SAFETY EVALUATION REPORT
Docket No. 71-9336
Model No. HI-STAR 60 Package
Certificate of Compliance No. 9336
Revision No. 0

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SUMMARY

By application dated August 27, 2007, as supplemented November 21, 2008, and May 15, 2009, Holtec International requested approval of the Model No. HI-STAR 60 as a Type B(U)F-96 package. Revision No. 2 of the package application, dated May 15, 2009, superseded in its entirety the application dated August 27, 2007.

The Model No. HI-STAR 60 package consists of a metal cask designed to hold 12 Pressurized Water Reactor (PWR) irradiated fuel assemblies in a fuel basket with Metamic neutron absorber panels fixed to the basket cell walls. The basket provides criticality control and the cask provides containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection. The fuel impact attenuators mitigate the G loads on the fuel assemblies due to secondary internal impact. Fastener strain limiters limit the axial stress imparted to the impact limiter attachment bolts. The outer diameter of the package is approximately 1,924 mm without the impact limiters. The maximum gross weight of the loaded package is 74.4 metric tons.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport and hypothetical accident conditions. The analyses performed by the applicant demonstrate that the package provides adequate thermal protection, containment, shielding, and criticality control under normal and accident conditions.

NRC staff reviewed the application using the guidance in "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617. Based on the statements and representations in the application, and the conditions listed in the certificate of compliance, the staff concludes that the package meets the requirements of 10 CFR Part 71.

References

Holtec International Report No. HI-2073710 “Safety Analysis Report on the HI-STAR 60 Transport Package,” Revision No. 2, dated May 15, 2009.

1.0 GENERAL INFORMATION

The Model No. HI-STAR 60 package is a Type B(U)F-96 package designed for the transport of radioactive material including commercial spent fuel assemblies, reactor-related Greater Than Class C (GTCC) waste and High Level Waste under either exclusive use or non-exclusive use shipment depending upon the temperature of the accessible package surfaces. The present application considers commercial irradiated undamaged PWR fuel assemblies as the package's only authorized contents.

1.1 Packaging

The HI-STAR 60 packaging is a multi-layer steel cylinder with a welded base-plate and a bolted lid (closure plate). The multi-layer shell provides a natural barrier against crack propagation in the radial direction across the cask structure. The outer surface of the package inner shell is buttressed with intermediate steel shells for gamma radiation shielding. The cask closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the cask inner shell, the bottom plate, the top flange, the top closure plate, the top closure inner O-ring seal, the vent port plug and seal, and the drain port plug and seal.

The inner shell of the packaging forms an internal cylindrical cavity that houses the fuel basket. The fuel basket offers a complete edge-to-edge continuity in the cell walls of the basket to provide for an uninterrupted heat transmission path.

Impact limiters are installed at each extremity of the package to protect it under all angular drop orientations. The impact limiter's aluminum honeycomb crush material and neutron shield are completely enclosed by an all-welded stainless steel skin. The HI-STAR 60 package is engineered for transport by either rail, road, or by a sea going vessel using appropriate packaging supports and restraints. The transport cradle, the longitudinal stops, the support saddles, the tie-down systems and wedge shims are non-integral appurtenances to the package and, as such, are not designated as packaging components.

The approximate dimensions and weights of the packaging are:

Inside Diameter of the Cask Cavity:	1,080 mm
Length of the Cask without Impact Limiters:	4,037 mm
Length of the Cask with Impact Limiters:	6,969 mm
Nominal Empty Packaging Weight:	64,650 kg
Maximum Gross Weight of the Package;	74,400 kg

1.2 Contents

The fuel basket holds twelve (12) PWR undamaged irradiated fuel assemblies with each basket cell holding one fuel assembly. Damaged fuel is not permitted for transport. The active fuel length of a PWR fuel assembly is 2,900 mm.

Table 1.2.1 of the package application lists the physical characteristics of the 15x15 PWR fuel assemblies to be transported. The fuel assemblies have a maximum burnup of 45,000

MWd/MTHM, a minimum cooling time of 5 years, a maximum enrichment of 4.1 wt.% ^{235}U and a maximum decay heat of 0.875 kW per assembly. In addition, specific requirements, to ensure that the fuel assemblies parameters and the host reactor operating conditions (average and maximum rod power, minimum reactor coolant inlet temperature, maximum reactor coolant outlet temperature, maximum soluble boron content in core, etc.) are within the realm of the U.S. regulatory experience, and are listed in Table 1.2.5 of the package application. Such requirements must be met for the fuel assemblies to be transported.

The maximum weight of the authorized contents is 5,650 kg. There are no moderating materials or neutron absorbers in the contents, nor any other material that would create a chemical, galvanic, or other reaction leading to the release of combustible gases.

1.3 Materials

The Holtite-A neutron shielding material surrounds the layered steel shells and is itself encased in an outer enclosure shell. Metamic neutron absorber panels, qualified as Important to Safety Category A items, are enclosed in stainless steel sheathing that is stitch-welded to the fuel basket cell walls along their entire periphery to provide criticality control. Material and manufacturing control processes are carried out using written procedures to ensure that all critical characteristics are met.

Staff reviewed the materials selected for use in the fabrication of components of the HI-STAR 60 package and found that they meet the service requirements of such components.

1.4 Criticality Safety Index

The Criticality Safety Index (CSI) for the HI-STAR 60 package is zero, as an unlimited number of packages will remain subcritical under the procedures specified in 10 CFR 71.59(a).

1.5 Drawings

The packaging is constructed and assembled in accordance with the following Drawing Nos.:

HI-STAR 60 Cask:	Drawing 5238, Sheets 1-7, Rev. 4
HI-STAR 60 Fuel Basket:	Drawing 5217, Sheets 1-3, Rev. 6
HI-STAR 60 Impact Limiter:	Drawing 5237, Sheets 1-3, Rev. 4

1.6 Evaluation Findings

A general description of the HI-STAR 60 package is presented in Section 1 of the package application, with special attention to design and operating characteristics and principal safety considerations. Drawings for structures, systems, and components important to safety are included in Section 1.3 of the application.

The package application identifies the Holtec International Quality Assurance Program for the Model No. HI-STAR 60 package and the applicable codes and standards for the design, fabrication, assembly, testing, operation, and maintenance of the package.

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. HI-STAR 60 package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL REVIEW

The objective of the structural review is to verify that the structural performance of the package meets the requirements of 10 CFR Part 71, including performance under the tests and conditions for both normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

2.1 Structural Design and Analytical Modeling

The licensing basis for structural performance of the HI-STAR 60 package was predicated on successful analytical modeling rather than experimental tests. Analytical modeling has distinct advantages over physical testing in that an analyst can extract far more information from a structural simulation than can be obtained from a test. However, it is always necessary to ensure that the results obtained from such a model are reliable and consistent with data and the overall structural behavior obtained from a full scale or scale model test.

The staff provided the applicant with several options to demonstrate that the HI-STAR 60 was a safe design. The chosen methodology consisted of a limited analytical benchmark model to demonstrate that the applicant's methodology was sufficient to capture the rigid body dynamics and gross structural performance of a quarter scale version of the HI-STAR 100 package.

The applicant found that this analytical model compared well with test data as well as a Classical Dynamics Method, which was previously approved as part of the HI-STAR 100 package licensing basis. Subsequently the applicant proceeded with the evaluation of the HI-STAR 60 by utilizing a two-step approach which included classical and numerical dynamic modeling to determine a peak deceleration which was then amplified and incorporated in a quasi-static numerical stress and deformation evaluation which was similar to the methodology used for the HI-STAR 100 package.

2.1.1 Description of Structural Design

The HI-STAR 60 containment assembly consists of a cylindrical steel shell with an inside diameter of 1850 mm which is welded to a bottom steel base-plate and a top steel forging. The forging is machined such that it can receive one steel lid with two independent elastomeric seals. Immediately external to the containment boundary liner, intermediate steel shells attached to the cask also provide gamma shielding as well as additional structural rigidity and strength. The two lifting trunnions at the top of the packaging are welded to the outer shell, to the radial rib, to the outermost intermediate shell, and to the inner shell of the containment boundary.

The fuel basket is designed to exhibit physical integrity, i.e., no brittle or ductile fracture, and minimal plastic deformation under the most structurally demanding conditions of transport. Multiple steel plates are welded together to form individual square cells for spent fuel assemblies. Adjacent to the fuel basket are basket supports which are manufactured to fit securely within the remaining space between the external basket walls and the internal surface of the steel containment shell.

Impact energy absorbing compressible spacers, referred to as Fuel Impact Attenuators (FIAs), are designed to mitigate G loads on fuel assemblies due to secondary internal impact. The

FIAs are designed to limit internal gaps between the fuel assembly end fittings and internal surfaces of the package in addition to serving as an energy absorbing medium capable of dissipating up to 40% of the impact energy imparted to the fuel.

Fastener Strain Limiters (FSLs) are collapsible devices designed to limit the axial stress imparted to the impact limiter attachment bolts. These devices are allowed to fail at a specified load which will unload the attachment bolts upon failure while allowing the impact limiter to remain attached to the package.

The impact limiters are configured in such a manner that the collision of the package with the surface (essentially unyielding target) will always occur in the crush material space. The impact limiters are comprised of a steel skeleton that fits over the top forging and bottom base-plate and aluminum honeycomb block material, and are designed to provide energy absorption during impact under all angular drop orientations.

2.1.2 Design Criteria

The structural design criteria are developed to assure that the HI-STAR 60 package has adequate structural strength to meet NCT and HAC requirements. The structural design criteria are designated as those that affect the containment boundary and those that affect other package structures and contribute to the overall structural performance.

The containment boundary is evaluated based on the American Society for Mechanical Engineers (ASME) code requirements for level A and D service and is consistent with Regulatory Guide 7.6. Other miscellaneous structural failure modes such as brittle fracture, fatigue, and buckling are evaluated and found by the applicant to be satisfactory. The staff agrees that brittle fracture and buckling are adequately characterized and evaluated.

The remaining design criteria for structural analysis address the shielding cylinders that are required to remain in place and functional after all NCT and HAC conditions, the fuel basket and fuel basket supports that are required to maintain their physical integrity under all NCT and HAC conditions, and the impact limiters and impact limiter attachments that are required to be designed in such a way that the containment and shielding components do not fail to meet their specified requirements.

2.1.2.1 Loading and Load Combinations

Loads and Load Combinations are evaluated using Regulatory Guide 7.6, and 10 CFR 71 for NCT and HAC Loads as well as ANSI N14.6 for handling loads.

2.1.2.2 Acceptance Criteria

The acceptance criteria established for the HI-STAR 60 package apply to the containment boundary, the fuel basket, the shielding components, and the impact limiters.

Containment boundary:

1. The containment boundary must meet the stress intensity limits of Subsection NB of the ASME Code for design pressure under level A conditions.

2. The containment boundary must meet sealing performance requirements under the Free Drop event as well as satisfy the ASME code limits for Section III, level A and D stress intensity limits for the respective drop heights.

3. Under a penetration event, the containment boundary must not be breached, must remain leak-tight, and level D stress intensity limits must be satisfied away from the point of impact.

4. The containment boundary materials must not be susceptible to brittle fracture.

5. Closure lid seals must remain functional under all events to maintain a sufficient seal such that established leak rates are not exceeded.

Fuel Basket:

1. The fuel basket must meet the level D primary stress intensity limit from Section III, Subsection NG of the ASME code. This assumes an elastic behavior for the basket.

Shielding Components:

1. The shielding should not separate from the cask or suffer extensive damage.

2. Brittle fracture damage resulting in through thickness cracks, thereby causing a loss in shielding function, is not allowed.

Impact Limiters:

1. The impact limiters must perform impact limiting functions such that ASME Section III, Subsection NB stress limits are satisfied for the applicable service condition.

2. Impact limiters must remain permanently attached to the package.

3. Impact limiters must have adequate crush characteristics to prevent bottoming out of the cask body.

4. Decelerations under the 9 m drop must be limited such that the peak flexural plastic strains do not exceed the failure strain for the cladding material.

5. Gasketed joints in the containment boundary must remain fully functional.

2.1.3 Weights and Centers of Gravity

Table 2.1.15 of the application provides the locations of the calculated centers of gravity for an empty package and a loaded package.

2.1.4 Codes and Standards

Table 2.1.16 of the application lists each major structure, system and component of the HI-STAR 60 package along with its applicable code and standard. Table 2.1.17 of the application lists alternatives to the ASME Code where appropriate.

Specifically, the containment boundary is designed to Section III of the ASME code, including Subsection NB and Appendix F. The application also lists and classifies structures, systems, and components as important to safety (ITS) or not important to safety (NITS), based on guidance presented in NUREG/CR 6407. Other materials and components that do not comprise the containment boundary are designated as meeting ASTM specifications with the exception of Holtite (neutron shielding), which is a proprietary specialty material not covered by ASTM specifications.

2.2 Material Properties and Specifications

The applicant states that, unless otherwise specified in the licensing drawings or Tables 2.1.17, 2.1.18, 2.2.8, 2.2.9, or 2.2.10 of the application, components which are important to the safe operation of the package will be constructed from the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PVC) certified materials.

The properties of these materials have been tabulated in the package application and found to be acceptable for their respective applications.

2.2.1 Fuel

The package is limited to carrying 12 assemblies of low-burnup (< 45GWD/MTU) irradiated fuel. The fuel cladding material is Zr-4 per ASTM B 811-1997. The individual rods must be undamaged, with no more than pinhole leaks or hairline cracks, and must be handled by normal means. The fuel assemblies do not contain non-fuel hardware.

The Certificate of Compliance (CoC) specifies the operating conditions of the host reactor, and a maximum end-of-life hoop stress of 90 MPa in the cladding at a temperature of 400^{o}C. It is incumbent upon the owner of the fuel to ensure that both the reactor and fuel characteristics meet the requirements specified in the CoC. The fuel cladding temperature for the HI-STAR 60 is below the 400^{o}C limit recommended by ISG-11, Rev. 3, during normal transport operations.

2.2.2 Containment

The primary containment is provided by high-strength cryogenic steel (SA 203E or SA 350 LF3) to assure protection from fracture under sub-zero transport conditions. Samples of material intended for containment, including weld material, at a minimum shall meet the requirements for low-temperature performance under Section III, Division 1, Subsection NB of the ASME B&PVC. The code alternatives listed in Tables 2.1.17, 2.1.18, and 8.1.2 of the package application do not affect the safety of the containment. The cask containment boundary shall be tested by a combination of methods (including, helium leak test, pressure test, MT, and/or PT, as applicable) to verify that it is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce the effectiveness of the packaging during its licensed service life.

Materials for containment have been found to be satisfactory for their applicable service conditions.

2.2.3 Shielding and Criticality

The gamma shield is a multi-layered steel shell that uses SA 516 Grade 70 steel. The impact resistance of the steel is based on acceptance criteria from Section III, Subsection NF, of the

ASME B&PVC. This approach was previously approved with the HI-STAR 100 design and is acceptable to the staff.

As with the previously licensed HI-STAR 100 package, the shielding against neutron fluence in HI-STAR 60 packaging is provided by Holtite-A, a hydrogen rich, radiation resistant, polymeric material impregnated with boron carbide.

Panels of Metamic aluminum boron carbide metal matrix composite act as the neutron absorber in the transportation cask, preventing an inadvertent criticality in a flooded state. The Metamic panels are completely enclosed in stainless steel sheathing that is stitch welded to the fuel basket cell walls along their entire periphery. Calculations for criticality use a maximum k_{eff} <0.95, all fuel assemblies are assumed to be fresh with no burnup credit, and the moderation assumption is fully flooded under both normal and accident conditions.

2.2.4 Seal

An elastomeric material, such as the Parker O-ring EPDM Rubber E0740-75 which meets the critical characteristics listed in Table 2.2.8, is used as a seal for part of primary containment.

Some of the critical characteristics of the seal include the applicable temperature of retraction, the hardness range at ambient temperature, and the minimum elongation of the material. The seal material has acceptable performance during normal conditions of transport and drying conditions. Based upon test data in Sandia Report SAND94-2207, "Performance Testing of Elastomeric Seal Materials under Low- and High-Temperature Conditions: Final Report," generic high temperature data for EPDM-type materials available from Parker Hannifin Corporation, and a bounding compression set test for E0740-75 at elevated temperatures (also from Parker Hannifin Corporation), the staff considers that the high-temperature characteristics of the elastomeric O-ring material are sufficient to meet the Part 71 requirements under accident conditions. The staff notes that this finding is not based upon an experimental test of the seal on a mock-up cask but is based on the review of the data available for qualifying seal materials. The specified helium permeability of the seal should not impede leakage testing. Table 2.2.8 of the application includes a maximum helium permeation rate that will ensure a measured helium permeation threshold of half of the acceptance criteria per ANSI N 14.5. Future findings on the use of elastomeric seals for spent fuel transportation casks may be modified by written guidance.

The total dose rate to the seals is computed to be 3.66 rad/h or 3.21×10^4 rads if continuously irradiated with the design basis fuel for one year. This is significantly less than the 10^6 rads necessary to cause radiation damage elastomeric materials, and an order of magnitude lower than the minimum radiation resistance of 5×10^5 rads specified for the seal material. Once installed and compressed, the seals should not be disturbed by removal of the closure fasteners; however the seal is replaced following the removal of the closure plate bolting.

The staff finds the critical characteristics and description of the cask sealing surfaces adequate for the application.

2.2.5 Drying

Cavity drying is carried out using a vacuum drying process. After the bulk water has been removed, the pressure inside of the canister is lowered to below 3 torr. When the canister has demonstrated that the internal pressure remains below 3 torr for greater than or equal to 30 minutes, with the vacuum pump turned off, it shall be considered dry.

During the vacuum drying process, the maximum temperature of the fuel cladding remains below the 400^{o}C limit recommended by ISG-11, Rev. 3. Following the fuel drying operations, the cask cavity is backfilled with helium gas. The helium backfill ensures adequate heat transfer and provides an inert atmosphere for fuel cladding integrity during and after transport of the spent nuclear fuel.

2.2.6 Impact Limiters

The critical characteristics of the impact limiter crush material, insulation board, and fastener strain limiters are specified in Table 2.2.9 of the package application.

The aluminum honeycomb material used to fabricate the impact limiters is biaxial. The material is tested by the material supplier to verify that the crush strength is within the limits specified in Table 2.2.9 throughout the temperature range under normal operating conditions. The thermal conductivity of the insulating material remains essentially unchanged in the temperature range of the application.

Table 2.2.9 also contains the required critical characteristics of the fastener strain limiters which protect the impact limiter attachment bolts from yielding during a drop accident.

The applicant specifies that the critical characteristics of the impact limiting materials will be maintained during normal operating temperatures. The staff finds this acceptable.

2.2.7 Fuel Impact Attenuator

Fuel Impact Attenuators (FIAs) made of X2NiCoMo steel are affixed to the underside of the closure lid so that they are aligned to be coaxial with the center of each cell of the basket. Table 2.2.10 of the application documents the critical characteristics of the FIAs. Additional property data for the X2NiCoMo material obtained by staff indicated a significant (15%) decrease of yield strength between room temperature and 300^{0}F.

The staff estimates that the elastic modulus of the material will not decrease by more than 5% between room temperature and 300^{0}F as is characteristic of ferrous materials.

Due to the conservatisms assumed by the applicant in the FEA model, the staff finds that the spring constant of the material will be bounded by the applicant's analysis. This finding is based in part on conservatisms inherent to the HI-STAR 60 design and is therefore limited to the HI-STAR 60 package application only.

2.2.8 Welding and Weld Repair

The following conditions are applicable:

1) Welding, examination, and repair of the containment boundary and weld overlays of cask sealing surfaces will be done in accordance with Section III, Division I, Subsection NB.

2) Welding, examination, and repair of the fuel basket and basket shims shall be done in accordance with Section III, Division I, Subsection NG.

3) Welding, examination, and repair of welds not pertaining to the containment boundary, fuel basket, or basket shims shall be done in accordance with Section III, Division I, Subsection NF, unless otherwise noted in the CoC or on the licensing drawings.

4) Non-ASME Code welds shall be called out on the licensing drawings. Welding, examination, and repair of non-ASME code welds shall be done in accordance with AWS D1.1, D1.3, and D1.6, as applicable.

Exceptions and Additions to the Codes of Construction for Welding include the following:

1) Thermal spray coatings or weld overlays (excluding cask sealing surfaces) which are added as corrosion resistant barriers are not required to follow the mandated inspection procedures listed in the applicable code of construction. Excluding cask sealing surfaces, carbon steel surfaces of the cask that come in contact with the spent fuel pool water will be protected by an anti-corrosion facing achieved by thermal spray or a thin overlay of stainless steel. The facing surface shall be resistant to corrosion in the pool water environment and shall be free of macroscopic pores & hide-out ridges as determined by a qualified visual examination procedure. Discrepant surface areas shall be refaced using suitable mechanical means.

2) Acceptance testing of weld metal pertaining to the containment boundary shall be per ASME Section III, Division 1, Subsection NB, with fracture toughness criteria listed in Table 2.1.12.

3) Acceptance testing of weld metal pertaining to the gamma shield shall be per ASME Section III, Division 1, Subsection NF, with fracture toughness criteria listed in Table 2.1.13.

4) As an alternative to ASME Section III, Division 1, Subsection NB-5120, radiography after Post Weld Heat Treatment (PWHT) will not be performed. All welds (including repairs) will be subject to radiographic examination prior to PWHT of the entire containment boundary. Confirmatory radiographic examinations after PWHT is not necessary because PWHT is not known to introduce weld defects in nickel steels.

The staff finds that the final list of welding codes provided by the applicant is acceptable and that all of the exceptions to the welding codes listed in the application are acceptable. The exception to NB-5120 is based upon prior approval in the HI-STAR 100 application.

2.2.9 Corrosion

There are no significant corrosion issues with the HI-STAR 60 packaging. Internal and external surfaces of the HI-STAR 60 are spray coated with stainless steel or weld-overlayed with stainless steel.

2.3 Fabrication and Examination

Sections 2.3 of the application indicates that the package is fabricated using conventional metal forming and welding techniques and all components are fabricated based on the requirements delineated on the packaging drawings. The applicant states that each component is examined as specified on the packaging drawings. Codes and standards used in packaging fabrication and examination are described in Section 2.1.4 above.

2.3.1 Fabrication

The applicant identifies seven key criteria necessary to ensure that the design can be readily constructed utilizing current and available manufacturing techniques. These criteria include the following:

- Tolerances are achievable
- The design is not overly reliant on stringent tolerances
- Combinations for welding compatibility are specified for dissimilar materials
- Post heat treatment or other residual stress relief is specified
- The manufacturing sequence must allow for unimpeded Non-Destructive Evaluation as well as remedial repairs
- The manufacturing sequence must allow for unimpeded access to relevant post weld machining of critical surfaces
- The manufacturing sequence does not engender unnecessary risk to worker safety

The applicant subsequently provides an overview of a typical fabrication sequence for this package. Staff reviewed this sequence and determined that it is of reasonable detail to fully describe the fabrication sequence.

2.3.2 Examination

The applicant identified eleven key fabrication control and required inspections, which are necessary to ensure that conditions of the CoC can be met. These criteria include the following:

1) Materials of construction must be specified on the licensing drawings. Materials Important to Safety (ITS) will be obtained with appropriate certification and documentation required by Sections II and III of the ASME code where applicable. All materials and components will be inspected for visual and dimensional defects, adherence to specification requirements, and traceability markings where applicable.

2) Welders and weld procedures will be qualified in accordance to Section IX and applicable Section III, and subsections of the ASME Code.

3) Welds will be examined utilizing Section V of the ASME code with acceptance criteria in accordance with Section III of the ASME code. The acceptance criteria for non-destructive examination will be consistent with the code requirements for the component that was fabricated. Post weld inspections will be identified in a weld inspection plan, which details the weld, examination requirements, the examination sequence and the acceptance criteria and is subject to a mandatory review and approval in accordance to the applicants QA program prior to its implementation. Non-destructive examination

inspections will be performed in accordance to written and approved procedures by qualified personnel.

4) The containment boundary will be examined and tested via a helium leak test, pressure test, ultrasonic testing, magnetic particle testing, and/or liquid penetrant testing as applicable. Category A and B welds are subject to volumetric examinations based on Subsection NB of the ASME Code.

5) Grinding and machining operations will be examined by ultrasonic testing to ensure that the metal wall thicknesses are not reduced beyond design limits.

6) Dimensional inspections will occur to confirm compliance with design drawings and to verify fit-up of individual components.

7) Trunnions are designed inspected, and tested in accordance to ANSI N14.6. A visual examination following a test at a maximum of 300% maximum design service loading applied for a minimum of 10 minutes will be performed to verify that no gross deformation or cracking has occurred.

8) Upon completion of hydrostatic or pneumatic pressure tests the internal surfaces of the package will be inspected for cracking or deformation. Subsequent discovery and repair for deformation or cracking will require retesting of the package and the test results shall be documented and incorporated in to the final quality documentation.

9) Each plate or forging used for the containment boundary will be drop weight-tested per Regulatory Guides 7.11 and 7.12 where applicable and ASME Charpy V-notch testing will be performed on these materials. Test results shall be recorded in the final quality document

10) Leak tests will be performed upon completion of the fabrication of the containment boundary.

11) All required tests, inspections, and examinations will be both documented and included in the final quality documentation report(s).

Fabrication materials for all Important to Safety (ITS) components are specified in the Licensing Drawings. Materials and components are receipt-inspected for dimensional acceptability, material conformance to specification requirements and traceability markings, as applicable.

The leakage test instrumentation will have a minimum test sensitivity of one half of the leak test rate. A volumetric examination of each bolt per Subsection NB acceptance standards is performed to ensure the absence of internal voids.

2.4 General Standard for All Packages (10 CFR 71.43)

The applicant demonstrated structural performance of the package by analysis, as explained in Section 2.1 above. The former is used primarily for evaluating the lifting and tie-down devices and the latter for the package dynamic response to NCT and HAC drop tests.

2.4.1 Minimum Package Size

The smallest overall package dimensions exceed the minimum overall dimension of 10 cm (4 inches). Therefore, the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamper-Indicating Features

The impact limiter attachment studs and nuts are fitted with a wire tamper seal. Removal of the impact limiter (hence damaging the tamper seals) is required to access the radioactive contents. This satisfies the tamper-indication requirement of 10 CFR 71.43(b).

2.4.3 Positive Closure

Positive closure is demonstrated by the use of a bolted closure lid weighing several thousand kgs as well as sealed and bolted port covers/caps. Opening of the cask requires specialized tools and a power source; therefore, inadvertent opening is not credible.

The package is adequately analyzed for maximum internal and external differential pressures as well as expected external and internal pressures during NCT and HAC.

Therefore, the containment system cannot be opened unintentionally and the requirements of 10 CFR 71.43(c) are satisfied.

2.5 Lifting and Tie-Down Standards for All Packages (10 CFR 71.45)

2.5.1 Lifting Devices

The applicant evaluates all devices or components related to a lifting operation, including the trunnions, the closure lid lifting holes, and the containment baseplate. The trunnions and lid lifting holes are evaluated by using the requirements of NUREG-0612 for storage which requires a factor of safety on yield strength of 6 and a factory of safety on ultimate strength of 10. Such requirements exceed those imposed by NUREG 1617 and 10 CFR 71.45 (a) which only require a factor of safety against yielding of 3.

All lifting attachments are evaluated for static lifting including a 15% inertial load multiplier and found acceptable based on ASME Level A allowable stresses. Both the trunnions and lid lifting holes have adequate margin for lifting operations.

The applicant also evaluates the effects of the failure of the lifting devices permanently attached to the cask and determines that such a failure would occur away from the containment boundary such that the containment and shielding functions would not be compromised.

Staff reviewed the calculations and justifications presented by the applicant and found them acceptable; therefore, the requirements of 10 CFR 71.45(a)(1) for lifting devices are met.

2.5.2 Tie-Down Devices

The package does not incorporate any structural feature that is used as a tie-down device. Thus, the requirements of 10 CFR 71.45(b)(1) are not applicable.

2.6 Normal Conditions of Transport (10 CFR 71.71)

2.6.1 Heat

Differential Thermal Expansion

The applicant identifies axial fuel growth of the fuel assemblies as a potential mechanism for applying load to the internal surfaces of the package during a cold condition. The applicant concludes that the restraint of thermal expansion is lower in the cold condition and the allowable stresses are larger, therefore the stresses on the fuel assembly and inner package surface are greater in the hot condition. The hot condition evaluation illustrates that the FIA is compressed 8 mm, which imparts a minimal load on the interior of the containment cavity. A buckling evaluation of the fuel assembly is performed and it is demonstrated that the load imparted by the compressed FIA does not cause a global or local buckling event to occur.

Gaps

When considering mitigation of secondary impact due to inherent gaps between the contents, including the fuel assemblies and the basket, the applicant employed Fuel Impact Attenuators to provide a deformable energy absorbing spacer to consume most of the existing gaps as well as absorb a significant portion of the energy imparted to the fuel assembly. Staff agreed with this approach and is also in agreement with the analysis provided by the applicant, in that maximum gaps were considered. Staff however disagrees with much of the following text present in the application on page 2.6-5. “As heuristic reasoning would suggest, increased internal gaps would produce increased impact loads during impact events due to the rebound of the unfixed masses (fuel assemblies and/or basket) from their support surfaces during the package’s free fall. For example, an elastic surface such as the baseplate or lid of the cask supports the weight of the fuel by flexural action when the cask is in a vertical orientation prior to the initiation of the drop event. As soon as the free fall begins, the “flexural spring” would begin to relieve its strain energy, resulting in the presence of a possible gap between the fuel assembly and the baseplate or lid surface at the moment of impact. The extent of separation depends on the flexibility of the support surface and weight of the supported mass. Scoping calculations show that the extent of separation between the fuel assembly and the cask and basket surfaces are rather minute at the instant of impact in any impact event. However, for conservatism, the initial gap is assumed to be at its maximum geometrically feasible value in any drop orientation. This is an evidently counterfactual assumption made to maximize the computed severity of the impact events.”

While staff does not disagree with the physics presented with respect to a vertical freefall event from rest, i.e., an elastic springback effect, staff specifically disagrees with the conclusion drawn by the applicant that the assumption of maximum gap is a counterfactual assumption. With respect to longitudinal gaps, the staff’s position is that when transported, the package is oriented in a horizontal position such that an accident event will tend to load the package in an axial direction without the benefit of having a completely closed geometric gap near the bolted lid due to the manner in which the package is loaded onto the conveyance. If the package is vertical, there exists a maximum geometric gap between the closure lid and the contents. As the package is upended in preparation for transport, this gap will still exist as the direction of gravity does not change to allow for the contents to translate relative to the lid and close the gap. Staff has concluded that only relying on a typical test configuration wherein the package has an initial condition of base-down or bottom-down, which allows gravity to act on the contents and close the existing geometric gap, is unconservative and ignores the as-shipped conditions of the package. The same logical exercise can be performed for the C.G. Over Corner or Slapdown orientations to show that consideration of maximum gaps is conservative and appropriate.

Structural Evaluation (Design Condition, Normal Operating Condition, and 0.3 meter Free Drop)

The applicant considered three load cases for the hot condition of the normal conditions of transport.

The design condition, load case N1, considered only the design pressure, the Normal Operating Condition, load case N2, considered thermal stresses, operating pressure, and the loads imparted due to differential thermal expansion of the FIA, and the 0.3 meter free drop consisted of a side drop evaluated against level A stress intensity limits.

The results from load cases N1 and N2 illustrate that the factors of safety are significantly large (> 13) that these loading conditions do not govern the safety conclusion for NCT. Load case N3, however shows much smaller factors of safety and the results of this case do govern the safety conclusion for NCT.

The bounding value of 40 g’s which was used in the final quasi-static analysis produced stresses resulting in factors of safety exceeding 2.0 for the containment boundary. The factors of safety for the basket exceeded 1.0 indicating that some reserve strength margin exists.

The applicant subsequently concluded that the average peak deceleration of 18 g’s, by ignoring the initial deceleration spike inherent with an increase in impact limiter stiffness due to a lower energy drop event, demonstrated that the results for stress intensity are conservative.

Lid Bolts and Seals

The applicant performed two independent evaluations to determine the state of stress in the lid bolts as well as the degree of compression in the seals due to the NCT loadings. The applicant concluded that no bolt overstresses occurred nor did the closure plate seals unload.

Basket Stability

Large deformation nonlinear finite element analyses performed by the applicant showed no evidence of incipient buckling of the fuel basket plates.

The staff reviewed the statements and conclusions made by the applicant, reviewed calculations presented in supporting documents, reviewed and replicated outputs from submitted finite element calculations, and determined that the structural performance of this package under the hot condition of the NCT satisfies the requirements of 10 CFR 71.71(c)(1).

2.6.2 Cold

The cold condition (-40ºC, -40ºF) was evaluated by the applicant with respect to internal pressure, allowable stresses, bolt stress, and differential thermal expansion. With respect to internal pressure and allowable stresses, the applicant concluded that the internal pressure will decline with decreasing ambient temperature while the material allowable stresses will increase under the same condition. The applicant concluded that decreasing load and increasing the available material strength would result in larger margins of safety than what would be expected for a hot condition.

Based on calculations performed to determine the relative change in stress in the closure bolts, the applicant found that there was an insignificant change and no effect on the effectiveness of the closure lid seals. In addition, the increase of allowable stresses also increases for bolts under the cold condition leaving the margin of safety relatively unchanged. The applicant identified axial fuel growth of the fuel assemblies as a potential mechanism for applying load to the internal surfaces of the package during a cold condition. The applicant concluded that the restraint of thermal expansion is lower in the cold condition and the allowable stresses are larger, therefore the stresses on the fuel assembly and inner package surface are greater in the hot condition.

Staff reviewed the calculations and subsequent conclusions made by the applicant and determined that the structural behavior of this package under the cold condition will satisfy the allowable stresses for the materials and components of construction.

The requirements of 10 CFR 71.71(c)(2) are satisfied.

2.6.3 Reduced External Pressure

Under a reduced external pressure of 25 kPa, the structural behavior is bound by the design internal pressure; therefore, the staff agrees that the requirements of 10 CFR 71.71(c)(3) are satisfied.

2.6.4 Increased External Pressure

Under an increased external pressure of 140 kPa, as required by Regulatory Guide 7.8, the structural behavior of the package is bounded by the requirements of 10 CFR 71.61 which requires that a Type B package be capable of withstanding an external water pressure of 2MPa for a period of one hour without collapse, buckling or inleakage. The staff agrees that the requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5 Vibration and Fatigue

The applicant calculates the natural frequencies of the fuel basket and cask and determines that those frequencies exceed by a significant margin the vibrations frequencies expected during Normal Conditions of Transport. Therefore, the possibility of resonance and subsequent elevated stress conditions are not credible.

The applicant evaluates the package for fatigue and determines, based on the ASME Code (NB-3222.4 (d) of Section III) that a detailed fatigue analysis is not required. Five conditions including Atmospheric to Service Pressure Cycles, Normal Pressure Service Fluctuation, Temperature difference at startup and shutdown, Temperature difference during normal service, and Mechanical Loads are evaluated to demonstrate that the package is exempt from such detailed fatigue calculations. The staff agrees that the exemption criteria of the ASME code are satisfied.

The applicant performs a fatigue analysis on the closure bolts and determines that the main closure bolts should not be torqued and untorqued more than 600 times. A calculation is also performed for the top forging closure bolt threads and a maximum service life of 1 million cycles is found. Staff reviewed the evaluation presented by the applicant for both fatigue and vibration and finds that the results presented support the conclusions made by the applicant relative to the effects of fatigue and vibration.

Based on the analyses presented, 10 CFR 71.71(c)(5) is satisfied for fatigue due to vibration.

2.6.6 Water Spray

Based on Regulatory Guide 7.8, the staff determined that water spray is not significant to the structural design of large packages.

2.6.7 Free Drop

The structural analysis of a 0.3 m free drop is simulated using LS-DYNA with full representation of elastic-plastic response.

The staff reviewed the results and agrees with the applicant’s conclusion that the package is capable of maintaining its structural integrity, and meets the requirements of 10 CFR 71.71(c)(7).

2.6.8 Corner Drop

The corner drop test does not apply since the gross weight of the package exceeds 50 kg, in accordance with 10 CFR71.71(c)(8).

2.6.9 Compression

The compression drop test does not apply since the gross weight of the package exceeds 5000 kg, in accordance with 10 CFR 71.71(c)(9).

2.6.10 Penetration

Based on Regulatory Guide 7.8, the NRC staff has determined that penetration is not significant to the structural design of large casks. The intent of 10 CFR 71.71(c)(10) is satisfied.

2.7 Hypothetical Accident Conditions (10 CFR 71.73)

As indicated in Section 2.1 above, the licensing basis for the HI-STAR 60 was predicated on analysis only and relied on test data solely to verify the analytical modeling capabilities with respect to rigid body dynamics. This preliminary analysis was designated as a benchmark study and consisted of simulation of quarter scale drop tests conducted by the applicant on the HI-STAR 100. The benchmarking of the HI-STAR 100 drop test provides additional assurance that the deceleration and gross deformation results obtained from the HI-STAR 60 evaluation are reasonably accurate and conservative. Since the HI-STAR 100 quarter scale tests were not designed as a benchmark test, they lacked the requisite instrumentation that would have allowed a more robust benchmarking effort to be completed.

As a result the applicant decided to initiate a dual path, independent dynamic analysis that would allow the use of a similar previously approved methodology for the HI-STAR 100 and the state-of-the-art explicit dynamics code, LS-DYNA. This approach had two distinct phases: 1) determination of the rigid body decelerations, and 2) determination of the stresses and sealing integrity.

In phase 1, the previously approved Classical Dynamics Method (CDM) and LS-DYNA would be utilized independently to produce a peak deceleration for each respective drop orientation that would be used as an input for phase 2. Each approach produced results (α_{max}) which were then compared, followed by the selection of the higher value of peak deceleration for each drop orientation, and subsequently amplified to the final design value (β_{max}) used to determine the stresses in the package components in phase 2.

Phase 2 consisted of utilizing the values for β_{max}, generating equivalent loads on various surfaces within a quasi static model in the finite element software, Ansys, and running the quasi-static analysis to obtain relevant stress and deformation results.

These results were then compared with results obtained in LS-DYNA to provide cross validation and ultimately provide additional reasonable assurance that the results obtained were consistent and conservative.

2.7.1 9-meter Free Drop

Staff observed significant discrepancies between the peak decelerations obtained from the Classical Dynamics Method and the LS-DYNA solutions in the initial request for licensing action for the HI-STAR 60. Specifically, the LS-DYNA results were predicting peak G loads that were 20% to 40% higher than the results obtained by the Classical Dynamics Method. The LS-DYNA results showed a significant short duration spike in the side drop deceleration time history that the applicant described as a localized impact of the steel skirt of the impact limiter into the sidewall of the cask due to failure of the Fastener Strain Limiter (FSL).

When an evaluation was made of the forces around the perimeter of the cask due to load transfer from the impact limiter, and those forces were converted into an equivalent G load, it was shown by the applicant that the results compared well (within 3% for the average peak deceleration for all cases except Center of Gravity Over Corner (CGOC) between the Classical Dynamics Method and the LS-DYNA solution.

A sensitivity evaluation was performed by the applicant to determine if the source for the deceleration spike was solely caused by the failure of the FSL, rather than secondary impacts due to internal gaps. Upon removal of the gaps the applicant found that the results changed by only 2.5%. The end drop and a portion of the CGOC discrepancies were governed by internal gaps, which the Classical Dynamics Method could not simulate. Again, when the forces around the perimeter of the cask due to load transfer from the impact limiter were summed, and those forces were converted into an equivalent G load, it was shown by the applicant that the results compared well with the classical dynamics method. The CGOC case was unique in that despite controlling for gaps and/or FSL failure, a significant discrepancy still existed between the Classical Dynamics Method and the LS-DYNA solution.

The applicant reasoned and verified that the most likely cause of this discrepancy was the fact that a portion of the aluminum homeycomb material achieve a localized lockup condition, subsequently inducing a 'hard spot' and causing a spike in the deceleration time history. Taken in total, the peak deceleration results produced by the two independent methodologies demonstrate that each method is a reliable predictor of average peak deceleration. However, LS-DYNA has superior capabilities in capturing more detail overall deceleration time history characteristics. In addition, the applicant decided to use the maximum transient short-term peak deceleration as the starting point for assigning the peak design acceleration for each drop orientation.

Staff reviewed the modeling methodologies, calculation packages, results from the dynamic and quasi-static evaluations, and comparisons with allowable stresses. Staff also verified values obtained from the analytical models by recreating reported data and associated pictorial plots showing deceleration time histories, stress contours, and deformed and undeformed configurations of the package. Finally the staff verified that energy balances for each of the drop scenarios were consistent with good practice in explicit dynamics finite element analysis.

The analytical modeling in aggregate satisfies the requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

This evaluation is not applicable due to the package mass exceeding 500 kg per 10 CFR 71.73(c)(2).

2.7.3 Puncture

The applicant performs two independent analyses of the puncture event at two locations on the HI-STAR 60 package. The first method employed is an analytical approach using the finite element code, LS-DYNA. The second method employed utilizes a strength of materials approach in which the strain energy required to shear a plug of material from a rigid plate.

The applicant concludes that bolted joints maintained their integrity, no through wall penetration occurs in the shielding components or containment boundary, all stress levels remain below ASME level D allowable stresses, and the shield shells do not exhibit through wall cracks.

Staff evaluated the analyses presented by the applicant for both the transient dynamic finite element approach and the strength of materials approach and found that the results presented support the conclusions made by the applicant relative to the puncture drop event. The requirements of 10 CFR 71.73(c)(3) are met.

2.7.4 Thermal

The applicant utilizes temperature information from the evaluated fire events to determine the effects on the structural integrity of the package. The main conclusions drawn by the applicant are that (i) the metal temperature averaged across any section of the containment boundary remains below the maximum allowable temperature for level A conditions for ASME Subsection NB components, (ii) the outer surface of the package directly exposed to the fire does not slump, (iii) internal part interferences do not occur due to differential thermal expansion both during and after the fire event, and (iv) the cask closure lid bolts do not unload leading to a reduction of compression load on the gasketed surfaces.

Staff reviewed the evaluation presented by the applicant and finds the reasoning and conclusions credible; therefore the requirements of 10 CFR 71.73(c)(4) are met.

This satisfies the requirements of 10 CFR 71.73(c)(4).

2.7.5 Immersion - Fissile Material

This requirement is bounded by the deep water immersion requirement, therefore the requirements of 10 CFR 71.73(c)(5) are met.

2.7.6 Immersion - All Packages

This requirement is bounded by the deep water immersion requirement, therefore the requirements of 10 CFR 71.73(c)(6) are met.

2.7.7 Deep Water Immersion Test

The applicant utilizes ASME Code Case N-284 to evaluate the buckling response due to deep immersion in water.

The staff performed independent calculations and agrees that the evaluation performed by the applicant is adequate and the requirements of 10 CFR 71.61 are satisfied with respect to stress limits and stability requirements.

2.8 Fuel Rods

The applicant adopted a single pin analytical model, developed by Pacific Northwest National Lab (PNNL) and used in NUREG-1864, to determine the strain ductility demand that a typical fuel pin may be subject to during a Hypothetical Accident Condition 30-foot drop. The model was constructed with shell elements for the fuel cladding, internal springs to prevent ovalization of the cladding, springs to represent spacer grids, a cask to fuel pin spring, an cask to ground spring representing the impact limiter. In two cases the applicant included a 5mm gap to represent a typical design basis case and a 17 mm gap to simulate the maximum available gap. The applicant also included internal pressure to ensure that all realistic loadings were considered.

The applicant utilized a lower bound strain ductility acceptance criteria of 1.7% which is below the range of values that have been published for High Burnup Fuel. Since this is low to moderate burnup fuel, staff agrees that this value for the strain ductility acceptance criteria is reasonable given the conservatisms already included in the analyses.

The applicant described four distinct models that were developed and are as follows:

Model 1: Benchmark model to simulate PNNL model to verify the analytical methodology is acceptable.

Model 2: Modified Model 1 with parameter changes to reflect the design basis fuel in the HI-STAR 60. The model incorporated zero between the end of the fuel rod and the cask spring.

Model 3: Identical to Model 2, but a 5 mm gap was incorporated between the end of the fuel rod and the cask to rod spring.

Model 4: Identical to Model 2, but a 17 mm gap was incorporated between the end of the fuel rod and the cask to rod spring.

A review of the results illustrated that Model 1 was effective at reproducing results obtained by the NRC and PNNL which give reasonable assurance that the principle structural behavior is being modeled correctly.

Results for the remaining models illustrate that the strain ductility demand for all cases are significantly below the acceptance criteria limit of 1.7% strain with minimum factor of safety greater than three.

2.9 Evaluation Findings

Staff requested the applicant to explain the differences in the peak accelerations predicted by the “Classical Dynamics Method” and the LS-DYNA methodology and to justify the calculated factor of safety for the lid bolts. Staff also requested the applicant to substantiate the input values for the cask-ground spring when evaluating the fuel performance characteristics, to justify the classification of restraint of free thermal expansion as a primary stress, to explain the methodology used to calculate primary stress intensities, to provide sensitivity studies demonstrating that the lower bound crush strength does not affect the G loads on drop orientations other than CGOC, and to justify the use of material rigid body decelerations when determining the peak deceleration for a given drop orientation.

Staff requested a recognized material and manufacturing specification for the Fuel Impact Attenuators in view of their importance in maintaining the integrity of the cladding during a drop accident. Staff also requested additional quantifiable data for the O-ring gasket, a justification for the lack of radiographic examination after post weld heat treatment, and a consistent and comprehensive language regarding requirements for codes of construction for welding.

On the basis of the review of the applicant’s responses and the statements and representations in the application, regardless of counterfactual assumption statements on the consideration of maximum gaps made by the applicant, the staff concludes that the package is adequately described and evaluated to demonstrate that its structural capabilities meet the requirements of 10 CFR Part 71.

3.0 THERMAL REVIEW

The objective of the review is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under both normal conditions and hypothetical accident conditions of transport and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of the Thermal Design

3.1.1 Packaging Design Features

To provide adequate heat removal capability, the applicant designed the HI-STAR 60 package with the following features:

1) Helium backfill gas for heat conduction which also provides an inert atmosphere for the fuel to prevent cladding oxidation and degradation.

2) Minimum heat transfer resistance through the basket by fashioning the basket like a honeycomb structure that is welded completely from the basket base to the top.

3) Top and bottom plenums for transverse flow of the helium gas aiding in convective heat transfer. Buoyancy-induced convective heat transfer is enhanced by low pressure drop flow passage within the open space of cask cavity.

4) Continuous metal heat conduction axially provided by the basket structure.

The staff verified that all methods of heat transfer internal and external to the fuel basket and outer cask are appropriate. The drawings in Section 1.3 of the application along with the material properties in Tables 3.2.1 through 3.2.10 provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package.

3.1.2 Codes and Standards

Appropriate codes and standards are referenced by the applicant throughout the application.

3.1.3 Content Heat Load Specification

The HI-STAR 60 package fuel basket is designed to transport 12 PWR fuel assemblies with a maximum design basis decay heat of 0.875 kW per assembly. The maximum cask decay heat for the design basis fuel is 10.5 kW.

The thermal loads are different for the normal transportation condition and the accident conditions, such as fire. The surface thermal load (combustion heat) is external during a fire accident, while the surface thermal load (insolation) is applied continuously during normal transport conditions. The decay heat load will be the same for normal transport and hypothetical accident conditions.

The staff has reviewed all the heat loads into the package. These heat loads are expected and acceptable.

3.1.4 Summary Table of Temperatures

An additional evaluation of the HI-STAR 60 package was performed using emissivity and absorptivity values of 0.11 and 0.42 respectively for polished stainless steel and is documented in Table E.1 of the thermal calculation package. The results show that the fuel cladding and package components temperatures increase slightly from the data presented in the application package but still remain well below the NCT limits. The summary table of the HI-STAR 60 package component temperatures, presented below, was verified.

Table 3.1: HI-STAR 60 Normal Transport and Hypothetical Fire Accident Maximum Temperatures

Component	NCT	NCT Limit	During Fire	Post-Fire Cooldown	HAC Limit
	Temperature in °F (°C)				
Fuel Cladding	662 (350)	752 (400)	662 (350)	707 (375) @ 22 hr	1058 (570)
Fuel Basket	640 (338)	800 (427)	640 (338)	687 (364) @ 22 hr	950 (510)
Containment Shell	268 (131)	400 (204)	343 (173)	417 (214) @ 1.5 hr	700 (371)
Neutron Shield	241 (116)	300 (149)	1137 (614)	1137 (614) @ 30 min	NA
Enclosure Shell	212 (100)	400 (204)	1236 (669)	1236 (669) @ 30 min	1450 (788)
Lid Seals	171 (77)	248 (120)[1]	178 (81)	307 (153) @ 3 hr	338 (170)[3] 410 (210)[4]
Lid Drain Port and Vent Seals	174 (79)	248 (120)[1]	199 (93)	282 (139) @ 6 hr	338 (170)[3] 410 (210)[4]
Impact Limiters		220 (104)[2]		@ 30 min	NA
Bottom					
- Bulk Average	153 (67)		1069 (576)	1069 (576)	
- Maximum	180 (82)		1366 (741)	1366 (741)	
Top					
- Bulk Average	138 (59)		1092 (589)	1092 (589)	
- Maximum	160 (71)		1366 (741)	1366 (741)	

1. Seal minimum upper operating temperature limit – sustained
2. To ensure adequate crush strength, the bulk temperature of Aluminum honeycomb must not exceed the operating limit
3. Seal minimum upper operating temperature limit – short term, 20 hours or less
4. Seal minimum upper operating temperature limit – short term, 3 hours or less

3.1.5 Summary Tables of Pressures in the Containment System

The summary tables of the containment pressure under the normal condition of transport and hypothetical accident condition (Tables 3.1.3 and 3.1.5 of the application) were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation Sections of the package application.

The maximum normal operating pressure is 8.7 kPa for the containment vessel with 3% of the rods assumed to be breached releasing 100% fill gas and 30% fission gas to the containment cavity in accordance with NUREG-1617 “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel.” The maximum pressure reported for the accident condition is 364.6 kPa for the containment vessel, assuming 100% fuel rods rupture.

The design pressure for the cask cavity is 34.5 kPa in normal conditions and 413.8 kPa for accident conditions.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The applicant provides material thermal properties such as thermal conductivities, densities, and specific heats for all modeled components and uses conservative values for thermal emissivity to model the radiation heat transfer to and away from the transportation package. The staff accepted the approach of specifying the natural convection heat transfer coefficient as a function of the product of the Grashof and Prandtl numbers. This product is a function of the diameter of the overpack, surface-to ambient temperature difference, and air properties. The applicant states that the long-term thermal stability and radiation resistance of Holtite-A has been confirmed through qualification testing, that Holtite-A would not degrade at elevated temperatures, and would not be affected by high neutron fluence and megarad gamma doses. Periodic thermal testing was added to Section 8.2.4 of the application to ensure the thermal conductivity of Holtite-A remains unchanged. The thermal properties used for the analysis of the package are appropriate for the materials specified and for the conditions of the package required by 10 CFR Part 71 during normal and accident conditions.

3.2.2 Technical Specifications of Components

The package materials and components are summarized in Chapters 2 and 3 of the application. These materials shall be maintained below maximum pressure and temperature limits for safe operation. The staff reviewed and accepted these specifications.

3.2.3 Thermal Design Limits of Package Materials and Components

Maximum pressure and temperature limits of package materials and components are provided by the applicant. The staff verified that they have been used consistently in the safety analysis. The applicant states that the cold service temperatures are limited to -40°C (-40^{0}F)and also describes the long-term stability of neutron shield materials (Holtite-A) under normal conditions of transport. The neutron absorber (Metamic) stability in excess of 538°C (1000^0 F) was discussed. Peak cladding temperature limits from ISG-11, Rev. 3, are adopted for the transport evaluation.

The staff reviewed and confirmed that the maximum allowable temperatures for components critical to cask containment, radiation shielding, and criticality were specified. The staff verified that the design basis fuel cladding temperature of 570ºC (1058^{0}F) for accident conditions was observed. This temperature limit is based on the Pacific Northwest National Laboratory (PNNL) report, PNL-4835, which is a methodology accepted by the staff.

3.3 Thermal Evaluation Methods

3.3.1 Evaluation by Analyses

Staff requested the applicant to take credit only for the thermal inertia of the fuel, water, basket, containment shell, lid, and baseplate to calculate the time limits for completion of wet transfer operations. Staff noted that the applicant initially failed to include the basket supports in the basket-to-cavity radial growth calculation and requested new calculations even though the minimum gap would not be exceeded. Staff also noted a discrepancy of 0.21 meters between the overall length of the model compared to the overall length of the package as provided in the engineering drawings and requested the applicant to revise the Fluent models. While reviewing the boundary conditions applied to the Fluent model, staff noted that the absorptivity and emissivity values of polished stainless steel were not conservative and requested that the applicant uses 0.11 for emissivity and 0.42 for absorptivity.

A detailed three dimensional quarter symmetry analytical model for the HI-STAR 60 system is developed using the FLUENT finite volume CFD code as shown in Figure 3.3.2 of the application. The design basis fuel assembly is modeled through the effective thermal conductivity (Keff) approach, i.e., modeling the detailed fuel assembly geometry and gaps between fuel rods as a uniform medium with equivalent thermal conductivity under different temperature conditions. The Keff approach is used throughout the analysis including both normal conditions of transport and hypothetical accident conditions. The ANSYS modeling package is used to obtain the equivalent planar conductivity of the fuel assembly storage cell space as shown in Figures 3.3.1 and 3.3.3. An effective thermal conductivity is also calculated for the fuel basket cell walls consisting of sheathing, helium-gap, Metamic, helium-gap, and cell wall. The primary heat transfer mechanisms within the HI-STAR 60 package are conduction and radiation. The principal paths of heat dissipation from the fuel assemblies to the environment is by metal conduction through the fuel basket, helium gaps, containment boundary, layered intermediate steel shells, steel radial ribs and Holtite A, and finally to the outer neutron shield enclosure shell.

Heat rejection from the cask surfaces to ambient is modeled by including natural convection and thermal radiation heat transfer from the vertical and top cover surfaces. Solar heat is included to comply with the regulation. Due to the large mass and size of the package, insolation at exposed surfaces is averaged over a 24 hour time period. The staff reviewed and accepted the overall analysis approach and assumptions.

For normal conditions of transport, the steady-state analysis produced a maximum cladding temperature of 350°C (662^{0}F), below the limit of 400°C (752^{0}F) for normal conditions of transport. For hypothetical accident conditions, the analysis showed a maximum cladding temperature of 375°C (707^{0}F) which occurred during the post-fire cooldown. This is below the limit of 570^{0}C (1058^0 F) for hypothetical accident conditions. The staff also reviewed all component temperature limits and maximum temperatures from the analysis. All the maximum temperatures comply with the temperature limits for both normal conditions of transport and hypothetical accident conditions.

3.3.2 Evaluation by Tests

Thermal tests of the cask or packaging are not required prior to using the packaging since a sufficient analysis model has been presented and accepted.

3.3.3 Temperatures

See Section 3.1.4.

3.3.4 Pressures

See Section 3.1.5.

3.3.5 Thermal Stresses

The applicant uses high conductivity materials to minimize temperature gradients as well as large fit-up gaps to allow unrestrained thermal expansion of the cask internals during normal transport. Basket-to-cavity radial growth and axial growth are evaluated based on the thermal expansion coefficients in the worst condition. The evaluation results are presented in Table 3.4.2 of the application. For hypothetical accident fire conditions, the gap growth in the radial and axial directions is bounded by the normal conditions of transport.

The staff reviewed and approved the evaluation. The methods presented are standard and the evaluation is under the worst operating conditions. Enough margins are shown in the results to exclude safety concerns.

3.3.6 Confirmatory Analyses

The staff reviewed the FLUENT models used in the thermal analyses. The staff checked the code inputs in the calculation packages and confirmed that the proper material properties and boundary conditions are used.

3.4 Evaluation of Accessible Surface Temperature

Under normal conditions of transport, the package is designed, constructed, and prepared for transport so that the surface temperature is 82°C (180^{0}F) with the design basis heat load and no solar insolation. This temperature is below the 10 CFR 71.43(g) maximum allowable surface temperature limit of 85ºC (185^{0}F) for exclusive use shipments. A personnel barrier, defined as optional hardware, is installed while shipping a loaded package to meet surface temperature and dose rate requirements, as defined in 10 CFR 71.43 and 10 CFR 71.47. The staff accepts this design.

3.5 Thermal Evaluation under Normal Conditions of Transport

The HI-STAR 60 package thermal analysis was performed using the FLUENT CFD code. Inside a fuel cell, the PWR fuel assembly is replaced with an equivalent square section characterized by an effective thermal conductivity in the planar and axial directions. The temperature dependent thermal conductivities are obtained using a two dimensional conduction-radiation ANSYS model. Heat rejection from the cask and impact limiter surfaces during normal conditions of transport dissipates by radiation and natural convection. Based on the product of the Grashof and Prandtl numbers which is a function of the diameter of the overpack, surface-to ambient temperature difference, and air properties, the turbulent condition is satisfied. Therefore, the corresponding turbulent heat transfer coefficient correlations are chosen to model the heat rejection by the cask and impact limiters to ambient. For solar heat model, the applicant used half of the 12-hour insolation specified in 10 CFR Part 71 and applied it over a

24-hour period to account for the dynamic time lag. A solar absorption coefficient of 1.0 is applied to the cask exterior surface.

3.5.1 Heat

Under a 37.8ºC (100^{0}F) ambient temperature, with still air and insolation, the applicant calculates the maximum cladding, fuel basket, containment shell, neutron shield, enclosure shell, seal (lid, vent, and drain), and impact limiter temperatures, as listed in Table 3.1 above. The staff confirms that these maximum temperatures are below the material temperature limits with a sufficient margin; therefore they are acceptable.

3.5.2 Cold

With no decay heat and an ambient temperature of -40°C (-40^{0}F), the entire package approaches the steady-state ambient temperature. Cask components, including the shielding and criticality materials, are not adversely affected by this low temperature.

3.5.3 Maximum Normal Operating Pressure (MNOP)

The MNOP is determined by different sources of gases – initial backfill helium, water vapor, release of fission products, and fuel rod failures. Generation of flammable gas is not considered because it is a non-credible event. Based on 10 CFR 71(c)(1) heat condition (37.8°C (100^{0}F), still air, and insolation) and the design heat load, the MNOP is -1.4 kPa for normal condition and 8.7 kPa for 3% rod rupture. The MNOP calculation shows compliance with the containment design pressure of 34.5 kPa, as reported in Table 2.1.1 of the package application.

3.5.4 Time-to-Boil Limits

Time limits for completion of wet transfer operations are defined in Table 3.3.4 of the application. For an initial pool water temperature of 37.8ºC (100^{0}F), the time limit is about 20 hours (19.97 hours). If this maximum allowable time is insufficient to complete operations, water in the cask cavity must be replaced with an inert gas or water must be circulated through the cask cavity to remove decay heat.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

The HI-STAR 60 package regulatory fire analysis is performed in two stages, a 30-minute engulfing fire at 802°C (1475^{0}F), and a post-fire cooldown. The accident scenario considers the cumulative damage from both the drop test and the penetration test.

The rupture of neutron shield pockets is considered by maximizing the heat input during fire and minimizing the heat rejection in the post-fire analysis. To minimize heat dissipation and the thermal inertia properties of undegraded neutron shield pockets, the thermal conductivity of air is applied to the neutron shield pockets during the post-fire cooldown phase. Also, the lower-bound surface emissivity of stainless steel (0.587) is assumed to minimize cooling.

The analysis simulates the engulfing fire by prescribing a combination of radiation and convection heat transfer on the cask surface. The applicant uses test data from the Sandia Report “Thermal Measurements in a Series of Large Pool Fires” (SAND85-0196TTC-0659UC-71, August 1987) to estimate the convection heat transfer coefficient adopted for the calculation. The ambient temperature during fire is 1475°F and the surface emissivity is 0.9. The staff notes that the applicant uses a surface emissivity of 0.25 for the impact limiters during the fire analysis. Because there is margin for the seals, the staff accepts these results, but future analyses should use the regulatory value for surface emissivity for all package outer surfaces

The staff verified the assumptions, found that the Sandia test data was appropriate for the HI-STAR 60 package because a large pool size ensures an engulfing cask fire, and approved the analysis methods.

3.6.1 Initial Conditions and Fire Conditions

The initial condition of the package, prior to the start of the fire accident, is based on a 38°C (100°F) ambient temperature, still air, and the solar insolation prescribed by 10 CFR 71.71(c)(1).

During the fire, the surface emissivity of the package is assumed to be 0.9. After the 30 minute fire, the 38°C (100°F) ambient temperature is restored and the damaged package is allowed to proceed through a post-fire cooldown phase. In the post-fire cooldown phase, no credit is taken for conduction through the Holtite neutron shield or the impact limiters and the properties of air are substituted instead. The ending condition for the 30-minute analysis is used as initial condition for the post-fire cooldown phase.

The peak temperatures of the key package components after the 30-minute fire with maximum decay heat are shown in Table 3.1 above. The seal temperature is recorded at ½ hour increments. The following table is representative of the data:

Time after fire (hours)	**Temperature in the seal region ºC (⁰F)**
2.0	149 (300)
2.5	152 (306)
3.0	153 (308)
3.5	152 (306)

After 3.5 hours the temperature continues going down. The peak is at 3 hours and based on the data it is not expected that the rate of change in the temperature would be greater than 6 degrees/hour during this time period (considering the rate change during hours 2 to 2.5 as a maximum). Therefore, the post-fire 30 minute time intervals capture the peak seal temperatures.

All of the component temperatures are below the short-term design basis temperatures. Based on these analyses, the staff has reasonable assurance that the cladding integrity will not be compromised during the fire or post-fire cooldown phase.

3.6.2 Maximum Temperatures and Pressures

The maximum temperatures calculated by the applicant are listed in Table 3.1 above. The accident temperatures reflect the peak temperature of a specified component from the time the fire is extinguished to the time the package reaches steady-state conditions. Time-temperature plots of critical components during fire and post-fire cooldown phases are provided in Section 3.4 of the application. The plots are extended to a sufficiently long duration to capture the peak of the temperature curves. For both normal and accident conditions, the inner cavity is assumed to be filled with helium.

For accident conditions, all of the materials remain below their respective materials temperature limits. The seals and the surrounding gaps are modeled as solid steel to maximize heat input to the area. When seated, the seal fills the entire groove so it is in direct contact with the steel and it is exposed to that temperature. The maximum temperature in the region of the seal is captured in time-temperature plots of the seal, as provided in the calculation packages of this application. A fire resistant insulation board is added to the design of the impact limiters to reduce as much as possible the maximum temperature of the seal region.

To maximize fire accumulated thermal energy, the undegraded thermal inertia properties of the neutron shield and of the aluminum honeycomb materials are assumed during the post-fire cooldown phase. Even though the neutron shield fails during the fire accident, the dose rates are shown to remain below the regulatory limit of a total dose of 10 mSv/hr at one meter. Based on these analyses and review, the staff has reasonable assurance that the cladding integrity will not be compromised during the fire or post-fire cooldown.

The applicant calculated the maximum hypothetical fire accident pressure assuming that 100% of the fuel rods fail, all rod fill gas is released, and 30% of the gaseous fission products are released. An understated cavity free volume is used. The maximum hypothetical fire accident pressure is 364.6 kPa, based on the average cavity bulk gas temperature of 256ºC (493^{0}F). The maximum hypothetical fire accident pressure is lower than the limit of 413.8 kPa listed in Table 2.1.1 of the application and therefore is acceptable.

3.6.3 Maximum Thermal Stresses

The HI-STAR 60 package is designed to ensure a low state of thermal stress. This design is ensured by using high conductivity materials to minimize temperature gradient and large fit-up gaps to allow unrestrained thermal expansion of cask internals. The differential thermal expansion analysis of the fuel basket during normal transport bounds the fire condition because of the expansion of the cask body under direct fire heating. The staff reviewed and accepted this argument.

3.7 Evaluation Findings

The staff reviewed the package description, the material properties, component specifications and the methods used in the thermal evaluation, and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71. The staff reviewed the accessible surface temperatures of the package as it will be prepared for shipment and found reasonable assurance that the temperatures satisfy 10 CFR 71.43(g) for packages transported by an exclusive-use vehicle.

The staff reviewed the package preparations for shipment and found reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during normal conditions of transport, consistent with the tests specified in 10 CFR 71.71.

The staff also found reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-time limits during hypothetical accident conditions, consistent with the tests specified in 10 CFR Part 71.

4.0 CONTAINMENT REVIEW

The staff reviewed the HI-STAR 60 package containment design to verify that it has been adequately described and evaluated under normal conditions of transport and hypothetical accident conditions, as required per 10 CFR Part 71.

4.1 Description of the Containment System

The containment system boundary for the HI-STAR 60 packaging consists of the containment shell, the containment baseplate, the containment closure flange, the closure lid, the closure lid bolts, the vent/drain port cover plates, the vent/drain port bolts, the vent/drain port test plugs and their respective mechanical seals. Applicable ASME Code requirements and alternatives are presented in Tables 2.1.16, 2.1.17, 2.1.18 of the application.

The closure lid uses two concentric elastomeric seals to form the closure between the containment closure flange surface and the closure lid. Table 2.2.8 of the application lists the critical characteristics of the O-Ring gasket under normal conditions of transport. The closure lid inner seal is tested for leakage through a small test port in the closure lid. To protect the sealing surfaces against corrosion, a stainless steel weld inlay is provided during manufacturing on the containment closure flange surface.

The containment boundary welds of the cask body include the welds forming the containment shell, the weld connecting the containment shell to the closure flange and the weld connecting the containment baseplate to the containment shell. Full-penetration welds are specified for the plates that form the containment shell. Full-penetration welds are also specified for the containment shell to the containment closure flange and containment baseplate welds. Welding, examination, and repair of the containment boundary, attachments to the containment boundary, and weld overlays of cask sealing surfaces shall be done in accordance with Section III, Division I, Subsection NB.

The closure lid is secured using multiple closure bolts around the perimeter. Torquing of the closure lid bolts compresses the closure lid concentric mechanical seals between the closure lid and the containment closure flange forming the closure lid seal. Bolt torquing patterns and torque values are provided in Table 7.1.1 of the package application. Table 4.1.1 provides a summary of the containment boundary design specifications.

All the components of the containment system are shown in the drawings. The licensing drawings specify the surface finish of the sealing surfaces, include the groove edge entrance roundness radius for the seal and also reference Table 2.2.8 of the application on the critical characteristics of the elastomeric seals. Information regarding the components of the containment system is consistent with that presented in the structural and thermal evaluation sections of the package application.

The staff reviewed the description of the containment system and found it to be adequate. To further ensure safe operation in maintaining containment integrity, the staff requested additional information on the seal material critical characteristics, such as quantitative data from the seal manufacturer, minimum and maximum operating temperatures, compatibility with boric acid and water, radiation tolerances, and specific pre-qualified choices for elastomeric seals. Based on the information provided, the staff finds that a seal material meeting the critical characteristics in Table 2.2.8 is acceptable for maintaining containment.

4.2 Containment of Radioactive Material

The HI-STAR 60 package is not a leak-tight package according to the ANSI N14.5-1997 criterion. The design leakage rate listed in Table 4.1.1 of the package application is determined in accordance with ANSI N14.5–1997 to ensure the requirements of 10 CFR 71.51 are met. The measured leakage rates shall not exceed the values presented in Table 4.1.1.

The applicant follows the methodology in NUREG/CR-6487, *Containment Analysis for Type B packages used to Transport Various Contents* to determine the maximum allowable leakage rate for both normal conditions of transport and hypothetical accident conditions. The applicant performs the analysis based on several assumptions of crud contents, release fractions, fuel rod area, internal pressure, and temperature. These assumptions documented in Section 4.2.4 of the application have been reviewed by staff to ensure conservatism. As to the source terms, following the approach of NUREG/CR-6487, the applicant assumes the releasable source terms are from crud spallation from the fuel rods as well as from the fines, gases, and volatiles which result from cladding breaches. Isotopes which contribute greater than 0.01% to the total curie inventory but have a half-life of less than 10 days are neglected. The staff reviewed the source term assumptions and found them consistent with the approach in NUREG/CR-6487. The source term does not include any damaged fuel consideration because only undamaged design basis fuel is authorized for transport. Therefore, the assumptions are acceptable.

To estimate the Co^{60} inventory in crud, the applicant used the average crud surface activity for typical PWR rods and the average surface area per assembly provided in NUREG/CR-6487. The inventory for the HI-STAR 60 package is based on a design basis 15x15 PWR fuel assembly with a burnup of 45,000 MWD/MTU, 5 years of cooling time and an enrichment of 3.6%. For isotopes other than Co^{60}, the inventory is obtained from the SAS2H and ORIGEN-S module of the SCALE 4.3 code package. Using the equations in NUREG/CR-6487 for crud spallation, release of fines, gases and volatiles from cladding breaches and associated release fractions, the total source term is obtained for both NCT and HAC conditions. An effective A2 value is calculated by weighing the fractions in the inventory.

Based on the allowable release rates defined in 10 CFR 71.51, the maximum allowable leakage rates for NCT and HAC at operating conditions are obtained by dividing the maximum release rates by the available activity concentration. The limiting allowable leakage rate is the NCT leakage rate. The allowable leakage rate at operating condition is then converted to the standard test condition through the formula in NUREG/CR-6487. A leak hole diameter was determined through iteration of the formula to satisfy the leakage rate at transport (operating) conditions. The leak hole diameter is then used to determine the standard test condition allowable leakage rate and actual test condition allowable leakage rate. The allowable leakage rate is determined as $2.7.10^{-4}$ atm.cm^3/s, He. The leak test sensitivity is determined to be half of the maximum allowable leakage rate, i.e., 1.35. 10^{-4} atm.cm^3/s, He. The final results are shown in Table 4.1.1.

In reviewing the available activity concentration, the staff noted the design basis inventory table does not match the inventory table used in the effective A2 calculation in the original application. The applicant revised the application for consistency. This revision did not affect the analysis. The applicant also revised the free volume calculation and corrected an error that includes the enclosure vessel volume. The final free volume reduces the allowable leak rate slightly from 2.8 E-04 to 2.7E-04 atm.cm3/s, He. The leak test sensitivity is half of the maximum allowable leakage rate, therefore, the value is 1.35E-04 atm.cm3/s, He. The final results, shown in Table 4.1.1 of the application, are acceptable to the staff.

To ensure correctness, the staff performed an independent containment analysis. Staff results are consistent with licensee's allowable leakage rate with negligible numerical deviation. Therefore, the staff accepts the analysis based on the compliance of NUREG/CR-6487 and 10 CFR 71.51 requirements.

4.3 Containment Under Normal Conditions of Transport

Under Normal Conditions of Transport, the containment system of the package is designed according to the containment boundary specifications in Table 4.1.1 of the application. Thermal and structural evaluations demonstrate no release of radioactive material under normal conditions of transport (see Section 4.2). The maximum normal operating pressure of the HI-STAR 60 is 8.7 kPa with 3% rods rupture, which is lower than the design internal pressure of 34.5 kPa. The normal conditions of transport maximum temperatures and temperature limits of the containment shell, fuel cladding, fuel basket, neutron shield, enclosure shell, lid seals are shown in Table 3.1.2 of the application. The temperature limits are not exceeded. In Section 2.6.1.4.2 of the application, the LS-DYNA finite element analysis indicates that the closure plate seals do not unload under load combination. Therefore, the seals continue to perform their function under Normal Conditions of Transport. Also no bolt overstress is indicated under any loading event associated with normal conditions of transport.

Based on the evaluation and containment analysis, the staff finds the containment design and evaluation acceptable to the requirements in 10 CFR 71.51(a)(1) and 71.71.

4.4 Containment Under Hypothetical Accident Conditions of Transport

Under HAC, the containment system of the package is designed according to the containment boundary specifications in Table 4.1.1 of the package application. Thermal and structural evaluation demonstrated no release of radioactive material under HAC (Section 4.2). The maximum hypothetical accident pressure of the HI-STAR 60 is 364.6 kPa with 100% rods rupture, which is lower than the design internal pressure of 413.8 kPa under HAC. The HAC maximum temperatures and temperature limits of the containment shell, fuel cladding, fuel basket, neutron shield, enclosure shell, lid seals, lid drain port, and vent seals are shown in Table 3.1.4 of the application. The temperature limits are not exceeded.

Section 2.7 of the application shows that all containment system boundary components are maintained within their code-allowable stress limits and that the seals remain compressed during all hypothetical accident conditions of transport.

Based on the evaluation and containment analysis, the staff finds the containment design and evaluation acceptable to the requirements in 10 CFR 71.51(a)(2) and 71.73.

4.5 Leakage Rate Tests for Type B Packages

Leakage rate testing of the HI-STAR 60 containment system boundary follows the guidance in ANSI N14.5-1997. Table 4.3.1 of the package application provides a summary of the containment system boundary components to be tested and the type of leakage test to be performed for post-fabrication qualification and for final pre-shipment qualification.

Pre-shipment leakage rate testing is performed by the user before each shipment, after the contents are loaded and the containment system is assembled. The specific tests for specific components for maintenance leakage test and periodic leakage test are identical to the leakage tests for pre-shipment. The periodic leakage test acceptance criteria in Table 8.2.1 of the application are in accordance with the criteria for pre-shipment requirements specified in Table 4.3.1 of the application. The acceptance criteria chosen in Table 4.3.1 are such that the sum of the various leakage tests for all components that comprise the containment boundary are less than the leakage rate acceptance criterion specified in Table 4.1.1.

The staff noted that if the loaded HI-STAR 60 package is not immediately transported, the pre-shipment leakage rate test would not need to be re-performed if the containment system has not been opened, unless 12 months have passed (See Section 7.3 of the application). Also, if the containment system is opened, then the seals must be replaced prior to another shipment and the pre-shipment leakage rate test must be performed per Section 8.2.2 of the package application.

The staff finds the leakage test scope and frequency of HI-STAR 60 in fabrication, maintenance, periodic and pre-shipment tests all comply with ANSI N14.5.

4.6 Evaluation Findings

Based on the review of the statements and representations in the application, the staff concludes that the HI-STAR 60 containment design has been adequately described and evaluated and that the package design meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING REVIEW

The purpose of this evaluation is to verify that the HI-STAR 60 package shielding provides adequate protection against direct radiation from its contents and that the package design meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.1 Shielding Design Description

The cask body includes a minimum of 282 mm of steel, 121 mm of Holtite-A, and a 19 mm Holtite enclosure shell for radial shielding. The cask body also includes a 252 mm steel lid and a 205 mm steel base for axial shielding. The fuel basket and the basket supports provide additional gamma shielding. Both impact limiters include two 50 mm steel plates, 60 mm of Holtite-A, and several hundred mm of crushable material. The central steel structures in the impact limiters are credited in the analysis as additional gamma shielding in the axial direction.

Gamma shielding is principally provided by the containment steel shell, the containment baseplate, the containment closure flange, the closure lid, the intermediate shells and the enclosure shell. The shielding against neutron fluence in the HI-STAR 60 package is provided by Holtite-A.

5.2 Radiation Source Specification

The applicant calculates the radiation source term using the SAS2H and ORIGEN-S modules of the SCALE 4.4 code package.

The design basis fuel assembly is a 15x15 PWR fuel assembly described in Tables 1.2.1 through 1.2.5 of the application. The major assumptions in modeling the design basis assembly in the MCNP5 shielding model are the following:

1. The cask is fully loaded with 12 design basis fuel assemblies at a 45,000 MWd/MtU burnup and a 5 year cooling time.
2. The fuel assembly is represented as six homogenized zones: bottom nozzle, gap, active fuel, plenum, gap, and top nozzle.
3. The fuel assembly masses are homogenized within the fuel assembly envelope.

Radiation source terms and isotopic inventories at the design basis are computed with SAS2H of the SCALE 4.4 code package. SAS2H calculates fuel depletion, decay isotopic inventories, and radiation source from fuel and activated materials. The design basis fuel assembly initial enrichment is lowered from a maximum of 4.1 wt.% U^{235} to 3.6% wt.% U^{235} to bound all fuel expected to be discharged from the host reactor.

5.2.1 Gamma Source

The gamma source term is comprised of three distinct sources: the source term from the active fuel region, the source term from Co^{60} activity of steel structural material, and the source term from (n, γ) reactions.

All gammas with energies in the range 0.45 MeV to 3.0 MeV are included in the shielding calculations. Photons with energies below 0.45 MeV are too weak to penetrate the steel of the package and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose. The gamma source term from the active fuel region is calculated with SAS2H and ORIGEN-S.

The primary source of activity in the non-fuel regions of a fuel assembly comes from the activation of Co^{59} to Co^{60} due to the impurities to a larger extent in the steel structural material above and below the fuel. The activity of the Co^{60} is calculated using ORIGEN-S using the in-core fuel region flux at full power and was modified using the appropriate scaling factors listed in Table 5.2.3 of the application.

The third source of photons arise from (n, γ) reactions in the basket and cask materials. The (n, γ) photons are properly accounted for in coupled neutron-photon MCNP5 calculations.

5.2.2 Neutron Source

Neutron source strength increases as enrichment decreases for a constant burnup and decay time. Neutron source terms were computed at an initial enrichment of 3.6 wt.% U^{235} which is a lower bound for the design basis burnup of 45,000 MWD/MTU.

Cm^{244} accounts for approximately 96% of the total number of neutrons produced with slightly over 2% originating from (α,n) reactions within the UO_2 fuel. The remaining 2% derive from spontaneous fission in various Pu and Cm isotopes. The neutrons generated from subcritical multiplication, (n, 2n) or similar reactions are properly accounted for in the MCNP5 calculations.

The fuel assembly data and the operating characteristics are transformed into SAS2H input. SAS2H computes fuel depletion, decay isotopic inventories, and radiation source terms from fuel and irradiated/activated hardware.

5.3 Shielding Model Specification

The shielding analysis of the HI-STAR 60 package is performed with MCNP5 using the continuous energy ENDF/B-VI neutron and photon cross section libraries. For a normal condition, the HI-STAR 60 package model includes the neutron shield and impact limiters while for the hypothetical accident condition, the neutron shield is replaced with void and the impact limiters are removed, except for the bottom of the cask which includes one impact limiter support plate.

Design basis fuel assemblies are modeled in each of the 12 basket locations with the active fuel region modeled as a homogenous zone. The bottom nozzle, plenum and top nozzle regions are also modeled as homogeneous regions of steel. Each of the source terms, fuel neutron, fuel photon, and hardware Co^{60} are calculated individually. The Co^{60} source in the hardware is assumed to be uniformly distributed.

The applicant selected 12 axial nodes for determining the axial fuel source distribution. This smaller number of nodes than what is usually used for other fuel types still gives comparable node heights. This is due to the fact that the active length of the fuel in the HI-STAR 60 package is only 2900 mm, compared to 3500 mm or more for many other fuel assemblies. The nodal height for each of the 12 nodes is therefore about 237 mm, while for an active length of 3500 mm and 16 axial nodes, the height is 225 mm. As a validation of this approach, more refined computations with 24 axial nodes have been performed, where the peak burnup is further increased by 2%. The calculations show that the maximum surface dose rate increases by 3% over the 12 node case, while the 1 m and 2 m dose rates are the same within the statistical uncertainty.

In order to account for the non-linear relationship between the neutron source strength and the assembly axial fuel burnup, the neutron source strength in 12 axial nodes is determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The increase in neutron source with a 4.2 power of the burnup was previously reported, justified and approved in the HI-STAR 100 and HI-STORM 100 package applications. This calculation results in 15% increase in the total neutron source strength. Therefore, the overall increase in neutron source strength due to the shape of the burnup is accounted for by multiplying the total cask neutron source by 1.15. This shape factor of 1.15 is the average of the multiplication factors of all nodes in an assembly, weighed by the nodal height if necessary, and represents the overall increase in source term compared with an assembly where each node would have the assembly average burnup. The 1.15 factor needs to be applied to the dose rates since MCNP re-normalizes the axial neutron source shape.

The NCT shielding configuration for the HI-STAR 60 MCNP5 model includes the radial steel shells, steel base, steel top lid, radial neutron shield with radial ribs, the impact limiters, and the lifting and rotation trunnions.

The HAC shielding configuration for the MCNP5 model of the HI-STAR 60 package includes the radial steel shells, steel base, steel top lid, no Holtite in the radial neutron shield, no Holtite in the trunnions, and no impact limiters, except that the lower support plate from the bottom impact limiter is assumed attached. Thus, the shielding analysis is conservatively performed by assuming that the entire volume of neutron shield is replaced by void.

Since all the materials used in the HI-STAR 60 package remain below their design temperatures during normal conditions, the shielding analysis does not include changes in the material density or composition as a result of temperature changes. In particular, the applicant does not include the changes in composition of Metamic and Holtite-A since the ^{10}B depletion in these materials over a period of 50 years is negligible.

5.4 Shielding Evaluation

The shielding analysis uses the MCNP5 code. The HI-STAR 60 package is modeled in full three-dimensional detail, the active fuel region is modeled as a homogenized mass and the upper and bottom end fitting regions of the design assembly are modeled as a uniform density steel. The cask body, neutron shield, radial steel ribs and impact limiters are modeled according to the engineering design drawings.

In MCNP5, surface flux type tallies, requested as input for the cask surface, personnel barrier, at 1 m from the surface and at 2 m from the edge of the rail car, are segmented into bands, axially along the side surfaces and radially at the top and bottom surfaces. Dose rates are computed from surface tallies, multiplied by the appropriate flux-to-dose factors and multiplied by the total source strength for each radiation source term. Maximum doses are determined at key locations along the side of the cask, i.e., radial mid-plane, top nozzle, bottom nozzle, and along the centerline of the impact limiters.

Table 5.1 lists the maximum dose rates at various locations under NCT and HAC.

Table 5.1

Maximum Dose Rates for the HI-STAR 60 Package Under Normal and Accidental Conditions of Transport for design basis fuel (45,000MWD/MTU and 5 year cooling time)

Condition	Dose Rate Location	Maximum Dose Rate (mSv/hr)	10 CFR 71 Limit (mSv/hr)	Deviation (%)
NCT	Surface of HI-STAR 60 Cask	0.374	10	1.6
	2 m from the edge of the railcar	0.037	0.1	2.1
	1 meter from the Surface of Cask	0.095	0.1	0.9
HCA	1 meter from the Surface of Cask	2.249	10	0.7
	1 meter from the Top of the Cask	1.232	10	2.7
	1 meter from the Bottom of the Cask	2.030	10	1.8

5.5 Evaluation Findings

The staff reviewed the description of the package design features related to shielding and the source terms for the design basis fuel and found them acceptable. The methods used are consistent with accepted industry practices and standards. The staff reviewed the maximum dose rates for normal conditions of transport and hypothetical accident conditions and determined that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51.

Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance, and that the package design and contents satisfy the shielding and dose limits in 10 CFR Part 71.

6.0 CRITICALITY REVIEW

The staff reviewed the criticality evaluation of the HI-STAR 60 package application using the guidance in NUREG-1617.

6.1 Description of Criticality Design

The HI-STAR 60 package criticality safety relies on the geometry of the fuel basket, the incorporation of permanently fixed neutron absorbing material in the basket structure and fuel enrichment limits for criticality control.

6.1.1 Packaging Design Features

The staff reviewed Section 1 of the application and verified that the information important for criticality safety was both specified and consistent with the drawings, HI-STAR 60 Cask Drawing No. 5238, Rev. 4, and HI-STAR 60 Cask Drawing No. 5217, Rev. 6.

The applicant performed sensitivity studies to determine the most conservative dimensions for parameters such as cell ID, cell wall thickness, flux trap width, and neutron absorber thickness. The staff verified that the applicant used for the criticality analyses the most conservative fuel and package dimensions considering the measurement tolerances.

With respect to the criticality evaluation, the packaging is described in sufficient detail to provide an adequate basis for its evaluation, and the description includes types and dimensions of materials of construction and materials specifically used as non-fissile neutron absorbers or moderators. Therefore, the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5).

The applicant identified codes, standards, and regulations applicable to the criticality design, as required in 10 CFR 71.31(c). Section 6.5.1 of the application specifies that the analyses were performed using an infinite number of packages. Therefore there is no limit (with respect to criticality) to the number of packages that can be transported in a single shipment. The staff finds that, by specifying the allowable number of packages that may be transported in a single shipment, the applicant meets the requirements of 10 CFR 71.35(b).

6.1.2 Summary Table of Criticality Evaluations

The staff reviewed Table 6.1.1 of the criticality evaluations. The staff found that the applicant performed criticality evaluations for a single package in damaged and undamaged conditions, optimally moderated with water and most reactive form of the fissile material and that had close full reflection[1] of the containment system on all sides using 300cm of water. In addition to these analyses, the applicant also performed an analysis with full internal moderation and no external moderation, as this was found to be the most reactive configuration. The staff finds that this meets the requirements of 10 CFR 71.55(b), (d) and (e).

Table 6.1.1 of the application also shows that the applicant performed criticality evaluations for an infinite array of packages in the undamaged and damaged conditions. The staff finds that this meets the requirements of 10 CFR 71.59(a)(1) and 10 CFR 71.50(a)(2).

The staff verified that Table 6.1.1 includes the maximum value of k_{eff}. The applicant states that the value includes two standard deviations that range between 0.0002 and 0.0006 and the uncertainty and bias in the code which is 0.0031. The applicant also shows that the limiting condition for the HI-STAR 60 package is the single package, un-reflected, with full internal moderation. This analysis gives a maximum k_{eff} of 0.9212. The package meets the subcriticality criterion.

6.1.3 Criticality Safety Index

[1] "Close reflection by water" is defined in 10 CFR 71.4 as immediate contact by water of sufficient thickness for maximum reflection of neutrons.

In Section 6.5.2 of the application, the applicant calculates the Criticality Safety Index (CSI) to have a value of 0. This is based on the analysis performed that uses a value of N to be infinity (i.e., an infinite array of packages). The staff finds that the CSI was appropriately determined per 10 CFR 71.59(b). The staff finds that the applicant meets 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Fissile Material Contents

The only contents authorized for transportation in the HI-STAR 60 package are 12 undamaged 15x15 PWR spent fuel assemblies. Fuel assemblies' physical characteristics are presented in Table 1.2.1 of the application. Table 1.2.2 of the application lists the maximum enrichment, maximum decay heat, maximum burnup and minimum cooling time of the fuel assemblies. The maximum enrichment is 4.1%. The applicant does not request credit for the burnup of the fuel.

The applicant performed sensitivity studies to determine the most conservative dimensions for the fuel density and the water temperature in the package.

The staff finds that the applicant has described the contents in sufficient detail to provide an adequate basis for this evaluation. The staff finds that the applicant has defined adequately the type, maximum quantity, and chemical and physical form of the fissile material. The staff finds that this meets the requirements of 10 CFR 71.31(a)(1), 10 CFR 71.33(b)(1), 10 CFR 71.33(b)(2), and 10 CFR 71.33(b)(3).

6.3 General Considerations for Criticality Evaluations

The applicant uses full three dimensional calculations and conservatively neglects the absorption in the neutron shielding material. The calculational model defines the fuel rods and cladding, the guide tubes, and the neutron absorber panels on the stainless steel walls of the basket cells. The manufacturing tolerances of the basket are also included and the Monte Carlo N-Particle (MCNP4a) code determines the manufacturing tolerances that produce the most adverse effect on the criticality. Based on these calculations, conservative dimensional assumptions (cell ID, cell wall thickness, flux trap width, and neutron absorber thickness) are then determined for the basket design. These values are shown in Figure 6.3.1 and Table 6.3.2 of the package application.

The composition of the major components of the HI-STAR 60 package is listed in Table 6.3.4 of the package application. There is no difference in the material properties between normal and accident compositions.

The applicant uses the MCNP4a code for the criticality analysis and the CASMO-4 code to establish the direction of reactivity in the determination of some incremental reactivity effects.

The applicant analyzes several conditions to confirm or identify the most reactive configurations and demonstrate compliance with the individual requirements of 10 CFR 71.55 and 10 CFR 71.59. Such conditions include different package spacing in an array, partial and preferential flooding of the package, and flooding of the pellet to clad gap of the fuel rods.

6.3.1 Model Configuration

The staff reviewed Sections 2 and 3 of the package application to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents. The applicant's conclusion is that the hypothetical accident conditions have no adverse effect on the geometric form of the package contents important to criticality safety and the only damage is to the Holtite neutron shield during a hypothetical fire accident. Since the applicant does not model this shield in the criticality analyses, the staff finds that the criticality models are consistent with the effects of the hypothetical accident conditions.

The staff examined the sketches of the model used for the criticality calculations and verified that the dimensions and materials are consistent with those in the drawings of the actual package. The staff verified that the application considers deviations from nominal design configurations by analyzing the most reactive configuration possible. This is discussed in Section 6.1.1 above. As discussed in Section 6.1.1, the applicant's model differs slightly to ensure the package has been modeled conservatively.

6.3.2 Material Properties

The staff verified that the appropriate atom densities are provided for all materials used in the models of the packaging and contents. This information is included in Table 6.3.4 of the application. There are no materials in the casks that need to be adjusted to be consistent with accident conditions, i.e., there are no materials used in the model that change form, such as a potential melting of the neutron shield or neutron absorbers, that are assumed in the calculations and needed to maintain sub-criticality.

The applicant uses Metamic for the neutron absorbing material. The criticality analysis assumes the Metamic has a ^{10}B areal density of 0.0245 g/cm^3, i.e., 90% of the minimum neutron absorber content. NUREG/CR-5661 states that: "a percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented." In Section 8.1.5.4 of the package application, information about the tests used to verify the presence and uniformity of the neutron absorber is included. Therefore, the staff finds that assuming 90% of the minimum neutron absorber content is acceptable.

The applicant does not request credit for burnable poisons in the fuel.

During the review of the HI-STAR 100 package, the applicant provided information demonstrating that the neutron flux from the irradiated fuel results in a negligible depletion of the ^{10}B content in the B_4C for 20 years. This was approved by the staff in the *Safety Evaluation Report Holtec HI-STAR 100 Cask System,* September 1999, ADAMS Accession No. ML060730443. The staff verified that there were no other materials assumed in the criticality safety analysis that could degrade during the service life of the packaging.

6.3.3 Computer Codes and Cross Section Libraries

The applicant uses the Monte Carlo N-Particle Code, Version 4a (MCNP4a) which is appropriate for criticality evaluations per the guidance in NUREG-1617. In addition the applicant uses the CASMO-4 code but only to determine some incremental reactivity effects.

The applicant uses the ENDF/B-V nuclear data set. The staff recognizes this as a standard in criticality safety and finds its use acceptable for the criticality analyses for the HI-STAR 60 package. The staff verified that the applicant provided other key input data for the criticality calculations. These are summarized below in Table 6- 1.

Table 6- 1
Criticality Key Input Data

Parameter	Value
Number of Neutrons per Generation	10,000
Number of Generations	250 (including skipped cycles)
Number of Cycles Skipped Before Accumulating Data	50

The staff verified that the applicant provided representative input and output files. The staff verified that the multiplication factors from the output files agree with those reported in the evaluation. The staff also verified that the information regarding the model configuration, material properties and cross sections was properly represented in the input files.

6.3.4 Demonstration of Maximum Reactivity

The applicant performs analyses to determine the optimum combination of internal and interspersed moderation for the allowable contents (15x15 PWR fuel rod lattice). This is shown in Table 6.3.6 of the package application. The applicant determines that the reactivity increases with increasing internal moderation and is not dependent on the external moderation. The applicant shows that the most reactive configuration is with 100% water density for internal moderation and that the single cask is more reactive than the array of casks. The applicant also addresses the effects of partial flooding. The applicant shows in Table 6.3.11 of the application that the fully flooded condition bounds all of the partial flooding cases. Due to the design of the basket, preferential flooding is not possible in the HI-STAR 60 package.

In addition to the sensitivity cases mentioned in Section 6.1.1 above, the applicant performs the following sensitivity studies to ensure that the most reactive configuration is modeled:

- Clad gap flooding – the applicant determined that assuming flooding of the pellet-to-clad regions is the most reactive configuration (Table 6.3.12 of the application).

- Eccentric positioning of assemblies in fuel storage cells – the applicant determined that the cell-centered configuration is the most reactive configuration (Table 6.3.5 of the application).

- Partial Loading – the applicant shows that the fully loaded condition bounds the partially loaded condition (Table 6.3.13 of the application).

In addition, the applicant discusses the reactivity effects due to manufacturing damage and tolerances of the neutron absorber panels. The applicant states that the effects are negligible and bounded by other conservative assumptions in the analyses. The staff finds this acceptable and finds that the applicant's analysis demonstrated that the maximum reactivity was identified per the requirements of 10 CFR 71.55(b).

6.3.5 Confirmatory Analysis

The staff performed independent calculations to confirm the applicant's results and verify that the most reactive conditions had been correctly identified. The staff performed calculations with the CSAS26 criticality sequence of the SCALE 5.1 suite of codes. SCALE 5.1 was developed by Oak Ridge National Laboratory for use in criticality and shielding analyses. The CSAS26 sequence is a criticality sequence that uses KENO-VI geometry and multi-group cross sections. Staff used the 238-group cross section library derived from ENDF-VI data.

Staff independently constructed its model based on design information and data specified in Table 1.2.1 of the application, the engineering drawings and the information from the Chapter 6 of the application. The results of the staff's evaluation are summarized in Table 6-2.

Table 6-2

Independent Staff Calculations for the HI-STAR 60 package

Case	Staff's keff from SCALE	Standard Deviation	Applicant's keff from Package Application*
Infinite Array of Damaged Packages, 100% internal and 100% external moderation	0.9110	0.0019	0.9197
Infinite Array of Undamaged Packages, 0% internal and 0% external moderation	0.32303	0.00051	0.3373
Single Package, Damaged, 100% internal and 0% external moderation	0.9129	0.0018	0.9212
Single Package, Damaged, 100% internal and 100% external moderation	0.9095	0.0019	0.9197

*applicant's values include 2 x standard deviation and combined code bias and bias uncertainty

As shown in Table 6-2, the evaluations from the staff and the applicant are in general agreement.

6.4 Single Package Evaluation

The applicant performs calculations for full internal and external water moderation for both a single containment and a single package under normal conditions of transport and for a single package, internally flooded, with full external water moderation under accident conditions of transport to address the requirements of 10 CFR 71.55(b) and (d). All of the applicant's results are below the acceptance level of 0.95 for k_{eff}.

6.4.1 Configuration

The staff verified that the applicant's evaluation demonstrates that a single package is subcritical under both normal conditions of transport and hypothetical accident conditions. As stated in Sections 6.3.1 and 6.3.2 above, damage to the cask is limited to the neutron shield during accident conditions and the applicant conservatively neglects the presence of the shield in all calculations. The applicant floods the inside of the cask with water when performing calculations for the damaged condition.

The applicant models the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents as discussed in Sections 6.1.1 and 6.3.4 above.

6.4.2 Results

For the single package evaluation, the most reactive configuration calculated by the applicant is the 100% internal moderation with 0% external moderation. The applicant's analyses give a k_{eff} of 0.9212.

Since k_{eff} is less than 0.95 under the tests specified in 10 CFR 71.71, the staff verifies that this meets the requirements of 10 CFR 71.55(d)(1) which requires that the contents be subcritical.

The staff verifies that the geometric form of the package contents would not be substantially altered as specified in 10 CFR 71.55(d)(2). The staff finds that the applicant meets the intent of 10 CFR 71.55(d)(2).

The staff did not verify that there would be no leakage of water into the containment system per 10 CFR 71.55(d)(3) because the applicant assumes full in-leakage of water at its most reactive extent for the single package evaluation for normal and hypothetical accident conditions. The staff finds that the applicant meets the requirements of 10 CFR 71.55(d)(3).

Under the tests specified in 10 CFR 71.71, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention. The staff verified that there is no reduction in the effectiveness of the packaging, including (1) the total volume of the packaging will not be reduced on which the criticality safety is assessed, (2) the effective spacing between the fissile contents and the outer surface of the packaging is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm cube. The staff finds that the applicant meets the requirements in 10 CFR 71.55(d)(4).

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

The applicant performs calculations for an infinite array of undamaged packages and finds a maximum value of k_{eff} of 0.3373. In these calculations, the packages were internally and externally dry. All the results are below the acceptance level of 0.95 for k_{eff}

6.5.1 Configuration

The applicant specifies a CSI of 0.0 and performs calculations using an infinite array of packages for normal conditions of transport. The applicant models the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents, as discussed in Sections 6.1.1, 6.2, and 6.3.4 above. The applicant models a case with no water inleakage and no interstitial water moderation.

6.5.2 Results

The k_{eff} as calculated by the applicant for the infinite array of packages under normal conditions is 0.3373. The staff finds that the applicant meets the requirements of 10 CFR 71.59(a)(1) by demonstrating that an array of at least 5N packages (infinite array in this case) with nothing between the packages is subcritical.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The applicant performs calculations for the most reactive credible configuration of touching internally flooded packages with full external water reflection. The maximum k_{eff} is 0.9197, below the acceptance level of 0.95.

6.6.1 Configuration

The applicant specified a CSI of 0.0 and performed calculations using an infinite array of packages for the hypothetical accident conditions of transport.

The applicant models the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents, as discussed in Sections 6.1.1, 6.2, and 6.3.4 above. The applicant's case models full inleakage of water and full interstitial moderation. As discussed in Section 6.3.4 above, the applicant finds that the amount of interstitial moderation has no effect on the reactivity of the package.

6.6.2 Results

The k_{eff} as calculated by the applicant for the infinite array of packages under hypothetical accident conditions is 0.9197. The staff finds that the applicant meets the requirements of 10 CFR 71.59(a)(2) by demonstrating that an array of at least 2N packages (infinite array in this case) with nothing between the packages is subcritical.

6.7 Benchmark Evaluations

The applicant is using the MCNP4a code with the cross section data from the ENDF/B-V library. This code is widely used in industry applications for criticality calculations and has therefore been extensively benchmarked against critical experiments

6.7.1 Experiments and Applicability

The applicant performs benchmark calculations with the same computer codes and cross section data that were used to calculate the k_{eff} values for the package. The staff verified that the design parameters for the HI-STAR 60 package, i.e., enrichment, type of fissile material, rod pitch and diameter, ^{10}B loading, and energy of the average lethargy causing fission (EALF), are within the benchmark experiments cited by the applicant.

The staff found that the experiments included in the bias determination in Table 6.A.1 of the application are applicable to the HI-STAR 60 package. Regarding some experiments that include mixed-oxide (MOX) fuel, the applicant re-calculated the bias without these experiments because the bias is typically positive for these configurations and MOX fuel is not included in the proposed contents of the HI-STAR 60 package. The staff finds the new bias, as described in Appendix B to the criticality evaluation, is acceptable for the HI-STAR 60 package.

6.7.2 Bias Determination

The applicant determines that there are no trends in the k_{eff} results for the following effects: (1) enrichment, (2) ^{10}B loading, (3) reflector material and spacings, and (4) fuel pellet diameter and pitch.

Since there are no trends in the data for any of these individual effects, the applicant chooses to represent all of the data by the “energy of the average lethargy causing fission” (EALF) and performs linear regression of all of the data to determine a total calculational bias. This approach has been reviewed and approved previously by the staff and found acceptable for the HI-STORM 100 and the HI-STAR 100 packages. The applicant shows a bias of 0.0031 and states that this bias added to all of the calculated k_{eff} values (along with 2 standard deviations) in Table 6.1.1 of the application (summary table of the criticality results).

6.8 Burnup Credit

The applicant does not request credit for burnup.

6.9 Evaluation Findings

The criticality evaluation performed by the applicant is consistent with the appropriate codes and standards for criticality safety analyses, and NRC guidance. A summary of the criticality evaluation results is reported in Table 6.1.1 of the package application. The results show that the package meets the requirements of 10 CFR Part 71 for criticality safety and that all values of the neutron multiplication factor (k_{eff}), after being adjusted for uncertainty and biases, fall below the acceptance limit of 0.95 given in NUREG 1617. The maximum k_{eff} reported is 0.9212.

The applicant’s analyses consider an infinite array of HI-STAR 60 packages under both normal conditions of transport and hypothetical accident conditions. Results are below the regulatory limit, i.e., N is infinite and the Criticality Safety Index (CSI) is zero. The applicant does not take credit for burn-up.

Based on the review of the statements and representations in the application and confirmatory analyses performed by the staff, the staff finds reasonable assurance that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the criticality safety requirements of 10 CFR Part 71.

7.0 PACKAGE OPERATIONS

Chapter 7.0 of the application provides a description of package operations, including package loading and unloading operations, and preparation of an empty package for shipment.

7.1 Package Loading

Package loading operations include package preparation activities, fuel assembly loading, package closure and preparation for transport.

Package preparation activities include (i) visual inspections to verify that there are no indications of impaired physical conditions on either the cask surface itself, the containment closure flange seal surfaces, the closure lid bolts, the cask neutron absorber panel sheathing, etc.., (ii) the performance of a radiological survey, (iii) the removal of the impact limiters, if previously attached, and of any road dirt or debris or any foreign material, (iv) the upending of the cask and, (v) the removal of the cask lid.

Prior to fuel loading, the user identifies the fuel to be loaded, verifies that it meets the conditions of the Certificate of Compliance, and performs a visual verification of the fuel assembly Identification Number. New seals are installed in the groove in the lid prior to the lid installation in the spent fuel pool. The lid is visually inspected to confirm it is properly seated. The user performs a site-specific Time-To-Boil evaluation to determine a time limitation to ensure that water boiling will not occur in the cask prior to the beginning of the draining operations. The maximum allowable time for completion of fuel loading operations is 16.40 hours, assuming a pool water temperature of 48.9ºC (120^{0}F), or 20.86 hours for a pool water temperature of 35ºC (95^0 F). If operational malfunctions occur and if it appears that the Time-To-Boil limit will be exceeded prior to draining operations, the user implements specific procedures to either replace the water in the cask cavity with an inert gas or circulate the water through the cask cavity to reset the Time-To-Boil clock.

The seal surface is protected from damage during fuel loading by a surface protector. The new seal, installed in the lid groove prior to installation of the cask lid in the spent fuel pool, is held in place by the groove geometry: no lubricant is used on the main seal.

The closure lid bolts are torqued after the package is removed from the pool and the vent line opened but prior to the water being drained from the cask cavity. The HI-STAR 60 package torque requirements are specified in Table 7.1.1 of the application. The vacuum drying system is connected to the cask and used to remove moisture from the internal cavity. The cask drying operation is critical to the spent fuel cladding integrity. The dryness criteria are specified in Section 7.1.2.1 of the application.

7.2 Package Unloading

Package unloading operations include the receipt of the package from the carrier, the cooling of the fuel assemblies, the flooding of the cask internal cavity, the removal of the lids and bolts, the unloading of the fuel assemblies, and the release of the package for future transport operations.

Upon receipt from the carrier, the package is visually inspected to verify there are no indications of impaired physical conditions; a radiological survey is performed and the impact limiters are removed; the cask is then upended and moved to a designated preparation and unloading area of the fuel building. The closure lid port cover and drain port cover are removed to access the vent and drain ports and gas sampling is performed to assess the condition of the fuel assembly cladding. Before cask unloading under water, the cask is cooled if necessary to reduce the internal temperature to allow water flooding without thermally shocking the fuel assemblies or over-pressurizing the cask. The closure lid bolts are removed and the cask is placed in the unloading area or in the pool. After fuel unloading, the cask is raised to the top of the pool and the dose rates at the top of the cask are measured in accordance with plant radiological procedures. The cask is then returned to the designated preparation area where any cask cavity water is pumped back into the pool or any approved system, and the cask is decontaminated.

7.3 Preparation for Shipment

Preparation for shipment includes (i) the removal of the seal surface protector, (ii) the contamination survey, (iii) the installation of the closure lid followed by appropriate torque requirements for the bolts, (iv) the installation of the closure lid port covers, if necessary, (v) the verification that the cask neutron shield pressure relief devices are installed, intact and not covered by any covering, (vi) the installation of the impact limiters, (vii) the installation of a security seal and its appropriate recording on the shipping documentation, (viii) the installation of a personnel barrier, if desired, and (ix) package marking, labeling, and vehicle placarding.

7.4 Evaluation Findings

To further ensure safe operation in maintaining containment integrity, the staff requested additional information on the detailed sealing installation and bolt torque procedures. The applicant revised the loading procedure to add adequate details about underwater seal installation procedure. The sealing surface will be inspected in the installation procedure to ensure that the seal surface is clear of potential solid contamination and free from gross damage that might affect the seal performance. The licensee also clarified that the bolt torque procedure would be performed above water.

The staff reviewed the Operating Procedures in Chapter 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation. On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the proposed design basis fuel in accordance with 10 CFR Part 71.

Further, the Certificate of Compliance is conditioned such that the package must be prepared for shipment and operated in accordance with the Operating Procedures specified in Chapter 7 of the application.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application identifies the inspections, acceptance tests and maintenance programs to be conducted on the HI-STAR 60 package and verifies their compliance with the requirements of 10 CFR Part 71.

8.1 Acceptance Tests

Visual inspections and measurements ensure that the packaging conforms to the dimensions and tolerances specified on the licensing and fabrication drawings and that its effectiveness is not significantly reduced. Visual inspections and measurements include the repair and replacement of any important to safety component found to be under the minimum specified thickness, the verification that neutron absorber panels are present as required by the fuel basket design, the verification that the Fuel Impact Attenuators are present as required by the packaging design, the proper marking and labeling of the package and its inspection for cleanliness, and preparation for transport in accordance with written and approved procedures.

The examination of the package welds is performed in accordance with Section 8.1.2 of the application, including alternatives specified in Table 2.1.17.

The Hi-STAR 60 containment boundary is tested by a combination of methods as required by Section III, Subsection NB of the ASME Code. The criteria and basis for leak rate and leakage testing are per ANSI N14.5 requirements. The acceptance criteria from Table 4.3.1 of the application are such that the sum of the various leakage rate tests for all components that comprise the containment boundary are less than the leakage rate acceptance criterion specified in Table 4.1.1 of the application for the containment boundary design specifications.

The integrity of the neutron shield is attained by the manufacturing and installation requirements discussed in Section 8.1.5.3 of the application. Similarly, the integrity of the gamma shield material is attained by the material and dimensional inspections discussed in Section 8.1.5.5 of the package application. Shielding effectiveness tests are performed on the neutron shield and the overpack to verify the integrity of these shields. The applicant maintains samples of each manufactured lot of neutron shield material.

Metamic underwent qualification testing for use in spent fuel pool environments and has adequate mechanical and thermal durability for use in the HI-STAR 60 package. The staff finds that the structural integrity and low porosity of metamic precludes blistering of the metamic panels during vacuum drying, which was verified by vacuum testing of metamic panels during the Metamic qualification program.

Acceptance testing of Metamic is performed using neutron attenuation measurements with a collimated thermal neutron beam, 1-inch diameter, in conjunction with wet chemistry analysis. If the results of the wet chemistry analysis performed on a mixed batch do not meet the acceptance criteria as specified for B_4C content, all panels from the mixed batch are rejected. If a coupon fails the neutron attenuation test, all panels from the mixed batch are also rejected. Visual inspection of each plate is performed to ensure acceptable dimensional tolerances, surface finish, and workmanship criteria. Tests and inspection results are documented and become part of the package quality records documentation.

The first fabricated HI-STAR 60 unit shall be thermally tested to confirm its heat transfer capability. Section 8.1.7 of the application provides a basic description of the testing sequence and the condition for its acceptability.

8.2 Maintenance

Cask closure bolting and impact limiter fasteners shall be visually inspected for damage such as excessive wear, galling, or indentations on the threaded surfaces prior to installation. Damaged bolting and/or fasteners shall be replaced in accordance with the Operations and Maintenance (O&M) manual. Damaged internal threads may be repaired using thread inserts (e.g., HeliCoil®).

Closure lid bolts shall be replaced after every (no more than) 240 bolting cycles. One bolting cycle is the complete sequence of torquing and removal of bolts. Torque requirements are specified in Table 7.1.1 of the package application.

Cask trunnions shall be inspected prior to each fuel loading to verify that no deformation, distortion, or cracking has occurred. Any evidence of deformation (other than minor localized surface deformation due to contact pressure between the lifting device and the trunnion), distortion or cracking of the trunnion or adjacent cask areas shall require repair of the trunnion and/or the cask. Following any major repair of a lifting trunnion, as defined in ANSI N14.6, the load testing shall be re-performed and the components re-examined in accordance with the original procedure and acceptance criteria.

Accessible external surfaces of the packaging (including impact limiters) shall be visually inspected prior to each fuel loading for surface (superficial) and component damage including surface denting, surface penetrations, weld cracking, chipped or missing corrosion resistant veneer, etc. Where necessary, any damage shall be restored per the O&M manual. Damage to components shall be evaluated for impact on packaging safety and components shall be repaired or replaced accordingly. Wear and tear from normal use will not impact cask safety.

Repairs or replacement in accordance with written and approved procedures, as set down in the O&M manual, shall be required if unacceptable conditions are identified. Prior to installation or replacement of a closure seal, the cask sealing surface shall be cleaned and visually inspected for scratches, pitting or presence of an unacceptable surface finish. The affected surface areas shall be restored as necessary in accordance with written and approved procedures. The closure lid seal permeation is less than half of the acceptable helium leakage rate. The HI-STAR 60 closure seals are replaced each time the closure bolts are removed.

A periodic thermal test will be performed on each package at least once within 5 years prior to shipment to demonstrate that the thermal capabilities of the cask remain within its design basis. Section 8.2.4 of the application provides a basic description of the testing sequence and the condition for its acceptability.

8.3 Evaluation findings

The staff has reviewed the qualification and acceptance testing criteria for Metamic, and finds such criteria adequate for the application. The fabrication methods used to manufacture Metamic for production runs will be identical to the methods used to fabricate samples for qualification testing. Therefore, provided that Holtec International, or one of its divisions, remains the sole provider of METAMIC, additional qualification testing of Metamic is not necessary.

The first fabricated HI-STAR 60 package shall undergo thermal testing to confirm its heat transfer capability. If the acceptance criteria specified in the application are not met, the package shall not be accepted until the root cause is determined, appropriate corrective actions are completed, and the package is re-tested with acceptable results.

The staff reviewed the acceptance tests and maintenance programs for the HI-STAR 60 package. The staff found them acceptable.

Based on the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71. Further, the Certificate of Compliance is conditioned to specify that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

CONDITIONS

The following conditions are included in the Certificate of Compliance:

(a) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

(b) The package must be tested and maintained in accordance with Chapter 8 of the application.

(c) The personnel barrier shall be installed and remain installed during transport if necessary to meet package surface temperature and/or package dose rates.

(d) Air transport of fissile material is not authorized.

CONCLUSION

Based on the statements and representations contained in the application, and the conditions listed above, the staff concludes that the Model No. HI-STAR 60 package has been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9336, Revision No. 0, on May 22, 2009.