Request for Additional Information
Holtec International, Inc.
Docket No. 71-9336
HI-STAR 60 Transportation Package

By letter dated August 30, 2007, Holtec International, Inc. (Holtec or applicant) submitted an application requesting a Certificate of Compliance (CoC) for the HI-STAR 60 transport package. This request for additional information (RAI) identifies information needed by the U.S. Nuclear Regulatory Commission staff (the staff) in connection with its review of the safety analysis report (SAR). The requested information is listed by chapter number and title in the SAR. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," was used for this review.

Each individual RAI describes information needed by the staff for it to complete its review of the application and to confirm whether the applicant has demonstrated compliance with regulatory requirements.

General Information

G-1 Revise the application to include a complete evaluation of the package with Indian Point 1 (IP1) fuel as a content, in accordance with 10 CFR 71.35, and clarify if the IP1 fuel is damaged or will be loaded in a damaged fuel canister; alternatively, delete the references to the IP1 fuel.

To authorize the package with the IP1 fuel content, the application should include a complete evaluation that demonstrates that the package, with the IP1 fuel, meets the performance standards in 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions. In several sections of the HI-STAR 60 Application, such as Sections 5.0, 5.2.1, 5.2.2, 1.1, 7.1, and B.1 of Report HI-2073722, the applicant compares the Qinshan Nuclear Power Center (QNPC) design basis source term to the IP1 source terms in Tables B-8, B-9, and B-10 (Section B.1). This section further states that "This comparison demonstrates that the QNPC design basis also bounds the IP1 fuel, and therefore the IP1 fuel is acceptable for transport in the HI-STAR 60 cask." However, there is not a complete evaluation of the package with the IP1 fuel as contents. To show that IP1 fuel is bounded by another fuel type (i.e., QNPC fuel), the application should include detailed calculations that demonstrate that it is bounded.

This information is needed to determine compliance with 10 CFR 71.35.

G-2 Remove all references to an alternative polymeric foam impact limiter from the application or provide a complete thermal and mechanical analysis of the package which incorporates the alternative polymeric foam impact limiter.

Section 2.2.1.1.5 states that "The candidate energy absorbing material evaluated for the HI-STAR 60 is Cross-Core Aluminum Honeycomb (Hexcel Corp.). A potential

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alternative candidate is “Last-A-Foam” FR-3700 rigid closed cell Polyurethane Foam (General Plastics Corp.).” A structural and thermal analysis under normal conditions of transport and hypothetical accident conditions with “Last-A-Foam” material properties has not been provided; therefore the staff cannot evaluate the performance of the material.

This information is needed to determine compliance with 10 CFR 71.33(a)(5), 10 CFR 71.35, 10 CFR 71.71, and 10 CFR 71.73.

G-3 Revise the discrepancies throughout the application by performing a search and replace on “180” or “100” to ensure the applicant does not mean “60.”

For example, Section 3.2.2 on page 3.2-2 of the application states, “To facilitate evaluation of cold events defined by transport regulations, the HI-STAR 180 package cold service temperatures are conservatively limited to -40°F (-40°C).” Section 3.2.1, on page 3.2-1 states, “Periodic thermal testing of the HI-STAR 100 is not required.” For each occurrence, confirm that the statement is applicable to the HI-STAR 60.

There appears to be errors throughout the application; it is not clear that the application evaluates the HI-STAR 60.

This information is needed to determine compliance with 10 CFR 71.31.

G-4 Provide and justify the cladding performance limits for the Qinshan Nuclear Power Center (QNPC) fuel.

The justification should describe the composition of fuel, cladding and structural components, fabrication and quality assurance of QNPC fuel assemblies, QNPC reactor conditions that impact cladding integrity, and other factors relevant to performance of cladding during Normal Conditions of Transport and Hypothetical Accident Conditions. The description should clarify any similarities or differences to the fuel assemblies and associated reactor conditions in the United States, in which cladding performance limits have been well established (e.g., Interim Staff Guidance Document No. 11, Rev. 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel”), for temperature limits, etc.). The application should verify if these limits are appropriate to the QNPC fuel.

This information is required by staff to assess compliance with the requirements of 10 CFR 71.51, 10 CFR 71.55, 10 CFR 71.71 and 10 CFR 71.72.

Chapter 1: General Information

1-1 Provide a definition of undamaged fuel assemblies to clarify the proposed contents of the package.

Based on the evaluation in the application, the contents of the package include only undamaged fuel assemblies (i.e., no damaged fuel).

The definition of "intact fuel assemblies" in the proposed Certificate of Compliance and the application appear to match the definition of undamaged fuel assemblies in Interim Staff Guidance Document No. 1, Rev. 2, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function."

Interim Staff Guidance Document No. 1, Rev. 2 provides guidance for providing a definition for undamaged and damaged fuel. Note that the term "intact" indicates no breach, i.e., no pinholes or hairline cracks, whereas the term "undamaged" may allow rods with this type of breech.

This information is needed to determine compliance with 10 CFR 71.33(b)(3), 10 CFR 71.51 and 10 CFR 71.55.

1-2 The application requests approval of spent fuel with stainless steel cladding and must be revised to address the issues associated with stainless steel cladding.

(a) Specify the appropriate temperature limits for the stainless cladding in the HI-STAR 60 application during loading, drying, normal and accident conditions, and provide a basis for these limits.

The temperature limits set in Interim Staff Guidance Document No. 11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," were developed for zirconium-clad fuel. It is unclear what affects (if any) the expected cladding temperature during loading, drying, normal conditions of transport (NCT), or hypothetical accident conditions (HAC) will have upon stainless steel cladding.

(b) Describe the mechanical integrity of the stainless steel cladding under NCT and HACs, and show that it is sufficient to maintain integrity under normal and accident conditions.

The mechanical properties of irritated stainless steel fuel rods may be significantly different from the mechanical properties of irradiated zirconium-based fuel rods.

This information is required by staff to assess compliance with the requirements of 10 CFR 71.5, 10 CFR 71.55, and 10 CFR 71.59.

1-3 Specify the internal pressure of the Qinshan Nuclear Power Center (QNPC) fuel rods and provide a reference for the reported pressure. Discuss the affect of the internal pressure on the mechanical stability of the fuel rods on the containment of radioactive particulate in the event of an HAC.

The internal pressure, of the QNPC fuel rods, which is not specified in the application, could affect the mechanical integrity of the QNPC fuel and containment of radioactive particulate under normal and accident conditions.

This information is required by staff to assess compliance with the requirements of 10 CFR 71.51.

Chapter 2: Structural Evaluation

2-1 Revise the structural analysis where applicable to demonstrate that the package meets the performance standards in 10 CFR Part 71.

The methodology used by Holtec to evaluate the structural performance of the package was not acceptable to the NRC staff. As discussed in a teleconference on August 8, 2008, Holtec committed to an alternate and revised methodology to demonstrate a reasonable assurance of safety.

This information is required by staff to assess compliance with the requirements of 10 CFR 71.71 and 10 CFR 71.73.

Reference:
"Ltr., T. Morin, Holtec Int'l, HI-STAR 60 Structural Model Telephone Call (Docket No. 71-9336) (TAC L24121)." ADAMS Accession Number: ML082240743

2-2 Clarify what reference is being cited to determine the lower bound for the elastic modulus of fuel assemblies which use stainless steel fuel rods, as given in Table 2.11.1, Section 2.11, of the application.

Table C.1 of "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," is cited as the source for the elastic modulus of the fuel assemblies using zirconium-based fuel rods, but it is unclear what source is cited to model the elastic modulus of fuel assemblies which use stainless steel fuel rods.

This information is needed to determine compliance with 10 CFR 71.43(f) and 10 CFR 71.55.

2-3 Evaluate the mechanical effects of the excess space around the smaller fuel assemblies.

Assemblies for use in this system have different widths; this allows the smaller assemblies to rattle in the baskets. This behavior could lead to fracture or other undesired or unforeseeable events during transport.

This information is needed to determine compliance with the requirements of 10 CFR 71.71 and 10 CFR 71.73.

2-4 Revise Section 2.1.2.2(iv) of the application to include:

(a) An acceptance criterion that states the impact limiters stay attached to the package after a 30-foot drop. If the criterion permits the impact limiters to detach from the package after the 30-foot drop, then justify that the package still meets 10 CFR Part 71 performance standards assuming the impact limiters become detached.

(b) An acceptance criterion that states the impact limiters do not bottom out after a 30-foot drop. If the criterion permits the impact limiters to bottom out after the 30-foot drop, then justify that the package still meets 10 CFR Part 71 performance standards assuming the impact limiters bottom out.

The applicant does not address the issue of post-accident impact limiter attachment or bottoming out with respect to acceptance criteria.

This information is needed to determine compliance with 10 CFR 71.73.

2-5 Demonstrate that the angle used for the slapdown analysis (7 degrees) is the worst-case orientation.

It is unclear what basis was used for this determination.

This information is needed to determine compliance with 10 CFR 71.73.

2-6 Provide a definition for the sealing performance of bolted joints, including a description of the components evaluated, evaluation method, and acceptance criteria, as applicable.

The second paragraph on page 2.1-5 in Section 2.1.2.1 of the application states that the "Stress analysis of the containment boundary under the Design Pressure is required to demonstrate compliance with the "NB" stress limits for the containment boundary material and to demonstrate the sealing performance of the bolted joints (See Chapter 4)." The staff requests a definition as to what level of sealing performance is required when evaluating design condition loads.

This information is needed to determine compliance with 10 CFR 71.71 and 10 CFR 71.73.

2-7 Provide an analysis for the hypothetical drop test accident conditions that demonstrates that fuel rod integrity is maintained when the maximum axial gaps are considered between the fuel assembly and the closure lid or base plate.

The application uses data from NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," wherein the evaluated load drops are associated with the accidental drop of the transfer cask or canister during transfer operations in preparation for putting the canister, containing spent nuclear fuel, into storage. These are real load drops and as such, no gaps were assumed to exist between the canister and transfer overpack and between the fuel assemblies and canister when impacting a *deformable* concrete substrate. In contrast, the 30-foot drop of the HI-STAR 60 onto an *unyielding* surface evaluates the integrity of the cask's containment boundary to withstand a hypothetical transportation accident event. For a transportation accident event, these gaps will exist and their effect on the impact forces to which the cask contents are subjected may be significant and must be evaluated. Since gaps have not been considered in the qualitative evaluation performed in the application Section 2.11, the methodology used therein has not necessarily demonstrated cladding integrity under the HAC of transport.

This Information is needed to show compliance with 10 CFR 71.73.

Reference:
U.S. Nuclear Regulatory Commission, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” NUREG-1864, March 2007.

2-8 Provide data which show that each alternate bolt material used for cask closure lid bolts has sufficient ductility and impact resistance under the design accident conditions at -29^{o}C (-20^{o}F).

Table 2.2.2 of the application, page 2.2-7, does not describe the ductility or impact resistance of the alternate bolt materials at low temperature 29^{o}C (-20^{o}F). However, the staff is unable to find, based upon American Society of Mechanical Engineers specifications, that these materials are immune to nil-ductility transition temperature (NDTT) issues or have sufficient toughness to meet design accident conditions.

Following the guidance in Section 5 of NUREG/CR-1815, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick,” “…in cases where a particular bolt is determined to be a fracture-critical component, the toughness requirements for that bolt should be specified at the same category level as other components of the system.”

Since the bolts in question are part of the containment boundary, demonstration of their NDTT performance and toughness is necessary.

The material specifications for the alternate bolting materials do not require Charpy impact testing at lower temperatures, as a mandatory part of the specification (some are supplementary tests), and the ductility requirements, where stated, appear to be inadequate compared to the canister material. Therefore, low temperature Charpy testing and ductility data should be provided to ensure the materials specified are adequate for the design conditions. Absent these tests, there is no assurance of the bolting material performance.

The staff accepts that SB-637-N07718 is immune to NDTT issues and has adequate toughness.

This information is necessary to determine compliance with 10 CFR 71.73.

Reference:
W. R. Holman and R. T. Langland. Lawrence Livermore Laboratory, “Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick,” NUREG/CR-1815, page 10, 1981.

2-9 Remove the statement, “Except for containment boundary components, ASME , [American Society of Mechanical Engineers] materials may be substituted by ASTM [American Society for Testing and Materials] materials” from the Licensing Drawings.

Only ASTM materials which have identical or superior properties and levels of quality to their ASME counterparts can be used as substitutes for ASME materials designated for use in components which are important to safety. Such substitutions must be approved by the NRC staff.

The NRC staff is only evaluating a specific design with specific materials designations.

This information is needed to determine compliance with 10 CFR 71.31(c) and 10 CFR 71.33(a)(5).

2-10 Revise the application to address the following, with respect to the codes, standards, and specifications for the materials of construction:

(a) Specify in the Bill of Materials, and throughout the application, the precise designation and specification (e.g., SA 240-304) of materials used in components that serve a safety function in transportation. In addition, American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PVC) materials should be used for components that serve a safety function.

Providing a name for a generic group of materials (e.g., “Stainless Steel”) in the Licensing Drawings and noting that the generic group of materials will meet the requirements of American Society of Mechanical Engineers (ASME) Code Section III Subsection NB does not sufficiently specify a material. Generic terms for subsets of steels (e.g., “Cryogenic Steel”) or broad classes of steels (e.g., “Carbon Steel” and “Stainless Steel”) can not be used to establish quantitative material properties.

(b) Citing specific material codes or manufacturing specifications as references (e.g., ASME Boiler and Pressure Vessel Code Section IID) for materials used in this application, tabulate the relevant mechanical and thermal properties of structural materials, steels, bolting materials, etc., over the temperature ranges of that are expected to be seen under Normal Conditions of Transport and Hypothetical Accident Conditions.

The NRC staff is only evaluating a specific design with specific materials designations. Tabulated values for the mechanical and thermal properties of materials which serve a safety function are required by NUREG-1609.

It should be noted that the thermal conductivity of “Carbon Steel” stated in the HI-STAR 60 package in Table 3.2.2 on page 3.2-4 is significantly different from the thermal conductivity of “Carbon Steel” listed in the HI-STAR 180 package (Table 3.2.2: page 3.2-4).

This information is needed to determine compliance with 10 CFR 71.33(a)(5) and 10 CFR 71.31(3)(c).

2-11 Address the following items with regards to the aluminum honeycomb used in the impact limiters:

(a) Clarify the “Crush Material” which is listed in the Bill of Materials.

The exact designation of the impact limiting material(s) used in the HI-STAR 60 package should be defined in the Licensing Drawings. Specifications for the crush material (e.g., density, grade, pre-crushed, orientation relative to the cask, etc.) should either be stated in the application and referenced in the License Drawings, or explicitly stated in the License Drawings.

(b) Supply detailed, tabulated, materials properties data for the impact limiting materials over the temperature ranges under which impact may occur. If the impact limiting is anisotropic, specify the materials properties of the impact limiting material in three orthogonal directions.

The properties of the impact limiter material must be specified in order to evaluate the performance of the HI-STAR 60 package. The mechanical properties of honeycomb materials are typically anisotropic. Specific and detailed test data is needed to support the statements in Table 4.3.1 of Holtec Report No. HI-2073725, “Finite Element Impact Analyses Supporting HI-STAR 60 SAR,” that the honeycomb is isotropic.

This information is necessary to determine compliance with 10 CFR 71.33(a)(5).

2-12 Revise Table 2.1.17, by replacing the reference to Regulatory Guides 7.11, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)” and 7.12, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)” with a reference to Table 2.1.12.

The staff considers that Table 2.1.12 provides more specific information than the general guidance in Regulatory Guides 7.11 and 7.12.

This information is necessary to determine compliance with 10 CFR 71.31(c) and 10 CFR 71.33(a)(5).

References:

U.S. Nuclear Regulatory Commission, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m),” Regulatory Guide 7.11, June 1991.

U.S. Nuclear Regulatory Commission, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m),” Regulatory Guide 7.12, June 1991.

2-13 The mechanical properties of the impact limiters may be affected by the temperatures seen under Normal Conditions of Transport (NCT).

(a) Revise Section 2.7 to evaluate the performance of the impact limiters under hypothetical accident condition assuming that the impact limiters will be at elevated temperatures, as demonstrated by the thermal analysis of the NCT.

(b) Provide impact limiter temperatures in Tables 3.1.2 and 3.1.4.

The FLUENT case file for NCT with solar insolation, appears to indicate that the impact limiter temperatures during NCT are high enough to affect the mechanical properties of the impact limiter. The impact limiter temperatures were not included in Table 3.1.2 or in Table 3.1.4.

This information is needed to determine compliance with 10 CFR 71.73.

2-14 Demonstrate or justify that the SA-516 Grade 70 gamma shield material has adequate ductility and impact resistance at -29^{o}C (-20^{o}F) to withstand the impact loads that would occur during a design basis accident 30-foot drop (with impact limiters), and the one (1)-meter drop puncture-test. The SA-516 Grade 70 material is specified to require a minimum average Charpy impact energy of 15 footpounds (ft.-lbs.) at -40^{o}C (-40^{o}F) for three specimens and a minimum of 10 ft.-lbs for any single specimen (Table 2.1.13). It is not clear that the impact energy absorbed (10 to 15 ft.-lbs.) translates to sufficient ductility and energy absorption to prevent cracking of the gamma shield with resulting radiation streaming.

This information is needed to determine compliance with 10 CFR 71.51(a)(2).

2-15 Clarify if SB637-N07718 is an alternate cask closure lid bolt material.

The list of potential materials in Table 2.2.7 on page 2.2-13 of the application does not include SB637-N07718, which is listed as a potential bolting material in Section 2.1.2.2 on page 2.1-11 of the application.

This information is needed to determine compliance with 10 CFR 71.33(a)(5).

2-16 Clarify if A106 grade C steel, which is listed in the Bill of Materials in the Licensing Drawings is the material of construction for the support rib hub.

A107 grade C steel is not mentioned elsewhere in the application. (Note: A106 grade C steel is not an American Society of Mechanical Engineers Code material, see RAI 2-10).

This information is needed to determine compliance with 10 CFR 71.33(a)(5).

Chapter 3: Thermal Evaluation

3-1 Provide the basis for the thermal properties of Holtite-A, including any effects from long term thermal or radiation degradation, and justify that the properties ensure thermal performance is "essentially unchanged."

Holtite-A is a polymer and may be subject to heat and radiation degradation, including weight loss. Section 3.2.1 of the application states that acceptance and periodic thermal tests are unnecessary. Periodic thermal tests will be required if the thermal performance (i.e., thermal conductivity) of Holtite-A cannot be shown to remain unchanged after being exposed first to thermal and then irradiation testing.

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii) and 10 CFR 71.73(c)(4).

3-2 Rewrite Section 3.4 to justify that the damage caused by hypothetical accident conditions (HACs) resulted in the most conservative fire analysis.

(a) Clarify why the impact limiter properties were changed to air during the post-fire cooldown.

The impact limiters properties were changed to air during the post-fire cooldown, but no justification was given in the application.

(b) Present additional fire analyses to include drop test damage effects on the impact limiters to ensure that the current analysis is the worst case scenario.

Section 3.4 should take into account how structural damage to the package incurred during drop and puncture accidents could affect the thermal HAC model.

This information is needed to determine compliance with 10 CFR 71.73.

3-3 Given the temperature limits of the seal material, the small seal temperature margins under hypothetical accident conditions (Section 3.4), and other potential uncertainties or errors (e.g., RAIs 3-2 and 3-5), provide the following information:

(a) Include the expected stored thermal energy that the neutron shield material and impact limiter may contribute to the post-fire accident model.

In the post-fire analysis the neutron shield material and impact limiter are assumed to have the material properties of air, which is significantly more insulating than the initial materials. This approach is generally considered to be adequate for the thermal analysis of Part 71 transport packages under hypothetical accident conditions.

In light of the small temperature margin for the seal material, however, the applicant should also consider taking into account the expected stored thermal energy that the neutron shield material and the impact limiters would actually contribute to the post-fire accident model.

(b) Explain the different heat capacity values of air used in the post-fire thermal analysis.

The heat capacity value for the impact limiters and Holtite-A in the post-fire FLUENT case file is different from the heat capacity value for air-fluid in the same model. This discrepancy needs to be clarified, and if necessary, a new post-fire cooldown analysis should be provided.

This information is needed to determine compliance with 10 CFR 71.73(c)(4).

3-4 Provide the derivation of the convective heat transfer coefficient that was generated from the Sandia National Laboratory report for large pool fires. Also, provide the ratio between convective heat transfer and radiation heat transfer and justify the convective heat transfer coefficient is applicable for the HI-STAR 60 configuration.

According to the application, the convective heat transfer coefficient used in the fire analysis 4.5 BTU/(ft^2-hr-°R) is obtained from the Sandia National Lab experiment results. The Sandia report, however provided the heat flux measurement instead of heat transfer coefficient. The heat transfer coefficient used in the application is a derived value. The applicant should provide the derivation. In addition, based on the FLUENT calculation, the applicant should provide the ratio between convective heat transfer and radiation heat transfer and compare it to the Sandia Lab experiment results.

Reference
"Thermal Measurements in a Series of Large Pool Fires," Sandia Report SAND85-0196TTC-0659UC-71, August 1987

This information is needed to determine compliance with 10 CFR 71.73(c)(4).

3-5 Perform an in-depth thermal analysis of the hypothetical fire accident.

(a) Provide a temperature vs. time graph showing temperatures of critical components over the entire duration of the hypothetical fire accident, including post-fire cooldown.

Temperature vs. time plots are necessary for the staff to assess the integrity of critical components (e.g., seals) during the fire accident. The plot should clearly show that the analysis considered a sufficiently long time period and with appropriate time steps to show peak temperatures for all components.

This information is needed to determine compliance with 10 CFR 71.73(c)(4).

(b) Provide the results of detailed thermal analysis of the lid seal region. Justify the lid seal temperature model is adequate in the fire analysis and that the maximum temperature of the seal is reached in the analysis.

In Table 3.1.4 of the SAR, the lid seal temperature is 203°C. The application should describe how the lid seal temperature is determined considering the seal is not modeled in FLUENT. The lid seal region includes the grooves, elastomeric seal, and residual gases. The contact resistance and the seal material property should affect the lid seal temperature in the thermal analysis. Currently no detailed lid seal region is modeled. The lid seal provides containment and the applicant should demonstrate that the seal will possess an acceptable integrity during the fire condition. In light of the extremely small margin (1°C) toward the limit (204°C), the applicant should analyze the lid seal region in detail and demonstrate the lid seal temperature complies with the limits.

This information is needed to determine compliance with 10 CFR 71.35, 10 CFR 71.55, and 10 CFR 71.73(c)(4).

3-6 Clarify the statement "The maximum local temperature in the primary and secondary lid seals are lower than the design limits."

Section 3.4.6 of the application states, "The maximum local temperature in the primary and secondary lid seals are lower than the design limits." Yet there appears to be no secondary lid in the HI-STAR 60.

This information is needed to determine compliance with 10 CFR 71.33(a)(5) and 10 CFR 71.73.

3-7 Provide the heat capacity and density for air in Table 3.2.5 of the application.

This information should be provided to confirm that accuracy of the thermal analyses.

This information is needed to determine compliance with 10 CFR 71.71 and 10 CFR 71.73.

3-8 Provide estimates of all uncertainties in the calculation of the maximum cladding temperature reached during vacuum drying for the Qinshan Nuclear Power Center (QNPC) fuel, and verify that the cladding will not exceed the temperature and/or stress limits established in RAI G-4.

If the application intends to demonstrate that Interim Staff Guidance Document No. 11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel" (ISG-11) are appropriate limits for low burnup QNPC fuel, then the fuel cladding temperature may be allowed to exceed 400°C (752°F) during drying, provided that, "the applicant can show by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed." Given the lack of calculated margin between the fuel cladding temperature during vacuum drying and the maximum cladding temperature of 400°C (752°F) in ISG-11 for U.S. fuel, the application should verify that the maximum cladding temperature reached during drying is conservatively realistic. The application should also use the internal pressure of the fuel rods specified in RAI 1-3.

For other transport cask designs with U.S. fuel contents, the Pacific Northwest National Laboratory report below has been referenced as justification that a calculation of the hoop stress does not have to be performed if the cladding temperature is within the range of 400 – 570^{o}C (752 – 1058 ^{o}F). If this approach is proposed for the QNPC fuel, then the application should demonstrate the QNPC fuel is within the analytical bounds of the given reference.

This information is required by staff to assess compliance with the requirements of 10 CFR 71.51.

Reference:
D. D. Lanning and C. E. Beyer, Pacific Northwest National Laboratory, January 2004. “Estimated Maximum Cladding Stresses For Bounding PWR Fuel Rods During Short Term Operations For Dry Cask Storage.”

3-9 Clarify the type of symmetry used to model the HI-STAR 60 as part of the thermal evaluation.

There appears to be a discrepancy between the symmetry used in the thermal model of the HI-STAR 60, when comparing the description in Section 3.3, page 3.3-4 of the application with the model presented in Figure 3.3.2, page 3.3-12, of the application.

This information is needed to determine compliance with 10 CFR 71.71 and 10 CFR 71.73.

Chapter 4: Containment

4-1 Clarify the design leakage rate of closure lid elastomeric seals in the summary of the containment boundary design specifications (Table 4.1.1 of the application).

The definition of the design leakage rate and the basis of the numerical value (1.0e-5) should be clearly stated. The difference between the design leakage rate and the leakage rate acceptance criterion in the same table should be clearly defined.

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

4-2 Provide justification that the seal material meets all the critical characteristics specified in Table 2.2.8 of the application. Specify the exact type or types of methyl vinyl silicone rubber that are to be used for the elastomeric sealing material in the application and Licensing Drawings (The term “Elastomeric” is not sufficiently precise). If the material is to be purchased from Parker Hannifin Corporation, specify the Parker compound number.

The applicant does not provide enough detail to show how the silicone rubber material qualifies for the critical characteristics specified in Table 2.2.8. Given the importance of the seal material for maintaining containment, the applicant should provide quantitative values for the critical characteristics of the material (stating that the material has “good low temperature flexibility” is not sufficient). The supporting documentation should cite substantive references, e.g., technical reports, nationally recognized industry standards, etc. Documentation in general sales catalogs is not acceptable, but data sheets for specific materials from well-established vendors may be considered.

The high temperature stability of the material is of particular concern to the staff.

This information is needed to determine compliance with 10 CFR 71.33(a)(5), 10 CFR 71.43(d), and 10 CFR 71.51.

4-3 Provide details for the sealing surface and the groove for the seal in the Licensing Drawings.

(a) Specify the half dovetail groove entrance edge roundness radius in Licensing Drawing 5238, Sheet 5, DETAIL P.

According to the Parker O-ring Handbook, the groove entrance edge roundness radius is critical to o-ring installation with respect to potential damage or extrusion. This radius is specified according to the o-ring size. This dimension is not shown in the drawing. The applicant should specify the radius in Licensing Drawing 5238, Sheet 5, DETAIL P.

(b) Specify the surface finish of the sealing surfaces in the Licensing Drawings.

The surface finish of the sealing surfaces should be specified to ensure proper operation of the seal.

This information is needed to determine compliance with 10 CFR 71.33(a)(5).

4-4 Provide a detailed void volume calculation for the Multi-purpose canister (MPC) and the Enclosure Vessel in Table C-2 of the HI-STAR 60 containment analysis (Holtec Report No. HI-2073728, HI-STAR 60 Containment Analysis).

In Table C-2, the value of MPC and Enclosure Vessel volumes are listed with no supporting calculations.

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

4-5 Add a closure lid inner elastomeric seal leakage test, and a vent/drain port cover inner elastomeric seal leakage rate test into the fabrication leakage rate test.

According to Section 7.3 of American National Standards Institute N14.5, “Radioactive Materials - Leakage Tests on Packages for Shipment,” the fabrication leakage rate test involves all joints and seams on the containment system to the extent practicable. It does not limit the fabrication leakage rate test to the welded joints only. Lid seal and vent/drain port cover seal are part of the containment system. The leakage rates for these components should be included in the fabrication acceptance test category in Table 4.3.1 of the application.

This information is necessary to confirm that the package meets the requirements of 10 CFR 71.37 and 10 CFR 71.51.

4-6 Clarify that the frequency of pre-shipment leakage rate tests for the HI-STAR 60 packaging will meet the requirements of Section 7.6 of American National Standards Institute (ANSI) N14.5-1997, "Radioactive Materials - Leakage Tests on Packages for Shipment," and remove any statement in the application that directly or indirectly implies that a pre-shipment leakage rate testing could be mitigated.

Section 4.3.2 of the application states that "Pre-shipment leakage rate testing is performed by the user before each shipment, after the contents are loaded and the containment system is assembled (if not previously tested in the prior 12 months excepted as indicated in Section 7.3)." In Section 8.2.2, the applicant states "The elastomeric seals on the cask containment boundary seals shall be replaced as defined in Table 8.2.1. After each replacement, a helium leak test of the seals shall be performed." According to ANSI N14.5-1997 Section 7.6, the pre-shipment leakage rate is required for each shipment regardless of prior tests, particularly after reloading and seal replacements.

This information is needed to determine compliance 10 CFR 71.51 and 10 CFR 71.127.

4-7 In Section 3.2.2, clarify the term overpack seals and clarify if the overpack seals are leak tight according to American National Standards Institute (ANSI) N14.5-1997, "Radioactive Materials - Leakage Tests on Packages for Shipment."

In the second paragraph of Section 3.2.2 the applicant states "The overpack seals will continue to ensure leak tightness if [the] manufacturer's design temperature limits are not exceeded." Clarify if the overpack seals refer to the lid seals, if not, provide a description and temperature limits for the overpack seals. Clarification is needed regarding the leak tightness of the overpack seals, specifically if they are leak tight according to ANSI N14.5-1997.

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

4-8 Clarify if the seals will be submerged in the spent fuel pool.

In Table 2.2.7 of the application, the elastomeric seals are, "Not installed or exposed [to the spent fuel pool] during in-pool handling." According to Loading Operation, step 4 in Section 7.1.2.1 of the application, however, the seal installation is performed underwater in the spent fuel pool.

This information is needed to determine compliance with 10 CFR 71.43(d).

Chapter 5: Shielding Evaluation

5-1 In the Bill of Materials, clarify the use of Holtite-A as the neutron shielding material used in the HI-STAR 60 package.

The application and Note 10 on Sheet 1 of 7 of the Licensing Drawings state or imply that Holtite-A is used as the neutron shielding material in the HI-STAR 60. The Bill of Materials however, specifies a generic grade of Holtite as the neutron shielding material. Since there are at least two different grades of Holtite available, the applicant should confirm in the Bill of Materials, that the Holtite-A is used in the HI-STAR 60.

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii).

5-2 List uncertainties associated with each of the maximum dose rates shown in Tables 5.1.1 and 5.1.2, and the dose rates listed in Tables 5.4.3 through 5.4.10 from the Monte Carlo N-Particle (MCNP) calculations.

The tables aforementioned list maximum dose rates and individual dose rates at various distances from the HI-STAR 60 packaging system for both normal and accident conditions without specifying the uncertainties associated with the calculations. The application does not contain any statement regarding the uncertainties in the results. In order to determine the real margin of safety in the dose calculations, uncertainties corresponding to each of the items listed in the above tables are required.

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

5-3 Justify that the source distribution of fuel is not reconfigured after accident conditions sufficient to exceed the regulatory limit.

The calculated accident dose rates appear to be within the regulatory limit of 10 CFR 71.51. It is not clear, for example, if the source term in localized areas will not be increased as a result of potential damage to the fuel, and raise the accident dose rate closer/greater than the regulatory limit.

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

Chapter 6: Criticality

6-1 Provide a discussion on the experiments used to perform the criticality benchmarks. Provide justification regarding the inclusion of experiments for mixed oxide fuel (MOX) fuel (PNL-5803 and WCAP-3385). Discuss how the bias might change without the inclusion of the MOX experiments.

Based on the information provided, the staff requires additional information about the experiments used to benchmark the Monte Carlo N-Particle (MCNP) code to determine that the bias was appropriately determined for the HI-STAR 60. The applicant should provide a discussion of which experiments were used to determine the trend (or lack thereof) with the various parameters important to criticality (enrichment, effect of ^{10}B loading, etc.). In addition, it is not clear that the inclusion of MOX experiments is applicable, and since these experiments typically have a positive bias, they may be providing a non-conservative influence on the bias for the HI-STAR 60.

This information is needed to determine compliance with 10 CFR 71.31(a)(2), 10 CFR 71.35, 10 CFR 71.55, and 10 CFR 71.59.

Chapter 7: Package Operations

7-1 Describe the procedures that are in place to prevent water boiling in the HI-STAR 60.

Recent operating experience has indicated that a handling malfunction could suspend transfer operations of a fully loaded canister and require alternate cooling that may take several hours to be implemented. Describe the procedures that are in place to assure adequate cooling can be established for the HI-STAR 60 in the event of a handling malfunction or other adverse circumstances before the time-to-boil limit has been reached.

This information is needed to determine compliance with 10 CFR 71.43, 10 CFR 71.51, 10 CFR 71.55, and 10 CFR 71.59.

7-2 Provide details describing the thermal behavior of the package during loading operations in Chapter 7 of the application.

(a) Provide a reference to the Holtec Report HI-2073740 “Thermal Analysis of the HI-STAR 60” for the method of evaluating site specific “Time-to-boil” criteria in Section 7.1.2.1.4. Also provide a summary of “Time-to-boil” values in Chapter 7. Demonstrate how the users will avoid violating the “Time-to-boil” criteria.

Chapter 7 should clearly state how “Time-to-boil” criteria will be met by the users.

(b) Specify in Chapter 7 that the change in fuel cladding temperature is restricted to less than 65°C (117°F) and is limited to less than 10 cycles during drying, backfilling, and transfer operations.

Limitations on the fuel clad temperature are set in Interim Staff Guidance No. 11, Rev. 3. “Cladding Considerations for the Transportation and Storage of Spent Fuel” to prevent reorientation of embrittling zirconium hydride phases within the cladding.

This information is needed to determine compliance with 10 CFR 71.43, 10 CFR 71.51, 10 CFR 71.55, and 10 CFR 71.59.

7-3 Describe the seal installation procedures, including the underwater sealing surface inspection, sealing surface condition requirements, seal installation, and seal replacement in detail. Explain how these procedures and criteria are met in the sealing operation. Clarify what lubricant (if any) is applied to the seals in air or underwater.

The sealing of the HI-STAR 60 is not sufficiently described in the loading operations (Section 7.1.2.1 of the application). In addition, the sealing surface condition should be described in the application (see item 4-3b).

According to the o-ring handbook referenced by the applicant, the lubricant aids the seal when applied in air. It is unclear, however, if any lubricant is applied to the seal of HI-STAR 60, and if that lubricant is applied to the seal in air or water.

This information is needed to determine compliance with 10 CFR 71.33 and 10 CFR 71.51.

7-4 Describe detailed torque measurement procedures in the loading operations (Section 7.1.2.1 of the application) and clarify if the torquing procedure takes place underwater. Also evaluate the affect of thermal expansion on the bolt lid torque.

In the loading operations the torquing procedure is not described in sufficient detail. Step 4 and Step 5 of the loading operations imply that the bolt lid torquing is performed under water. It is not clear whether there is a step in the loading operations to lower the cask water in the lid region before torquing. If not, thermal expansion of the sealed cask cavity water may affect the torque measurement.

This information is needed to determine compliance with 10 CFR 71.37 and 10 CFR 71.51.

7-5 Clarify the term “sufficient concentration” of helium in Step 1 of the cask closure procedures (Section 7.1.2.2 of the application) and the method for which gas is introduced into the space beneath the cover lid port cover prior to testing the plugs.

The application should demonstrate that an adequate amount of helium will be used to assure any leakage would be identified during helium leak testing.

This information is needed to determine compliance with 10 CFR 71.37 and 10 CFR 71.51.

7-6 Clarify in Section 7.1.2.1, subsection 10 of the application that the pressure should be maintained below 3 torr for 30 minutes with the valve being closed, so as to isolate the system from the pump that is used to decrease the pressure inside the system.

The cask drying operation is critical to the spent fuel cladding integrity. The licensee needs to provide the adequate dryness criteria with the valve being closed to ensure the dryness criteria are reached in the cask.

This information is required by staff to determine compliance with the requirements of 10 CFR 71.51 and 10 CFR 71.55.

7-7 Specify in Chapter 7 that an inert gas will be used to backfill the canister while draining the canister of water.

The cladding must remain in a non-oxidizing atmosphere during all loading operations.

This information is required by staff to determine compliance with the requirements of 10 CFR 71.51 and 10 CFR 71.55.

Chapter 8: Acceptance Tests and Maintenance Program

8-1 Add shielding integrity and effectiveness tests to Chapter 8, of the application.

The staff does not agree that with the licensee's statement in Section 8.1.6 of the HI-STAR 60 application that "Shielding tests are not required for the assembled packaging." A shielding integrity test, however, is specified in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," Section 8.2.4.6, and in Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material," Section 8.16.

The integrity of the neutron and gamma shielding materials should be tested to verify the material composition, boron concentration, neutron shield density as part of the fabrication testing control process. The installation of the shielding materials should be performed with well documented quality assurance procedures. Further, users of the HI-STAR 60 implement should quality assured procedures to verify the integrity and effectiveness of Holtite-A neutron shield for each overpack. Finally, shielding effectiveness test should be performed for each package at the loading facility site to verify the effectiveness of the gamma and neutron shields using written and approved procedures.

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

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

U.S. Nuclear Regulatory Commission, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material," Regulatory Guide 7.9, Revision 2, March 2005.

8-2 Add periodic shielding tests in the "Maintenance Program" in Chapter 8 of the application.

The radiation survey specified in Chapter 7 of the SAR only ensures that the package meets 10 CFR Part 71 dose limits for a particular shipment. The periodic maintenance test should verify that the neutron shield performs as designed for any given contents and the verification should involve comparison of dose rate measurements for any given contents with values calculated for the same contents.

This information is required to confirm that the maintenance program adequately assures the package effectiveness throughout the service life of the package and to determine compliance with 10 CFR Part 71 Subpart E.

8-3 In Section 8.1.4 of the application, revise and include a table that specifies all leakage rate tests, the components tested, and the types of leakage rate tests from American National Standards Institute (ANSI) N14.5-1997. Also include the leakage rate acceptance criterion, and the leakage rate test sensitivity.

In Section 8.1.4, the applicant should include a table for all leakage rate test details for the package. The table should include a fabrication leakage rate test, a pre-shipment leakage rate test, a maintenance leakage rate test and a periodic leakage rate test. The table should list the components tested, the type of leakage rate test from ANSI N14.5-1997, the leakage rate acceptance criterion with numerical values, and the leakage rate test sensitivity with numerical values.

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

8-4 Specify in the proposed Certificate of Compliance that with the Code exceptions listed in Table 2.1.17, that the HI-STAR 60 will be designed, manufactured, and tested according to Section III NB of the American Society of Mechanical Engineers (ASME) Code.

The HI-STAR 60 transportation package should be designed, manufactured, and tested according to Section III NB of the ASME Code.

This information is needed to determine compliance with 10 CFR 71.31(c) and 10 CFR 71.47.

8-6 Describe in greater detail the “closure fasteners” mentioned in Section 8.2.3.6 of the application, and clarify the difference between “closure fasteners” and “closure plate bolting” (Section 2.4.2).

There is ambiguity in the application regarding the closure fasteners and closure plate bolting which should be clarified.

This information is needed to determine compliance with 10 CFR 71.33(a)(5) and 10 CFR 71.51.

8-7 Clarify the maintenance schedule for the elastomeric seals.

Table 8.2.1 in the application specifies that the closure lid seals will be replaced following the removal of closure plate bolting. Section 8.2.3.6, of the application however, states that “Removal of closure fasteners may require replacement of closure seals, if recommended by the seal manufacturer.” These two statements, which do not appear to require the same maintenance schedule for the elastomeric seals, should be reconciled.

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

8-5 Specify that the basket and basket shim welds mentioned in Section 8.1.2 of the application meet the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (B&PVC) Section III NG. Note, however, that any attachment made to the confinement boundary (including temporary welds) should meet the more stringent requirements of ASME B&PVC Section III NB.

Sub-paragraph three of Section 8.1.2 of the application states that basket and basket shim welds are non-code welds, which is *not* correct, and appears to be inconsistent with Section 2.2.1.1.3 of the application. Basket and basket shim welds should be made to ASME B&PVC Section III NG requirements, as specified in Table 1-1 of NUREG-1617. In addition, any attachment to the confinement boundary (including temporary welds) should meet ASME B&PVC Section III NB requirements.

This information is needed to determine compliance with 10 CFR 71.31(3)(c).

8-6 Confirm that the term “or equivalent” used in Section 8.1.2, sub-paragraph 2 of the application means the examination requirements and acceptance criteria of “equivalent” welds will be identical to those in the referenced American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

The term “or equivalent” as used in Section 8.1.2, sub-paragraph 2 of the application should be clarified or removed.

This information is needed to determine compliance with 10 CFR 71.31(3)(c).

8-7 Revise Section 8.1.2 to clarify or remove the term “or equal” used in the opening paragraph regarding the personnel qualifications of weld inspectors to the standards of the American Society for Nondestructive Testing, SNT-TC-1A.

The NRC staff requests clarification of removal of the term “or equal” as used in the opening paragraph of Section 8.1.2 of the application.

This information is needed to determine compliance with 10 CFR 71.31(3)(c).

8-8 Provide and justify the protocol for acceptance test sampling of METAMIC® plates that are to be taken from each lot of plates used in the production of casks for service.

Section 8.1.5.4 does not describe the acceptance testing protocol in sufficient detail. The applicant should consider (among other details) providing, the size of the samples used in wet chemistry tests and neutron-beam attenuation measurements, the beam size used in neutron attenuation experiments, the statistical basis that is used to verify the boron carbide weight or volume fraction in the METAMIC® absorber material, the ratio of the total area sampled to the total area of the neutron absorbing plates, the average particle size, and nominal particle size distribution of the boron carbide powder, etc.

Neutron attenuation tests should not be conducted on “pre-production trial runs” for verification of criticality safety. Acceptance testing should not be done on samples deliberately chosen from the sides of METAMIC® plates. Any thickness reduction that occurs in the plate materials must be accounted for in coupons samples randomly selected from the ends of METAMIC® plates.

This information is needed to determine compliance with 10 CFR 71.33(a)(ii) and 10 CFR 71.55.

8-9 Clarify whether each plate of neutron absorber material will be visually inspected for damage, and specify that the edges of the each plate will not contain fissures, fractures, cracks, or anomalies that are abnormal to the plate material in general.

Section 8.1.5.4 of the application states, “Each plate of neutron absorber shall be visually inspected for damage such as scratches, cracks, burrs, peeled cladding, foreign material embedded in the surfaces, voids, delamination, and surface finish, as applicable.” For the staff, all of these conditions seem to be applicable for inspections of neutron absorber plate materials.

This information is needed to determine compliance with 10 CFR 71.33(a)(ii) and 10 CFR 71.55.

8-10 Specify and justify that sufficient numbers of neutron attenuation tests will be conducted to establish the validity of wet chemistry results from acceptance tests conducted on actual production runs of METAMIC®.

Results of wet chemistry methods alone can not be used to validate the uniformity of the neutron absorbing plates without adequately bench marking those results against results from a sufficient number of neutron attenuation measurements.

This information is needed to determine compliance with 10 CFR 71.33(a)(ii) and 10 CFR 71.55.

8-11 Describe the critical characteristics and fabrication procedures for METAMIC® neutron absorbing plates in Chapter 8 of the application.

Control of the processing variables (e.g., particle size of the boron carbide powder, aluminum alloy designation, etc.) and the fabrication procedures (cold-isostatic pressing, extrusion, rolling, etc.) for METAMIC® neutron absorbing plates is important to manufacturing neutron absorbing plates with consistent materials properties.

This information is needed to determine compliance with 10 CFR 71.33(a)(ii) and 10 CFR 71.55.

8-12 Reference appropriate documents or technical reports which detail and control the critical characteristics, fabrication procedures, and testing of METAMIC® neutron absorbing plates.

Control of the critical characteristics, fabrication procedures, and testing of METAMIC® neutron absorbing plates are essential to the safe operation of the HI-STAR 60 package, and have been described in technical reports which the applicant has referenced in the past. These controlling documents should be referenced so that these reports become part of the Certificate of Compliance.

This information is needed to determine compliance with 10 CFR 71.33(a)(ii) and 10 CFR 71.55.

Editorials:

E-1 Revise Table 2.1.1 to label the pressure limits as gauge pressure (psig).

There appears to be a typographical error in Table 2.1.1.

E-2 Remove the statement “The maximum temperature of the containment boundary is well below the accident temperature limit” from Section 3.4.3.1 of the application.

Section 3.4.3.1 of the application states, “The maximum temperature of the containment boundary is well below the accident temperature limit.” Yet in Table 3.1.4 of the application the containment boundary lid seals temperature is reported to be 203°C (397°F) during the fire and post-fire cooldown while the accident limit is 204°C (400°F). In this case, the term “well below” does not appear to accurately represent a difference of 1^{o}C (< 1^{o}F).

E-3 Remove the dose rate computations using International Commission on Radiological Protection (ICRP) 74 conversion coefficients from the HI-STAR 60 application. Only the dose rates calculated using American National Standards Institute (ANSI) / American Nuclear Society (ANS) 6.1.1 – 1977 – Flux-to-Dose Conversion Factors need to be shown in the application.

Tables 5.1.1, 5.1.2, 5.4.3, 5.4.4, 5.4.5, and 5.4.6 list dose rates for various cases using the ANSI/ANS-6.1.1 Flux-to-Dose factors. Tables 5.4.7, 5.4.8, 5.4.9, and 5.4.10 list dose rates for various cases using the ICRP-74 Dose Response Functions. The applicant also provided dose rates for non-exclusive use per 10 CFR 71.47(a). The applicant has not addressed a rationale in the application for providing the two sets of dose rates, one based on ANSI/ANS standard and the other based on ICRP 74 standard. Furthermore, NRC regulations do not require applicants to provide dose rates based on ICRP 74 conversion factors. Therefore the applicant should remove these Tables with dose rates calculated with ICRP 74 conversion factors and any reference to them.

E-4 Correct Table 1.2.1 to indicate that the correct number of Guide and /or Instrument Tubes.

Table 5.2.1 lists the Number of Guide Tubes as 20, and Number of Instrument Thimbles as 1 making a total of 21. Table 1.2.1 “Fuel Assembly Physical Characteristics,” lists Number of Guide and/or Instrument Tubes as 20.

E-5 Correct the typographical error in Table 6.A.1 page 6.A-10.

Experiments 38, 39, and 40 cited as PNL-3626 appear to be a typographical error. The reference (6.A.12) appears to have these experiments correctly listed as PNL-3926.

E-6 Re-phrase the wording regarding closure seals in Section 8.1.5 of the application.

The applicant stated “Cask closure seals are self-energized elastomer seals and conservatively specified to provide a high degree of assurance of leak tightness under normal and accident conditions of transport.” The HI-STAR 60 package is not a leak-tight package. In addition, the NRC staff is not familiar with the term “self-energized” when applied to elastomeric materials. Revise the wording to avoid confusion.

E-7 The alloy designated, “Steel Material GH4169A” in Table A-3 on page A-4 of Appendix A in the Shielding Evaluation contains less than 19% iron, and thus, can not be defined as a steel. By definition, steels require at least 50% iron.

E-8 Confirm the composition of “Steel Material 0Cr18NilOTi” in Table A-3 on page A-4 of Appendix A of the Shielding Evaluation.

The alloy designated, “Steel Material 0Cr18NilOTi” in Table A-3 on page A-4 of Appendix A of the Shielding Evaluation would appear, by its own designation to contain titanium, but according to the composition listed in Table A-3, 0Cr18NilOTi actually contains no titanium. This may be a typographical error on the part of the applicant.

E-9 Revise the discrepancy in Table 3.4.1 of the application.

In Table 3.4.1 of the application, Row 5 refers to the conduction of the impact limiter throughout the fire analysis. Yet, the column for post-fire equilibrium states that air was assumed to replace the neutron shield.

E-10 Correct the reference given on page 3.4-2, [R.L], in Section 3.4.2 of the application.

This reference appears to be a typographical error and should be as [R.L], not [R.M].

E-11 Clarify the wording regarding the “containment system boundary” in Chapter 4 of the HI-STAR 60 application. Also clarify what components are apart of the containment boundary and the containment system.

The applicant states that the containment system boundary includes the closure lid bolts and vent/drain port bolts and vent/drain port test plugs. It appears that the term should be “containment system” as defined in 10 CFR 71.4, however.

The aforementioned components do not belong to the containment boundary, as the containment boundary is traditionally defined by the potential leakage path of radioactive materials. These components are outside of containment boundary.

This information is needed to determine compliance with 10 CFR 71.51

E-12 Clarify the mismatch of nuclides in Table A-8 and Table A-1 in the HI-STAR 60 containment analysis (Holtec Report No. HI-2073728, “HI-STAR 60 Containment Analysis”).

In Holtec Report HI-2073728, the nuclides in Table A-8 which are included in the effective A_2 calculation for fines are different than the design basis fuel assembly inventory listed in Table A-1.

E-13 Remove all references to a 40-year design life from the application.

The Certificate of Compliance will include a statement to the effect that HI-STAR 60 is certified for transportation only, and is not intended for long-term storage.

The HI-STAR 60 is a transportation cask, and is not intended for long-term storage of spent nuclear fuel.