnup, minimum cooling time, and minimum enrichment combinations for the remaining ZR-clad BWR assembly arrays/classes. One new combination is a maximum burnup of 10,000 MWd/MTU and a minimum of 5 years cooling for assemblies with a minimum enrichment of 0.7 wt.%, and the other combination is a maximum burnup of 20,000 MWd/MTU and a minimum of 7 years cooling for assemblies with a minimum enrichment of 1.35 wt.%.

The amendment also proposes to remove the HB assembly arrays/classes from the allowable contents of the MPC-68 and MPC-68F and allow transportation of these assemblies in the MPC-HB. The amendment proposes to allow transportation of the HB assemblies, 6x6D and 7x7C assembly arrays/classes, as intact, undamaged and damaged assemblies and fuel debris. Damaged fuel assemblies and fuel debris must be loaded in HB Damaged Fuel Canisters (DFC). The applicant proposes to load either 28 damaged assemblies in the peripheral basket locations or 40 damaged assemblies in a checkerboard pattern. These two configurations are illustrated in SAR Figures 6.I.3 and 6.I.4.

Based upon HB fuel records, there are no more than 337 linear inches of fuel fragments remaining at the HB reactor site. This fuel debris may include loose zirconium-clad pellets, stainless steel-clad pellets, unclad pellets and rod segments. Assuming all the fragments are clad with stainless steel and considering the cladding dimensions, the amount of stainless steel cladding that could be loaded is 1.25 kilograms. Therefore, the applicant proposed a limit for fuel debris of 1.5 kilograms of stainless steel cladding per cask. This limit is included in Section VII of Table A.1 in Appendix A to the CoC. The shielding analysis also includes the contribution of the steel cladding to the source terms used to demonstrate compliance with the regulatory dose rate limits. The design-basis HB assembly is the 6x6D class assembly with the characteristics provided in Table 5.I.1 of the SAR.

A definition for undamaged fuel was proposed to be added to the CoC. This definition is strictly limited in application to HB fuel assembly classes/arrays. Undamaged assemblies are those HB assemblies for which the condition of the assembly could not be completely verified to meet the CoC definition of intact fuel; however, visual examinations of these assemblies supported the determination that the outer rods of the assemblies are intact. These examinations also confirmed that the assembly interior rods are in place but could not confirm the condition of the

cladding of the interior rods. Thus, any changes to the assembly configuration due to accident conditions would only take place in the assembly interior and be confined by the outer rods to the assembly envelope (see Section 2.3.12 of this SER). The shielding analysis considers assemblies of this condition (see Section 5.4 of this SER).

5.2.1 Gamma Source

The applicant used the SAS2H and ORIGEN-S modules of the SCALE suite of codes to calculate the source terms, both neutron and gamma, for the proposed contents. SAR Table 5.2.5 lists the gamma source strength for design-basis ZR-clad BWR assemblies for the different maximum burnup, minimum cooling time, and minimum enrichment combinations analyzed. Table 5.2.6 lists the gamma source strength for the Dresden 1 assembly arrays/classes. Table 5.I.3 lists the gamma source strength for the HB assembly arrays/classes. Table 5.2.10 lists the Cobalt-60 source strength per assembly from assembly hardware, with Table 5.I.4 as the equivalent table for the HB assembly hardware. The staff reviewed the gamma source strengths provided in these tables and also performed confirmatory calculations of the gamma sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.2.2 Neutron Source

The neutron source strengths for the design-basis ZR-clad BWR assemblies, the Dresden 1 assemblies, and the HB assemblies are provided in SAR Tables 5.2.13, 5.2.14, and 5.I.2, respectively. The staff reviewed these neutron source strengths and performed confirmatory calculations of the neutron sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.3 Model Specification

The applicant uses the same shielding models as for analyses performed in the previously approved amendment, with the exception of a correction to the thickness of the impact limiter ribs. Sections 2 and 3 of this SER describe staff's evaluations of the structural and thermal performance of the HI-STAR 100 system as proposed in the amendment. None of the proposed changes were found to exceed the bounding conditions affecting shielding as evaluated in the previously approved amendment. Additionally, the shielding configuration of the HI-STAR HB is generally bounded by the shielding configuration of the design-basis HI-STAR 100. The neutron shield is 0.3 inches thinner for the HB overpack than it is for the standard HI-STAR 100 overpack, a condition which increases the neutron dose rates by about 30%. The applicant addressed this difference in the neutron shield in its evaluation of the HB system. Also, while the amendment proposes to allow use of Metamic as a neutron absorber, in addition to Boral, the models continue to use Boral. The modeled Boron-10 areal densities are the same and the plate thicknesses are essentially the same for the two absorbers; thus, the amount of aluminum and B_4C are essentially the same, resulting in no distinction between the two materials from a shielding perspective. Based on its review of these proposed changes, the staff finds that the shielding models, with the correction to the impact limiters, remain appropriate.

5.4 Evaluation

The applicant performed the shielding analysis with MCNP-4A, the same code the applicant used for the shielding analyses in the previously approved amendment. The applicant calculated the dose rates for design-basis BWR fuel with the proposed maximum burnup and minimum cooling time and enrichment combinations. These dose rates are presented in SAR Tables 5.4.9, 5.4.11, and 5.4.13 for the surface dose rates and two-meter dose rates for normal conditions and one-meter dose rates for accident conditions. The correction of the impact limiter thickness in the model affected the normal conditions dose rates at the cask base and top (surface and two-meter distance); therefore, updated dose rates for normal conditions are provided for all the MPCs in SAR Tables 5.4.8-11, 19, 20, 22-24, 26, 27, 29, 30, 32, and 33. These changes also necessitated updates to the summary tables, SAR Tables 5.5.1-3, used by the applicant to show compliance with the 10 CFR Part 71 regulatory dose limits. The maximum dose rates, however, continue to remain below the regulatory limits.

Dose rates were not calculated for the Dresden 1 assemblies with the proposed minimum enrichment. Instead, the assembly neutron and gamma source strengths were compared on a source strength per inch basis with the source strengths of the design-basis BWR fuel assemblies (39,500 MWd/MTU and 14 years cooling). The comparison was made for damaged Dresden 1 assemblies that had reconfigured under accident conditions and was initially done using only the total neutron and gamma source strengths. This comparison showed that the design-basis BWR fuel source bounds the Dresden 1 fuel source. The applicant then also compared the source strengths on an energy-group basis. This second comparison showed that the design-basis neutron and gamma source strengths bound those of the damaged Dresden 1 assemblies over all energy groups except the 1.0 to 1.5 MeV gamma source. However, dose rate calculations showed that the dose rates (on the overpack side) from the damaged Dresden 1 assemblies are significantly lower than those from the design-basis intact BWR assemblies (by about 20%). The applicant concludes, therefore, that the dose rates from the Dresden 1 fuel assemblies will always be bounded by the dose rates from the design basis BWR fuel assemblies. The staff reviewed this information and performed a confirmatory analysis and finds that the dose rates from the design-basis intact BWR assemblies bound the dose rates from the Dresden 1 fuel assembly arrays/classes.

The applicant used a similar source term comparison (on an energy-group basis) between the design-basis intact BWR assemblies and the HB assemblies. In this case, however, since the MPC-HB and MPC-68 hold different numbers of assemblies, the source per inch was multiplied by the number of assemblies in the respective MPC. For accident conditions, the MPC-HB source per inch accounted for reconfiguration of the damaged HB assemblies. The comparison was made with design-basis intact BWR assemblies having a burnup of 24,500 MWd/MTU and 8 years cooling. The applicant's comparison indicated that the MPC-HB source strength over all energy groups (both neutron and gamma) is bounded by the source strength of the design-basis intact BWR assemblies, meaning the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing a MPC-68 loaded with design-basis intact fuel.

This initial comparison looked solely at comparisons for the entire cask loading. However, staff questioned whether this comparison was sufficient given the different loading schemes for damaged fuel in the HB overpack and the relative importance of different MPC basket zones on the dose rates (interior versus exterior basket locations). For example, for side dose rates, fuel assemblies in the outermost rings of basket locations tend to dominate the gamma dose rates. This configuration will provide a better indication of the impact on dose rates of the 28 damaged

assembly pattern in the HB overpack. In response to staff's questions, the applicant modified its evaluation to account for the relative importance of the different MPC basket zones to dose rates. The modified evaluation continued to indicate that the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing the MPC-68. The modified evaluation also accounts for the difference in neutron shield thickness, increasing the neutron source by 30%, the assembly hardware, and the stainless steel cladding of fuel debris in the DFCs. For accident conditions, the applicant performed further comparisons with the assumption that all assemblies in the HI-STAR HB are damaged. This latter comparison was performed as a bounding approach for addressing the condition of assemblies where the condition of the cladding could not be verified for fuel rods interior to the assembly lattice. These assemblies are classified as undamaged, which classification applies only to the HB fuel arrays/classes. This comparison also indicates the dose rates from the HI-STAR HB are bounded by the dose rates from the HI-STAR 100 containing the MPC-68.

The staff reviewed this information and also performed its own comparisons of the source terms. The comparisons included accident conditions for both allowable loading configurations of damaged fuel and the presence of undamaged fuel as well as a comparison of the source terms for assemblies loaded in equivalent outer basket cell regions, under both normal and accident conditions. Based on its review of the applicant's analysis and its own comparisons, the staff finds that the dose rates from the HI-STAR HB will be bounded by the dose rates from a HI-STAR 100 loaded with design-basis intact BWR fuel since the HI-STAR HB source strength, on a per inch basis, is bounded by that of the HI-STAR 100 containing design-basis intact BWR fuel in all the evaluated comparison scenarios.

5.5 Conclusion

Based on its review of the information and representations provided by the applicant in the amendment request and the SAR and independent analyses, the staff has reasonable assurance that the changes to the package design and contents satisfy the shielding requirements and dose limits in 10 CFR Part 71.

6.0 CRITICALITY

The purpose of this review is to verify that the proposed amendment meets the criticality safety requirements under normal conditions of transport and hypothetical accident conditions. The objectives include a review of the criticality design features and fuel specifications; review of the configuration and material properties for the HI-STAR HB Overpack; and a review of the methodology and results found in the criticality evaluation.

The staff reviewed the criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STAR HB with the MPC-HB basket configuration meets the following regulatory requirements: 10 CFR 71.31, 71.33, 71.35, and 71.59. The staff's review also involved a determination on whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

6.1 Description of Criticality Design

6.1.1 Packaging Design Features

The MPC-HB is a fuel basket with an increased capacity to accommodate up to 80 assemblies while maintaining the same MPC outer diameter. The criticality safety design continues to rely on the geometry of the fuel basket, fuel enrichment limits, and poison plates for criticality control, but has added Metamic as an alternative to the Boral poison plates.

Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71.71 and 71.73. The staff reviewed the description of the package design and found that the important features were appropriately identified and adequately described. The engineering drawings and other information are sufficient to permit an independent evaluation.

6.1.2 Summary Table of Criticality Evaluations

A summary of the criticality evaluation results for the HB fuel is reported in Table 6.I.1 of the SAR. The table includes results for a single package and arrays of undamaged and damaged packages including damaged fuel and fuel debris being transported in the MPC-HB.

The results show that the package design meets the requirements of 10 CFR Part 71 for criticality safety. All values of k_{eff}, after being adjusted for uncertainty and biases, fall below the acceptance limit of 0.95 given in the Standard Review Plan (SRP).

6.1.3 Criticality Safety Index

The applicant's analyses considered an infinite array of packages under both normal conditions of transport and hypothetical accident conditions and showed that they were below the acceptable limit. Therefore, the Criticality Safety Index is 0.0 for the package.

6.2 Spent Nuclear Fuel Contents

The applicant requested the addition of the 6x6D and 7x7C fuel assembly types with or without channels in the MPC-HB for transport in a modified (shorter) HI-STAR HB. The maximum assembly average enrichment is 2.6% for intact fuel, undamaged fuel, damaged fuel, and fuel debris. Undamaged fuel assemblies are assemblies where all of the exterior rods are shown to be intact; while the interior rods of the assembly are in place the condition of the cladding on these rods cannot be verified. Damaged fuel and fuel debris must be canned in a Holtec designed Damaged Fuel Container (DFC). Damaged fuel and fuel debris may be loaded into the 28 peripheral cell locations in the basket or into 40 cell locations in a checkerboard pattern in the basket.

The staff has reviewed the description of the spent fuel contents and concludes that it provides an adequate basis for the criticality evaluation.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant used the same general assumptions and modeling methods as previously reviewed for intact fuel, damaged fuel, and fuel debris. Notable modeling features used for this application are: (1) modeling the MPC-HB with 80 fuel assemblies and a shorter length than the current canister and transportation overpack, (2) analyzing variations of assembly positioning within the fuel basket cells where the fuel assemblies are shifted toward the basket center as well as centered in the fuel cells, (3) analyzing the damaged fuel and fuel debris as bare rods of fuel, (4) modeling undamaged fuel assemblies in place of intact assemblies to analyze the effects on reactivity for both approved basket configurations (peripheral and checkerboard), (5) modeling cases with Metamic and Boral plates separately to assess their analytic similarity, and (6) simulating fabrication damage to the poison plates as a hole up to 1-inch diameter.

6.3.2 Material Properties

The material properties remained the same as previously reviewed except for the addition of Metamic where 90% credit was taken for the minimum boron content in this type of absorber plate.

6.3.3 Computer Codes and Cross Section Libraries

The applicant used the three-dimensional continuous energy code Monte Carlo N-Particle (MCNP4a) for the criticality analysis. The staff agrees that the codes and cross-section sets used in the analysis are appropriate for this application and fuel system.

6.3.4 Demonstration of Maximum reactivity

In the safety analysis, the applicant used fuel and basket dimensions within the tolerance limits which maximize k_{eff} as were found in previous analyses. These optimum conditions are: maximum active fuel length, maximum fuel pellet diameter, minimum cladding outside diameter, maximum cladding inside diameter, minimum guide type thickness, maximum channel thickness, minimum cell pitch, minimum cell inner dimensions, and nominal cell wall thickness.

The applicant performed a sensitivity analysis for the HB fuel and found that a package was more reactive when all fuel assemblies are shifted toward the center of the basket versus being centered in each cell but found no statistically significant difference between the cases of each poison plate being damaged with a 1 inch diameter hole in its middle versus an undamaged plate. In the final safety analyses, the applicant assumed all fuel assemblies were shifted toward the basket center and that all poison plates were damaged with a 1 inch diameter hole in the center.

Staff found the methods used to identify the parameters values which maximize k_{eff} to be appropriate and found the set of parameters used in the analysis to be acceptable.

6.3.5 Analysis Approach

The applicant performed comparative calculations for the two different HB fuel types and determined that the 6x6D was more reactive than the 7x7C assembly type. Subsequently, the

6x6D was used as the intact fuel type when analyzing a transportation package loaded with bounding combinations of intact, undamaged, and damaged fuel assemblies.

In addition to analyses to support the inclusion of HB fuel in the HI-STAR 100, the applicant also analyzed the acceptability of the alternative poison plate material. Analyses were performed with Metamic poison plates and then compared to a cask fabricated with Boral poison plates. In the criticality analysis, 75% of the minimum B-10 content is credited for Boral while 90% of the B-10 content is credited for the Metamic. The B-10 content in Metamic is chosen to be lower, and is chosen as the B-10 content so that both materials are the same in the analysis. The difference in k_{eff} was not statistically significant.

6.4 Single Package Evaluation

The applicant performed calculations for the MPC-HB which show the design is more reactive when internally flooded with full density water versus flooding with lower density water. Thus, a fully flooded package was used in the subsequent single package analyses. The applicant then performed a series of calculations for single HI-STAR HB packages containing the MPC-HB with 6x6D and 7X7C fuel for each case of an unreflected package, full reflection by water, and full reflection of the containment only. This was performed for two different cases: one with either all intact or undamaged assemblies; and one with damaged assemblies loaded with intact or undamaged assemblies. The bounding case was for the model with the damaged fuel (7x7 rod array) and the 6x6D undamaged fuel (in intact assembly positions) in an eccentric assembly positioning with assumed poison plate damage. The subsequent single package calculations used the results for a single unreflected package with full internal moderation but without the assumed poison damage. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

The applicant performed calculations for an infinite array of undamaged packages containing the MPC-HB with intact and undamaged assemblies. All of the applicant's results were below the acceptance level of 0.95 for k_{eff} In these calculations, the packages were internally dry and had no moderator between packages.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The applicant performed calculations for an infinite array of damaged packages containing the MPC-HB with the limiting case of full internal moderation as found for the single package. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.7 Benchmark Evaluations

The applicant's benchmarking procedures and methods have not changed and staff's evaluation is provided in a previous SER.

6.8 Evaluation Findings

The staff has reviewed the criticality description and evaluation of the package and concludes that it addressed the criticality safety requirements of 10 CFR Part 71.

The applicant analyzed the reactivity involving approved loading scenarios. The evaluation analyzed the effects of intact fuel, undamaged fuel, and damaged fuel loaded into peripheral and checkerboard loading configurations within the MPC-HB. A number of conservatives were used when determining the bounding condition. Reactivity results were evaluated for both approved loading configurations (peripheral and checkerboard) having bounding combinations of intact, undamaged, and/or damaged fuel.

Section 6.I.4.1 of the application states that assemblies with defects are considered damaged, and need to be placed into damaged fuel containers (DFCs). Calculations were performed in the application to ensure that assemblies with intact rods in the outermost rods but which may also have defects in the inner rods could be loaded without being placed into DFCs. As part of the evaluation, the rod pitch was varied inside the assembly to analyze the effects of reflection by water. The application shows that loading the MPC-HB with undamaged assemblies (in place of intact assemblies) along with damaged assemblies still yields results which are below the acceptance level of 0.95 for k_{eff}. Staff agree that the undamaged fuel assemblies with parameters described in the SAR report are acceptable to be loaded in the MPC-HB without being placed in a DFC as long as the overall integrity of the assembly (outer intact rod positioning, guide plates, etc.) remains structurally intact (See Section 2.3.12 of this SER).

Staff identified a minor error in the criticality evaluation regarding a supplemental report provided by the applicant (Holtec Report No: HI-2033010). Table C-1 stated that the bounding condition used for the 6x6 array with full water reflection came from C-55, which corresponds to a case where the model is centered. This is not consistent with the criticality evaluation found in Chapter 6 of the SAR, which states that the bounding case included the assembly being modeled in an eccentric manner. A telephone conference was conducted in which the applicant verified the assumption of NRC staff that it was merely a typographical error, and the applicant plans to correct this in the next revision of the report.

It should be noted that staff finds the 6x6D and 7x7C fuel assemblies, having parameters listed in the SAR, acceptable for loading into the MPC-HB. This is based on the conservatisms used in the analysis as well as the bounding reactivity in relation to the acceptance level of 0.95.

6.9 Conclusion

Based on the review of the presentations and information supplied by the applicant, staff finds reasonable assurance that the proposed amendment meets the criticality safety requirements of 10 CFR Part 71.

7.0 Package Operations

As part of the amendment, the applicant proposed a significant revision of the package operations. Instead of including the package operations explicitly in the CoC, the applicant proposed to place this information in Chapter 7 of the SAR and incorporate this chapter into the CoC by reference. Several items in Chapter 7 were modified as part of the proposed change in addition to inclusion of operations for the HI-STAR HB. Initially, the chapter included operations descriptions for dry loading as well as wet loading operations. However, due to concerns such as how dry loading operations would be performed and/or limited to prevent fuel oxidation, the applicant removed these operations.

The operations chapter also included a statement that indicated that HI-STAR 100 users could modify the sequence of, add or remove operations as necessary. This statement introduces uncertainty into what constitutes the essential package operations, the delineation of which is the purpose of Chapter 7. In response to staff's questions, the applicant removed the statement. Also, for operations for which the sequence does not impact the package preparation, the applicant modified Chapter 7 to explicitly indicate the operations which may be performed in any sequence in relation to each other. Staff also noted several important operations and the completion/acceptance criteria for other operations were removed from the descriptions in Chapter 7 as part of the proposed amendment. The applicant restored these descriptions in response to staff's questions.

Based upon its review of the descriptions in the application, the staff finds that the package operations meet the requirements of 10 CFR Part 71 and that the operations are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

8.0 Acceptance Tests and Maintenance Programs

As part of the amendment, the applicant proposed a significant revision of the acceptance test and maintenance program. Instead of including this information explicitly in the CoC, the applicant proposed to place this information in Chapter 8 of the SAR and incorporate this chapter into the CoC by reference. A number of items in Chapter 8 were modified as part of the proposed change. The acceptance tests and maintenance program is fully applicable to the HI-STAR HB without modification.

In its review of the Chapter 8 descriptions, the staff found that certain important acceptance criteria and tests related to shielding materials and shielding effectiveness had been removed or modified. In particular, the requirements for gamma shielding materials were not included in the proposed Chapter 8. However, staff considers that acceptance tests and criteria for all shielding materials should be included in the Chapter 8 program descriptions. The applicant modified the neutron shield acceptance test and periodic neutron shield integrity verification test to consist only of the radiation surveys performed prior to transport (and upon package receipt, for the periodic test) as described in Chapter 7 of the SAR. Staff considers that these surveys do not meet the purposes of the acceptance test and periodic integrity verification test of the neutron shield. The pre- and post-transport surveys only serve to ensure the 10 CFR Part 71 dose rate limits are not exceeded for a particular shipment. The acceptance and periodic verification tests ensure that the as fabricated neutron shield performs as designed, by comparing dose rates for given contents with the dose rates estimated by analysis for the same contents. Survey results that differ from estimated results, accounting for uncertainties in the measurements and the calculations, would indicate a problem with the as fabricated neutron shield. The tests in the currently approved CoC, fulfill these conditions. Based upon staff's considerations, the applicant included the acceptance tests and criteria for all shielding materials and modified the proposed acceptance and periodic neutron shield tests to retain the currently approved tests.

For the neutron shielding acceptance tests, the applicant describes two tests. The first test verifies shield integrity and is performed with either a check source (prior to first use) or the loaded contents at first use. This test examines the entire surface of the neutron shield, including the impact limiters. The measurements are compared with calculated values that are representative of either the check source or the loaded contents at first use. The second test, described as a shielding effectiveness test, is performed after the first fuel loading in a similar manner to the first test, except that the test does not cover the entire shield surface. This

second test is performed when a check source is used for the shield integrity test. Staff has reviewed these acceptance tests and finds that these tests are acceptable. This finding is based upon the verification of the performance of the cask's entire neutron shield, with the test measurements being compared versus calculated values for the respective test source.

The applicant, in response to staff's questions, also modified the proposed visual inspections in the acceptance tests in a manner that clarified the tests and their acceptance criteria.

Based upon its review of the descriptions in the application, the staff finds that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

The CoC has been revised as follows:

Condition No. 5(a)(2):

A packaging description HI-STAR HB was added.

Condition Nos. 5(a)(3):

Nine drawings were revised.

Condition No. 5(b)(1):

The definitions of damaged fuel, damaged fuel containers, and fuel debris were modified. The definition of undamaged fuel was added to account for the fuel specific to HB.

Condition Nos. 6.(a) and 6.(b):

Revisions were made to reference SAR Chapters 7 and 8 in the CoC.

Condition No. 7:

Revision to add the maximum gross weight of the HI-STAR HB to the CoC.

Appendix A:

Revisions were made to add some cooling times, burnups, and initial enrichments, add the fuel specifications for the HB fuel, and provide some clarifications.

Condition No. 20:

Allows the use of Revision 6 of this certificate for one year.

CONCLUSION

The staff has reviewed the requested amendment to Certificate of Compliance No. 9261. Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. Certificate of Compliance No. 9261 for the HI-STAR 100 transport package has been amended as requested by Holtec International.

Issued with Certificate of Compliance No. 9261, Revision No. 7,
on May 8, 2009.