ENCLOSURE 2

SAFETY EVALUATION REPORT

Model No. HI-STAR 100
Cask System
Certificate of Compliance No. 9261
Revision No. 0

Summary

By application dated October 23, 1995, as revised, Holtec International (Holtec) requested approval of the HI-STAR 100 cask system as a Type B(U)F-85 package. Based on the statements and representations in the application as supplemented, and the conditions listed in the Certificate of Compliance (CoC), the staff concluded that the HI-STAR 100 package meets the requirements of 10 CFR Part 71.

References

Holtec International Safety Analysis Report (SAR) for the HI-STAR 100 Cask System, Revision 8 dated February 22, 1999.

Background

Holtec International application dated October 23, 1995, as revised.

1 GENERAL DESCRIPTION

General Review Description Objective

The objective of the review of the HI-STAR 100 Cask System (HI-STAR 100) is to establish (1) that Holtec has included an overview of relevant package information, including intended use; and (2) a summary description of the packaging, operational features, and contents that provide reasonable assurance that the package can meet the regulations and operating objectives.

1.1 Package Design Information

Following the receipt of the initial application for a CoC, dated October 23, 1995, an initial acceptance review was conducted. The staff determined that the application contained sufficient information to begin review.

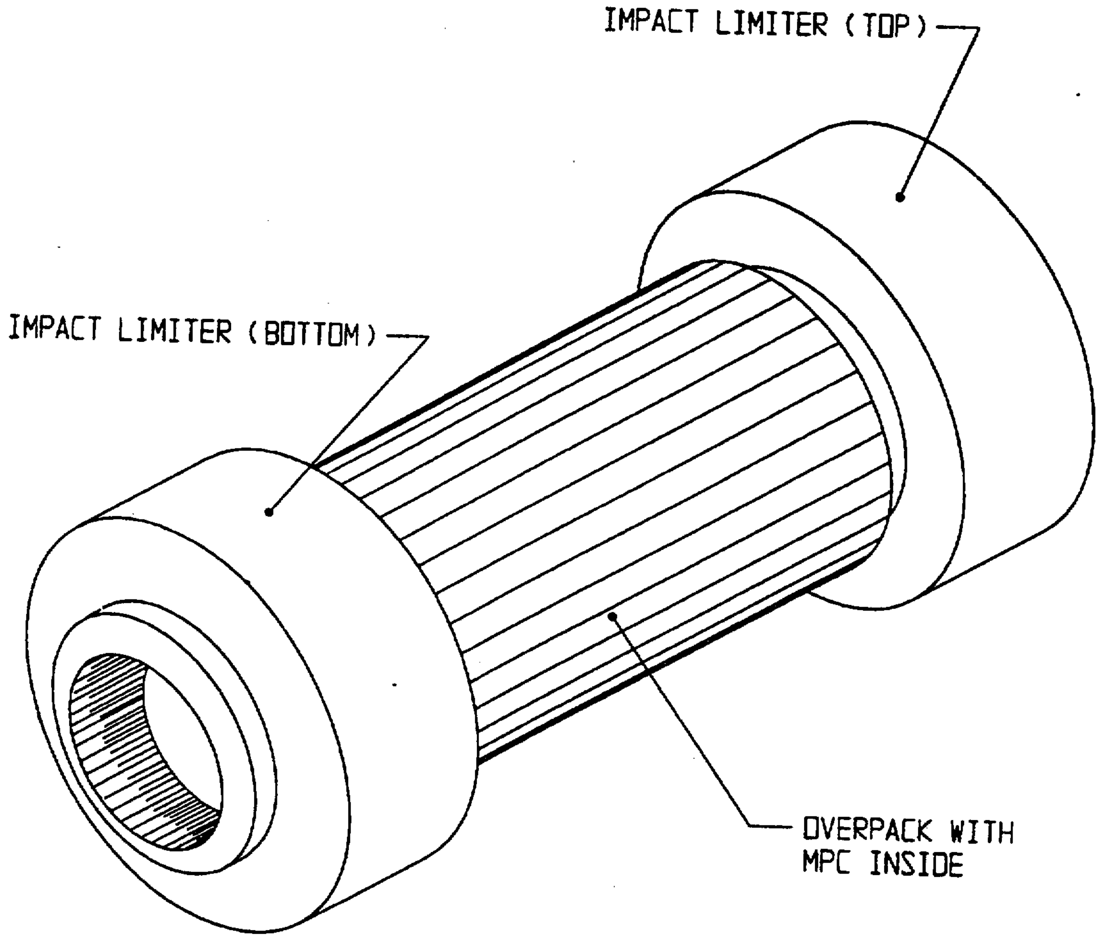
As documented in NUREG-0383, Volume 3, Revision 17, Holtec has an NRC-approved quality assurance (QA) program under 10 CFR Part 71. The approval covered design, fabrication, assembly, testing, procurement, maintenance, repair, modification, and use. Approvals were issued April 4, 1994, and January 13, 1998. The current expiration date is August 31, 1999.

The HI-STAR 100 Cask System (package) will be used for both on-site transfer and off-site transportation of HI-STAR 100 multi-purpose casks, in accordance with 10 CFR Part 72 for on-site movement and 10 CFR Part 71 and 49 CFR Part 173 for off-site transportation.

As indicated in the CoC, Holtec's HI-STAR 100 Cask System is comprised of the MPC, which contains the fuel, and the overpack which contains the multi-purpose canister (MPC). In addition, impact limiters are attached to the top and the bottom of the overpack during transport. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 is designed to contain up to 24 pressurized water reactor (PWR) fuel assemblies. The MPC-68 is designed to contain up to 68 boiling water reactor (BWR) fuel assemblies. A variation of the MPC-68, designated as MPC-68F, may contain BWR fuel debris, as defined in the technical specifications (TS). Both MPCs are identical in external dimensions and will fit into the same overpack design.

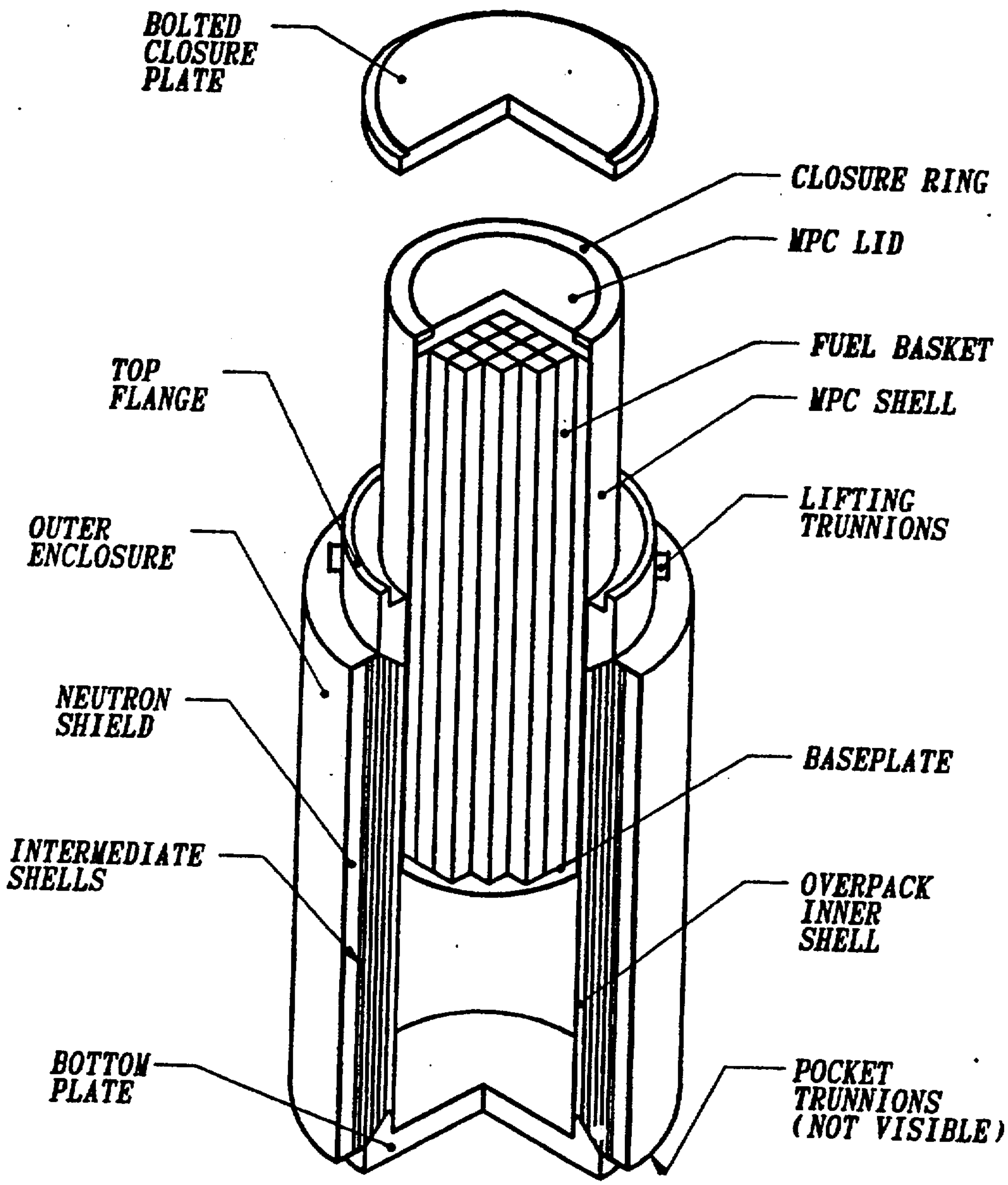
The general arrangement drawings for the HI-STAR 100 Cask System are contained in Section 1.4 of Revision 7 of the SAR. Figures 1 and 2 are basic representations of the HI-STAR 100.

The approved contents for the HI-STAR 100 include: uranium oxide (UO_2) 14x14, 15x15, 16x16, and 17x17 PWR fuel assemblies without control components; UO_2 6x6, 7x7, 8x8, 9x9, and 10x10 BWR fuel assemblies with or without channels; and mixed-oxide (MOX) 6x6 BWR fuel assemblies with or without channels. All PWR fuel assembly types must be stored as intact fuel. Certain BWR fuel assembly types may be stored as damaged fuel or fuel debris placed in damaged fuel containers (DFCs). The enrichment and physical, thermal, and radiological characteristics of the approved contents are given in the CoC. The CoC also provides definitions for intact fuel assemblies, damaged fuel assemblies (DFAs), and fuel debris.



PICTORIAL VIEW OF
HI-STAR 100 PACKAGE

Fig. 1



HI-STAR 100 OVERPACK WITH
MPC PARTIALLY INSERTED

Fig. 2

1.2 Evaluation Findings

1.2.1 General SAR Format

The package has been described in sufficient detail to provide an adequate basis for its evaluation.

1.2.2 Package Design Information

Drawings provided in the SAR contained adequate detail allowing their evaluation by staff against the requirements of 10 CFR Part 71. Each drawing was reviewed and was found to be consistent with the text of the SAR. Further, each drawing contains keys or annotation to explain and clarify information on the drawing.

1.2.3 Package Description

The application for package approval includes a reference to an NRC-approved QA program under 10 CFR Part 71.

1.2.4 Compliance with 10 CFR Part 71

The application for package approval committed to the use of acceptable codes and standards for the package design, fabrication, assembly, testing, maintenance, and use.

The package meets the general requirements of 10 CFR 71.43(a) and 10 CFR 71.43(b).

Drawings submitted with the application (as supplemented) are adequately detailed descriptions of the package to be evaluated for compliance with 10 CFR Part 71.

1.3 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

2 Structural Review

Structural Review Objective

Structural reviews are performed to ensure that the structural performance of the packaging meets the acceptance criteria and requirements of 10 CFR Part 71. Loads and load combinations are reviewed for the normal transport conditions and hypothetical accident conditions specified in 10 CFR Part 71. Structural material specifications are reviewed and compared with acceptable codes and standards. Packaging design assumptions, analysis, critical stresses, and the construction of package components are reviewed to ensure they meet the acceptance criteria of the design codes and standards.

2.1 Description of Structural Design

2.1.1 Descriptive Information Including Weights and Center of Gravity

The HI-STAR 100 transportation package consists of an MPC and a transportation cask with top and bottom impact limiters. The MPC is a completely welded, sealed, cylindrical steel container which contains the fuel basket structure and the spent nuclear fuel assemblies. The transportation cask is a multi-shelled, steel overpack which contains the MPC and serves as the primary containment for the package.

To demonstrate that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71, the applicant performed various structural analyses and evaluations including 1/4-scale drop tests of the impact limiters. The impact limiters were designed to lessen the severity of impact during the 30-ft free drop tests specified in 10 CFR 71.73. For the 1/4-scale drop tests, the impact limiters were modeled to 1/4-scale but the overpack and the internals were simulated for size and weight only.

The weights of the individual components and the overpack, including the impact limiters, are provided in Table 2.2.1 of the SAR. The locations of the calculated centers of gravity (CGs) are presented in Table 2.2.2 of the SAR. All CGs are located on the vertical axis of the overpack since the non-symmetrical effects of the overpack and contents are negligible. Table 2.2.3 of the SAR provides the calculated maximum lift weight when the package is lifted from the spent fuel pool with the heaviest loaded MPC. In addition, bounding weights used for the HI-STAR 100 package design calculations are provided in Table 2.2.4 of the SAR.

2.1.2 Codes and Standards

The ASME Boiler and Pressure Vessel Code (ASME Code), Section III, 1995 Edition with Addenda through 1997, is the governing code for the structural design of the HI-STAR 100 Package. The overpack top flange, closure plate, inner shell, and bottom plate forms the package primary containment boundary. These components are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB for Class 1 components. In addition, the allowable stress limits, the fracture toughness criteria, and design loadings for package containment boundary components are shown to be consistent with the requirements of RG 7.6 and the ASME Section III, Division 3 Code. The balance of the overpack structure is designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for support components.

The MPC consists of an internal fuel basket and the enclosure vessel. The fuel basket is designed and constructed as a core support structure in accordance with Section III, Subsection NG of the ASME Code. The enclosure vessel is designed and fabricated as a Class 1 component in accordance with Section III, Subsection NB of the ASME Code. The applicable sections of the ASME Code for the various components of the HI-STAR 100 Package System are summarized in Table 1.3.1 of the SAR. There are a few deviations from the ASME Code such as material suppliers, code stamping, and weld details and they are listed as exemptions to the Code for the HI-STAR 100 Package System in Table 1.3.2 of the SAR.

2.2 Material Properties

2.2.1 Materials and Material Specifications

The mechanical properties of materials used in analysis include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion.

2.2.1.1 Alloy X Structural Materials

Holtec defined a hypothetical material termed Alloy X for all MPC structural components. SAR Appendix 1.A, Alloy X Description, describes Alloy X as any one of four stainless steel alloys: Types 304, 304LN, 316, or 316LN. The material properties of Alloy X used in the SAR analyses are the least favorable values from the set of candidate stainless alloys (Types 304, 304LN, 316, or 316LN) to ensure that all structural analyses are conservative, regardless of the actual MPC material selected. Appendix 1.A also lists temperature-specific ASME Code values for design stress intensity (S_m) [Table 1.A.1], tensile strength (S_u) [Table 1.A.2], yield stress (S_y) [Table 1.A.3], coefficient of thermal expansion (α) [Table 1.A.4], and thermal conductivity (k) [Table 1.A.5]. Each table lists the minimum value among the four alloys for use in structural calculations (Table 1.A.4 also lists the maximum coefficient of thermal expansion for use in calculations). Table 2.3.1 of the SAR has listed the appropriate minimum (and maximum for thermal expansion) numerical values for the material properties of Alloy X stainless steel versus temperature. These values, taken from ASME Code, Section II, Part D, are used in all analytical calculations for the MPC.

The staff finds that the four alloys selected for Alloy X are very similar austenitic stainless steels with small variations in physical properties over the applicable temperature ranges due to slight variations in chemistry. Type 304 stainless steel may be considered the base alloy for Alloy X. Compared to Type 304 stainless steel, Types 316 and 316LN add 2% molybdenum to increase pitting corrosion resistance; Types 304LN and 316LN reduce carbon content from 0.08 to 0.03% for increased welded condition corrosion resistance; and Types 304LN and 316LN add approximately 0.13% nitrogen for strength to account for reduced carbon levels. In addition, as austenitic stainless steel alloys, there is no transition temperature for brittle behavior as is found in ferritic carbon steels. The staff concluded any of the above alloys are suitable for MPC use. The staff also independently verified the tabulated design values and found them acceptable.

2.2.1.2 Other Structural Materials

Tables 2.3.2 through 2.3.5 of the SAR provide the material properties of carbon-manganese steel and low or nickel alloy steel. These values were also taken from ASME Code, Section II, Part D. For all cask structural materials, the stress limits have been defined at or below the maximum temperature allowed by ASME Code, Section II, Part D, for each material. The materials selected for the cask are consistent with those allowed by ASME Code, Subsection NB. Acceptable requirements include the ASME-adopted specifications given in Section II, Part A, Ferrous Metals, Part B, Nonferrous Metals, and Part D, Properties. The HI-STAR 100 SAR contains detailed tables with temperature-specific material properties and allowable stresses in accordance with ASME Code, Section II for all structural materials. The staff concluded that the material properties used are appropriate for the load conditions of interest (e.g., static or dynamic, impact loading, hot or cold temperature, wet or dry conditions). The staff verified that the SAR clearly references acceptable sources, primarily the ASME Code, for all structural material properties.

The staff concluded material properties and characteristics needed to satisfy these functional safety requirements will be maintained over the approval period. The life cycle may include conditions experienced during cask fabrication, loading, transport, emplacement, storage, transfer, retrieval, and decommissioning. Service conditions include normal and off-normal operations, accidents, and natural phenomena events. The staff concluded the materials of construction used for the MPC and overpack, primarily stainless steels for the MPC and carbon-manganese, and low or nickel alloy steels for the overpack, are compatible with the environment during loading, storage, and unloading of the MPC. The stainless steels used for the MPC have a long, proven history in nuclear service.

2.2.1.3 Impact Limiters

For the aluminum honeycomb impact limiters, the force-deflection characteristics were verified by testing. Testing of the impact limiters was carried out statically and dynamically through a 1/4-scale drop test at Oak Ridge National Laboratory. The force-deflection curve of the impact limiter is provided in the SAR for all directions evaluated for the packaging.

2.2.1.4 Welds

The applicant stated that all materials utilized in the welding of the cask components comply with the provisions of the appropriate subsections of Section III and Section IX of the ASME Code. The staff reviewed confinement boundary weld designs for compliance with the design code used and found them acceptable. The MPC closure weld (0.75 inches on the MPC-24 and MPC-68, 1.25 inches on the MPC-68F) is a partial penetration weld but will perform its intended structural and confinement functions. A factor of 0.45 has been applied to the stress analysis of the MPC lid and closure ring welds, reducing the stress on the weld material. A redundant closure of the MPC is provided by the MPC closure ring in accordance with 10 CFR 72.236(e). The cask welds were well-characterized on engineering drawings and diagrams using standard welding symbols and/or notations in accordance with American Welding Society Standard A2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination." The HI-STAR 100 materials (stainless, carbon, and low alloy steels) are readily weldable with commonly available welding techniques. SAR Table 8.1.3 contains nondestructive examination (NDE) requirements and acceptance criteria for all HI-STAR 100 welds.

Additional materials requirements apply for structural designs governed by ASME B&PV Code, Section III, Subsection NB. SAR Table 2.2.15 states an ASME Code exception for material suppliers for the MPC, MPC basket, and overpack. Specifically, Holtec will use approved suppliers with Certified Material Test Reports in accordance with NB-2000 requirements. The staff concluded this practice is acceptable.

2.2.1.4.1 Critical Flaw Size Determination for MPC Closure Welds

In accordance with Interim Staff Guidance (ISG) - 4, Cask Closure Weld Inspections, Holtec proposed to examine the austenitic stainless steel MPC closure welds with a multiple-layer dye penetrant (PT) examination in lieu of the ASME Code required volumetric examination. The PT will be done in accordance with ASME Section V, Article 6, Liquid Penetrant Examination, with ASME Section III, NB-5350 acceptance standards. Holtec used a design stress-reduction factor of 0.45 applied to the weld design, exceeding the 0.8 factor required by ISG-4. Holtec calculated the critical flaw size for the MPC closure welds in Holtec Position Paper DS-213, Acceptable Flaw Size in MPC Lid-to-Shell Welds, Revision 2, dated February 23, 1999. The Holtec analysis applied a bounding stress (52.662 thousand pounds per square inch (ksi)) to a hypothetical 360° 50%-through-wall crack (0.375 inches deep for the MPC-24 and MPC-68, 0.625 inches deep for the MPC-68F). Position Paper DS-213 concludes these large hypothetical cracks would not grow under normal or accident conditions, thus preserving structural and containment integrity. This paper concluded that PT examination on the root, final, and every 3/8 (0.375) inches of the weld would be acceptable.

The staff performed an independent calculation to determine critical flaw size for the MPC weld using the pc-CRACK[1] software package. The stress levels used in the calculation were based on Section 6.0 of Position Paper DS-213, which was based upon the bounding 10 CFR Part 71 top end drop described in Section 2.L.8.1.1 of the SAR. Based upon a 624,000 lb-force load on the MPC lid (SAR Section 2.L.8.3), the shear stress in the MPC lid weld is 4.717 ksi (this calculation assumes a 5/8-inch weld as they neglect the root pass). Actual weld size is 0.75 inch for a stress level 3.931 ksi for the MPC-24 & MPC-68 and 1.25 inches for a stress level 2.358 ksi for the MPC-68F (the only MPC that is required for containment purposes). The staff used the 4.717 ksi stress value and factors of $\sqrt{2}$ (from ASME Section XI, IWB-3600 for emergency and faulted conditions) and 2 (to account for any uncertainties in stress calculations) to determine stress inputs for the critical flaw calculations. The staff used bounding material toughness properties for the Type 316 stainless steel submerged arc weld J-R curve taken from EPRI-TR106092. The results of the calculations are shown below:

[1]Structural Integrity Associates, Version 2.0.

Table 2.2.1-1
NRC Critical Flaw Size Calculation Results
Critical Flaw Depth [inches]

Weld Size [inches]	**Applied Stress (ksi)**		
	4.717 ksi (base)	6.671 ksi (base x $\sqrt{2}$)	9.434 ksi (base x 2)
0.75 in (MPC-24, MPC-68)	0.534 in	0.495 in	0.451 in
1.25 in (MPC-68F)*	0.870 in	0.802 in	0.727 in

* NOTE: The MPC-68F is the only MPC which is required for secondary containment purposes during transportation in accordance with 10 CFR 71.63(b) due to the presence of failed fuel. The HI-STAR 100 overpack is the primary containment for all MPC designs during transportation.

The values above represent the depth of a 360° circumferential surface crack which is the bounding critical flaw size for this design. The results of the staff's analysis show that the critical flaw size in the MPC design is greater than the depth interval for PT examination (0.375 inches), thus ensuring that any flaw in the MPC closure weld would be detected before it became large enough to affect the structural design of the MPC. It should be noted that only the MPC-68F is relied upon for containment during transportation. The PT will include the root and final layers and sufficient intermediate layers (every 3/8 inch) to detect critical flaws in accordance with ISG-4.

2.2.1.5 Bolting Materials

The material properties of the bolting materials used in the HI-STAR 100 system are given in SAR Table 2.3.5. Bolting materials used in the HI-STAR 100 system are specified in accordance with appropriate ASME specifications: SB-637-N07718, SA-564-630, and SA-705-630. Procurement in accordance with these specifications will help assure mechanical properties and proper heat treatment, as required by the applicable specification. The staff found these bolting materials acceptable. The staff also independently verified the tabulated design values and found them acceptable.

2.2.1.6 Non-structural Materials

In the case of non-structural materials, such as the neutron shield material (NS-4-FR), Boral™ neutron absorber, and aluminum heat conduction elements, no ASME standard is available. However, the SAR provides adequately documented material properties that are important for the design and fabrication of the packaging. Pertinent material properties needed to define the material for analysis are included. Non-structural materials such as the Boral panels and Holtite-A (Holtec neutron shield material designation for NS-4-FR) are included in the structural analyses by weight only. Materials that function as neutron absorbers and gamma shields should be fabricated from materials that can perform well under conditions of service that are appropriate for these components over the design period. Boral has a long, proven history in worldwide nuclear service and use in other spent fuel storage and transportation casks. In accordance with NUREG-1536, only 75% credit is taken for the B^{10} in Boral. The SAR includes technical information and scientific studies on the NS-4-FR which show it will provide acceptable properties over the service life of the HI-STAR 100 system. The staff concluded Boral and NS-4-FR should not creep or slump to an extent that impairs the capability to perform its safety function during storage and accident conditions.

2.2.1.7 Brittle Fracture of Materials

The applicant considered the potential for brittle fracture, especially for cask system components that are subject to impact during exterior handling and transfer operations.

2.2.1.7.1 MPC

Alloy X, used exclusively for the MPC, is an austenitic stainless steel alloy. Thus, there is no transition temperature for brittle behavior as is found in ferritic carbon steels. No testing is necessary as these materials are inherently resistant to brittle fracture.

2.2.1.7.2 HI-STAR 100 Overpack

For the ferritic steels used in the HI-STAR 100 overpack, SAR Section 2.1.2.3 specifies that each plate or forging for the helium retention boundary will be drop-weight tested in accordance with Regulatory Guides (RGs) 7.11 and 7.12 and Charpy V-notch tested in accordance with ASME Section III, Subarticle NB-2300. Tables 2.1.22 and 2.1.23 contain fracture toughness test criteria for the HI-STAR 100 overpack materials. The staff concluded that with the alloys selected and testing performed, no restrictions regarding cask handling at low temperatures are necessary.

2.2.1.7.3 Impact Limiters

The HI-STAR cask impact limiters used during spent fuel transport are made of corrugated sheets of aluminum alloy 5052 and are not subject to brittle failure.

2.2.1.7.4 Brittle Fracture Conclusion

The staff verified that the materials of structural components, whose structural integrity is essential for the package to meet regulatory requirements, have sufficient fracture toughness to preclude brittle fracture under the specified normal conditions of transport and hypothetical accident condition temperatures and loads. In addition, brittle fracture will be precluded for the containment vessel under severe impact loads at the lowest service temperature. Fracture toughness criteria for ferritic steel packaging containment vessels are demonstrated in accordance with RGs 7.11 and 7.12.

2.2.1.8 Materials and Materials Selection Conclusion

The staff reviewed packaging materials of construction and their specifications and found them to be acceptable. Material specifications and properties for structural materials (austenitic stainless steels for the MPC; carbon, low-alloy, and nickel alloy steels for the overpack; and aluminum alloys for the impact limiters) consistent with the design code selected (ASME Section III). Relevant material properties for structural materials are listed in detail in SAR Section 2.3.

The staff concluded that the material properties used are appropriate for the load condition (e.g., static or dynamic impact loading, hot or cold temperature, wet or dry conditions, etc.), and that appropriate temperatures at which allowable stress limits are defined are consistent with those temperatures expected in service.

2.2.2 Prevention of Chemical, Galvanic, or Other Reactions

The staff reviewed the cask structural materials that are in direct contact with each other and verified that they will not produce a significant chemical or galvanic reaction and the attendant corrosion or combustible gas generation [10 CFR 71.43(d)][2]. SAR Table 2.4.1 lists material compatibility with operating environments. No appreciable galvanic reactions are expected with the materials of construction. In addition, SAR Section 7.1.5, MPC Closure, step 25A, directs that a vacuum source be connected to the vent port to keep moist air from condensing on the MPC lid weld area. This will reduce the concentration of any combustible gases if any were present during welding operations.

The SAR describes three coatings to be used on the ferritic steels in the MPC overpack: Thermaline 450, Carboline 890, and Dow-Corning Sylgard 567. As shown in the product data sheets included in Appendix 1.C, Thermaline 450 is a polymer with outstanding barrier protection against chemical exposures, and Carboline 890 exhibits excellent chemical resistance. Carboline 890 is qualified for Nuclear Level I service in most nuclear power plants. Although unlikely in the short-term exposure during loading, if either were to react to the acidic spent fuel pool environment, there would be no generation of hydrogen. In addition, only the coating on the outside of the HI-STAR 100 overpack will be in contact with the spent fuel pool. The interior of the overpack will be filled with clean, non-Borated water and kept in place with the annulus seal before insertion. Dow-Corning Sylgard 567, a silicone sealant, will be used as a protective coating for the ferritic steel shells in the overpack. It will be injected through ports in the overpack after fabrication and then the injection ports sealed. No adverse reactions are expected as this coating will not be exposed to the environment during loading, transportation, or unloading operations.

The staff reviewed the materials and coatings of the package to verify that they will not produce a significant chemical or galvanic reaction among packaging components, among packaging contents, or between the packaging components and the packaging contents. The staff also considered inleakage of water, no credible reactions are expected.

The all-stainless steel construction of the MPC eliminates any credible generation of hydrogen or other flammable gases. No embrittling effects of hydrogen on the metallurgical state of the packaging materials should occur for the austenitic MPC or ferritic overpack. For metallic

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[2]NRC Bulletin 96-04: Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks

components of the package that may come into physical contact with one another, no eutectic reactions should occur.

2.2.3 Effects of Radiation on Materials

The staff considered any potential damaging effects of radiation on the packaging materials, including degradation of seals and sealing materials and degradation of the properties of coatings and structural materials [10 CFR 71.43(d)]. The MPC austenitic alloys, Alloy X, and ferritic alloys for the HI-STAR 100 overpack are not subject to radiation embrittlement during spent fuel transportation due to the low radiation dose and energy levels as compared to reactor vessel service.

2.2.4 Materials Conclusion

The Holtec SAR contains a detailed description of material properties in Section 2.3, Mechanical Properties of Materials. The staff concluded that the materials for construction of the HI-STAR 100 system are acceptable for the described structural, thermal, shielding, criticality, and confinement functions. To the maximum credible extent, there are no significant chemical, galvanic or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry or wet environment conditions. No adverse effects of radiation on materials are expected, and the package containment is constructed from materials that meet the requirement of RGs 7.11 and 7.12.

2.3 Lifting and Tie-Down Standards for All Packages

2.3.1 Lifting Devices

The HI-STAR 100 package has two types of lifting devices which are used for the lifting operations. The loaded package is designed to be lifted vertically by two lifting trunnions. The trunnions are located on the top forging at 0° and 180° azimuths of the overpack. Four 5/8-inch diameters, SB637-NO7718, eye bolts are used to lift the overpack top closure plate. The bolts are analyzed based on a minimum lift angle of 45° from the horizontal. The overpack is also equipped with two pocket trunnions near the bottom end. The pocket trunnions, however, are not used for lifting. They are used as pivots during upending and down ending of the package. In addition, they function as the rear support for resisting longitudinal tie-down forces. The lifting trunnions and the closure plate eye bolts are conservatively designed to meet the requirements of NUREG-0612 with a safety factor of six (6) against material yield strength and ten (10) against material ultimate strength. In addition, the structural analysis is based on lifting loads that have been increased by a load factor of 1.15 in accordance with the guidelines of the Crane Manufacturer's Association of America (CMAA), Specification No. 70, 1988, Section 3.3, which stipulates a dynamic factor equal to 0.15 for slowly executed lifts. The analysis evaluated the lifting capacity, shearing capacity, bending moments, and embedment length of the trunnions and the bolts to show the resulting stresses are within the allowable values specified in the ASME Code. The lifting device design exceeds the requirements of 10 CFR 71.45(a) which only requires a safety factor of three (3) against yielding for any lifting attachment that is a structural part of a package. The analysis shows that, under excessive loads, the lifting trunnions would fail by shearing the threads in the top forging, and the lifting bolts for the top closure plate would also fail by shearing. These failure modes will not impair the ability of the package to meet other safety requirements. Thus, it is concluded that the HI-STAR 100 package lifting devices have met the requirements of 10 CFR 71.45(a).

A loaded MPC can be inserted or removed from the cask by the lifting bolts installed on the MPC top lid. The strength of the lifting bolts and the base metal are evaluated to meet the requirements of NUREG-0612 for non-redundant lifting systems. The evaluation included an additional load factor of 0.15 for the loaded MPC and checked the adequacy of the thread engagement length in the top lid. The evaluation results showed that the lifting bolts provided more than six (6) times the material yield strength and ten (10) times the material ultimate strength when lifting a fully loaded MPC.

2.3.2 Tie-Down Devices

The package top end is supported by a saddle support structure. The package bottom end is supported by a transport frame with external supports to engage the pocket trunnion recesses. Vertical and transverse tie-down forces are resisted by bearing onto the saddle structure and the transport frame. Uplifting forces are resisted by the hold-down straps and the pocket trunnions. Forward longitudinal force is resisted by bearing of the saddle support to the shear-ring on the top forging. Longitudinal force toward the bottom is resisted by the external supports acting at the pocket trunnion recesses. Since the shear-ring and the pocket trunnions are structural parts of the package, they are evaluated and designed to meet the requirements of 10 CFR 71.45(b).

As specified in 10 CFR 71.45(b), the tie-down devices are analyzed for a static force applied to the CG of the package having a vertical component of two times the weight of the package, a horizontal component in the transverse direction of five times the weight of the package, and a horizontal component along the longitudinal direction of ten times the weight of the package. The weight of the package is taken as 280,000 lb which is the maximum weight of the HI-STAR 100 package with impact limiters installed. The analysis results have shown that the HI-STAR 100 package tie-down system is capable of withstanding the applied force specified in 10 CFR 71.45(b) without generating stress in package components in excess of the material yield strength.

However, 10 CFR 71.45(b)(3) requires that failure of any tie-down device which is a structural part of a package under excessive loads would not impair the ability of the package to meet other requirements of 10 CFR Part 71. The saddle support at the top end of the overpack and the external shaft that inserts into and provides the support at the two pocket trunnions at the base of the overpack are not an integral part of the package but are a part of the package tie-down system. The front and rear supports are designed to meet the American Association of Railroads (AAR) Field Manual, Rule 88. The design loading is a 7.5 g longitudinal, 4.0 g vertical, and 1.8 g lateral static force applied at the package CG. Since the AAR design loading is less than the tie-down force specified in 10 CFR 71.45(b), under excessive loads, either the front saddle support or the external shaft of the rear support would fail first or their connections to the rail car would fail. As a result, neither the shear-ring nor the pocket trunnions would subject to excessive loads to cause any damage to the overpack. Thus, under excessive loads, the overpack would remain intact and its safety functions would not be adversely affected.

2.4 General Considerations for Structural Evaluation of Package

2.4.1 Evaluation by Analysis

The structural components of the package (i.e., overpack, neutron shield, and MPC) were evaluated by hand calculations using well-developed theory or by finite element analysis using the ANSYS computer code. The SAR described the analytical models, assumptions, and methods of analysis. The specific analysis performed and their results are discussed in Sections 2.5 and 2.6, respectively, for normal conditions of transport and the hypothetical accident conditions. The analysis results and margins of safety are presented in the SAR to demonstrate structural adequacy of the package design.

2.4.2 Evaluation by Test

The impact limiters are the only HI-STAR 100 packaging components evaluated by testing. The impact limiters are constructed from uniaxial and cross-core (biaxial) aluminum honeycomb materials and encased in a stainless steel shell. The impact limiter has an inner stiffening cylinder and gussets made from SA516, grade 70 steels. The construction of the impact limiter is quite complex and is shown on SAR Drawing No. 1765, sheet 1 thru 7. The purpose of the impact limiter is to limit the package deceleration under the postulated free drop events to a specified maximum design value. For the hypothetical 30-ft (9-meter) free drops, the impact limiter is designed to limit the maximum package rigid body deceleration to 60 times the gravity (e.g., 60 g). The applicant performed 1/8-scale model static tests of the impact limiter to confirm the adequacy of the energy-absorbing capability of the impact limiter and the force-deflection relationships to be used in the analytical model. The impact limiter attachments and its performance under the hypothetical 30-ft drop conditions are confirmed by the 1/4-scale, 30-ft drop tests. The scale models simulated the dimensions, fabrication details, and design features of the full-size impact limiter. The force-deflection curves of the impact limiter obtained by the static tests validated the applicant's analytical model except for the end drop orientation. It was concluded that for the end drop condition, the static test was influenced by the elastic behavior of the stiffened impact limiter shell structure. However, the 30-ft end drop 1/4-scale dynamic test results were shown to be very close to the analytically predicted results. In an attempt to show the 30-ft end drop test results were not overly sensitive to filter frequency, the end drop test results were filtered by three filter frequencies, 450 Hz, 550 Hz, and 1250 Hz to show that the package rigid body maximum deceleration is not significantly changed due to changes of filter frequency. Based on the 1/4 -scale end drop test results, the impact limiter analytical model is considered to be acceptable for predicting the dynamic responses of the package during the 30-ft end drop condition. The design, testing, and computer simulation of the impact limiter is presented in Appendix 2H of the SAR.

2.5 Normal Conditions of Transport

2.5.1 Heat

Both differential thermal expansion and thermal stresses due to thermal gradients are evaluated for the normal heat condition. The differential thermal expansion analysis is performed to show the packaging components and the contents can undergo thermal growth without generating large thermal stresses because of interferences or restrictions by the adjacent components. Thermal stress analysis is performed to calculate the stresses in the overpack due to non-uniform distribution of temperatures and thermal gradients.

The applicant performed closed form hand calculations to show that adequate gaps are provided so that no physical interference (e.g., contact) will develop between the overpack and the canister, and between the canister and the fuel basket due to unconstrained thermal expansion of each component under the normal heat condition. The differential thermal expansion analysis is based on bounding temperatures (Tables 3.4.10 and 3.4.11) of the components from thermal evaluations in Chapter 3 of the SAR.

A three-dimensional finite element model of the HI-STAR 100 overpack is used to calculate the stresses and stress intensities due to the combined effects of thermal gradients, pressure, and mechanical loads under the normal conditions of transport. The finite element model is a three-dimensional, half symmetry model of the overpack and is constructed using the ANSYS Code. The temperatures and the temperature distributions applied to the analytical model was calculated in Chapter 3 of the SAR. The finite element analysis conservatively assumed a design internal pressure of 100 psig. The 100 psig pressure loads were applied to the cavity (i.e., inner surfaces) of the overpack. In addition to internal pressure, the analysis also included the closure bolt pre-loads and fabrication loads. The stress analysis is a linear-elastic finite element analysis using the ANSYS Code. The results showed that the combined stress due to thermal, pressure, bolt pre-loads, and overpack fabrication loads are within the allowable values of the design code as shown in Appendix 2.AE of the SAR.

2.5.2 Cold

As shown in Table 2.1.8 of the SAR, the normal cold condition was evaluated for two loading cases: (1) extreme cold ambient temperature at -40°F plus closure bolt pre-loads and the fabrication loads; and (2) cold ambient temperature at -20°F plus the 1-ft side drop impact load, closure bolt pre-loads, and the fabrication loads. Since the internal pressure will generally reduce thermal stresses due to contraction, both load cases have neglected the internal pressure. The stress analysis for the normal cold condition is similar to the analysis performed for the normal heat condition using the same finite element model and the ANSYS Code. The results showed that the stresses in the overpack were within the allowable limits and the cold condition would not adversely affect the structural performance of the package.

Differential thermal contractions of the package components were performed based on the steady-state temperature of the package. It was assumed that the steady-state temperature of all components in the package will go up or down by the same amount of change in the ambient temperature. The calculated results showed that the clearances between the MPC basket and canister, as well as those between the canister and the overpack inside surfaces, are adequate to preclude a cold temperature induced physical interference.

2.5.3 Reduced External Pressure

The effects of a reduced external pressure equal to 3.5 psia to the package containment boundary (e.g., overpack inner shell) is insignificant and enveloped by the design internal pressure during the hypothetical accident conditions under which the internal pressure were increased from 100 psig to 125 psig. The applicant also performed analysis to show that the enclosure shell of the neutron shield is capable of withstanding such a pressure drop without generating stresses above the design allowable values of the code. Thus, the package design meets the reduced external pressure condition.

2.5.4 Increased External Pressure

The package is designed for an external pressure of 300 psig under the hypothetical accident conditions. Therefore, an increase of external pressure to 20 psia will have negligible effects on the package.

2.5.5 Vibration

During normal transport, the loads generated from vibratory motions incident to transport are much smaller than the design tie-down force specified in 10 CFR 71.45(b)(1). The applicant stated that the MPC and basket structure has the lowest natural frequency of vibration among package components. The applicant performed analysis in Appendix 2.K of the SAR to show that the MPC and the basket structure natural frequency of vibration are in excess of 469 Hz, which is much higher than the vibration frequencies of the rail car during normal transportation. Consequently, resonance among package components during transport is unlikely and the vibration effects to the package are not significant.

2.5.6 Water Spray

All exterior surfaces of the package are welded steel. The water spray will have no effects on the package performance.

2.5.7 Free drop

The package is transported solely in a horizontal orientation by rail car. Because of the weight and size, once the package is secured on the rail car, it will not be moved or lifted again during transport. Therefore, the package is analyzed for a 1-ft free drop in the package transport orientation (i.e., horizontal) only. The design g-load for the 1-ft free drop is 17 g (SAR, Table 2.1.10) as calculated in Appendix 2.H of the SAR. The package is analyzed to show that, when exposed to a 17 g impact load, 100 psig internal pressure, and with the effects of closure bolt pre-loads and fabrication loads, the combined stresses in the overpack will meet the allowable stress limits specified by the design code. The smallest safety factor for the fuel basket structure is 1.26, for the MPC enclosure shell is 1.41, and for the overpack is 1.44 as shown in Appendix 2.AE and Tables 2.6.2 through 2.6.4 of the SAR. The finite element model and analyses performed for the MPC and the fuel basket structure are explained under the hypothetical accident conditions.

2.5.8 Corner Drop

The corner drop test is not applicable (per 10 CFR 71.71) because the weight of the package exceeds 220 lb and neither wood nor fiberboard is used as materials of construction.

2.5.9 Compression

The compression test is not applicable because the weight of the package exceeds 11,000 lb.

2.5.10 Penetration

The exterior shells and surfaces of the package are capable of withstanding the impact forces imposed by the normal condition penetration test. There are no valves or relief devices which could be impacted on by the 13-lb steel bar used in the test.

2.6 Hypothetical Accident Conditions

2.6.1 Free Drop

The evaluation of the package for the 30-ft free drops under hypothetical accident conditions included both finite element computer analysis and 1/4-scale model 30-ft drop tests. To lessen the impact load received by the package during a 30-ft drop, the package is equipped with top and bottom impact limiters to absorb the impact energy. The impact limiters are constructed from uniaxial and cross-core aluminum honeycomb materials encased in stainless steel shell. The applicant performed crush tests of the aluminum honeycombs to establish its crush strength. It was shown that the aluminum honeycombs were not sensitive to the range of ambient temperature changes but it should be pre-crushed to get rid of the initial peak of crush strength. The pre-crushed aluminum honeycomb will have almost constant crush strength until the crush distance reaches 60-70% of its thickness as shown in Figure 2.H.2.1 of Appendix 2.H of the SAR.

Based on the crush geometry of the impact limiter and the crush strength of the honeycomb material, the applicant developed force-deflection relationships (curves) for the various drop orientations of the impact limiter. These analytically developed force-deflection relationships of the impact limiter for end, side, and CG over corner drop orientations were validated by 1/8-scale model static compression tests as shown in Figures 2.H.10.1 thru 2.H.10.3 of Appendix 2.H of the SAR.

The force-deflection curves for each drop orientation were used to predict analytically the rigid-body responses of the package to a 30-ft drop impact in that orientation. The impact limiter is simulated by a nonlinear spring whose static force-deflection curve is known for the drop orientation. The 30-ft drop event is simulated by Newton's equations of motion. The solutions of the dynamic force equilibrium equation are obtained by the commercially available code WORKING MODEL. The WORKING MODEL Code has been validated by the QA system of the applicant for this purpose. The orientations of the drop analysis performed were end, side, CG over the corner, and oblique drops with slap down effects. The applicant also performed 1/4-scale model 30-ft drop tests to verify the analytically predicted results. It is shown in Table 2.H.4 of Appendix 2.H of the SAR that the analytically predicted cask rigid-body accelerations, maximum impact limiter crush distances, as well as the impact durations, all compared very well with those of the 30-ft drop test results. The applicant performed additional analysis by varying the aluminum honeycomb crush strength (within the specified manufacture tolerances) and the package weights. The responses to the various combinations of package weight and crush strength are shown in Table 2.H.5 of the SAR. It can be seen that the maximum rigid body deceleration of the package is less than 60 g regardless of the orientation of impact. Thus, a bounding rigid-body deceleration of 60 g is used in the evaluation of the package for the 30-ft free drops under the hypothetical accident conditions of 10 CFR 71.73. Dynamic load factors (DLF) are determined in Appendix 2.K of the SAR. For the MPC fuel baskets, modeled by multi-degree of freedom simulations, the DLF is shown to be less than 1.05 for the pulse durations expected for the transportation package 30-ft drop impacts. For other package components, modeled by single degree of freedom systems, there is no component where the DLF exceeds 1.04. Dynamic amplifications are considered in the stress analyses by showing the minimum safety factors of the analysis results are always greater than the DLF calculated in Appendix 2.K.

2.6.1.1 End Drop

The end drop evaluations consist of the stress analysis of the overpack and the MPC, buckling analysis of the overpack and the MPC, and the closure bolt analysis. The analysis performed and the results are discussed in the following sections.

2.6.1.1.1 Stress Analysis

The overpack is evaluated under both a top end and a bottom end drop impact. In both cases, the impact limiter reaction is assumed to act over the entire lid area. A 60 g deceleration force is assumed. The analysis is based on the combined effects of impact, thermal gradient, bolt pre-loads, and fabrication loads. The stress analysis is a linear-elastic analysis using a three-dimensional finite element model of the overpack and the ANSYS Code. The stresses in the overpack due to the combined loading were shown to be within the allowable limits. The results of the stress analysis are presented in Appendix 2.AE of the SAR and the safety factors are shown in Tables 2.7.5 and 2.7.6 of the SAR for the hot and cold environments.

The MPC is also evaluated for both a top end and a bottom end drop impact. For bottom end drop impact, the critical MPC component is the top closure plate. The closure plate is a 9-inch thick circular plate edge welded to the canister shell by partial penetration welds. The analysis conservatively assumed the closure plate is simply supported around the circumference and uniformly loaded by its own weight amplified by the 60 g DLF and an external pressure of 125 psig. The resulting stress in the closure plate is small because of the very large thickness of the plate. The edge welds to the canister shell were analyzed based on a joint efficiency factor of 0.45, an allowable stress of 0.3 S_u, and a weld thickness of only 0.625 inches (the actual weld size is 0.75 inches). The analysis was performed in Appendix 2.L of the SAR and the results showed that the closure plate and the welds to the canister shell met the stress requirements of the design code. The minimum safety margin is 1.37 and is in the welds of closure plate to canister shell.

For top end drop, the MPC baseplate and adjoining portion of the canister shell are analyzed by finite element analysis in Appendix 2.N of the SAR. The baseplate and the lower portion of the canister were modeled as a plate and shell structure. The analysis assumed a 60 g deceleration load and an external pressure of 60 psi. The calculated stress intensity at the center of the baseplate from the 60 g impact load and 60 psi external pressures is 22,120 psi. The margin of safety is computed to be 2.04. The maximum combined stress intensity in the canister shell is 31,474 psi and the margin of safety is equal to 1.139.

2.6.1.1.2 Buckling Analysis

The overpack inner shell, the MPC canister shell, the fuel basket panels, and the fuel rods were all evaluated for buckling for the 30-ft end drops. Structural stability of the overpack inner shell under the end drop is assessed in Appendix 2.J of the SAR. The inner shell is evaluated for elastic and plastic stability in accordance with ASME Code Case N-284. It was shown that all interaction equations set by the Code Case are met and that it is the material yield strength rather than the buckling load limits that governs the overpack inner shell. The minimum factor of safety against overpack inner shell buckling is equal to 2.2.

Structural stability of the MPC enclosure shell under the 30-ft end drop is also assessed in Appendix 2.J of the SAR. The enclosure shell is evaluated for elastic and plastic buckling in accordance with ASME Code Case N-284. It was shown that the material yield strength limit controls the MPC enclosure shell. Therefore, buckling of the enclosure shell will not occur and the minimum safety factor against buckling is calculated to be 1.92. Stabilities of the fuel basket panels were evaluated by comparing the calculated axial stress in the panel due to a 30-ft end drop to the elastic critical buckling stress of the panel. It was shown that the critical buckling stress is about 20 times greater than the calculated panel axial stress. Therefore, the fuel basket panels will not buckle under the hypothetical 30-ft end drop accident condition.

Structural stability of the fuel rods is evaluated based on Euler's buckling load. The buckling analysis included the fuel pellet weights and used the irradiated material properties. The analysis showed that local buckling of the fuel rod may occur when the equivalent end drop g-load exceeds 13.6 g. However, the fuel rods are supported by the fuel basket panels and the fuel rod lateral displacements are limited by the fuel rod spacings and available gap between the fuel assembly and the fuel cell panel. The applicant evaluated the fuel rod in the most limiting fuel assembly and assumed that the maximum fuel rod lateral displacement equals to the sum of all the spacings between fuel rods plus the gap between a fuel assembly and fuel cell panel. The analysis showed that the bending stresses in the fuel rod due to this relatively large lateral displacement is still below the fuel rod cladding material yield stresses. However, at this point the unsupported span length is approximately reduced to 50% of the original span length. The critical buckling axial load for the reduced span is shown to be 64.8 g. The analysis is based on Euler's elastic buckling load and conservatively assumed simply supported ends. As the result, it can be concluded that under the design g-load of 60 g for an end drop, the fuel rods will not rupture for the most limiting fuel assembly.

2.6.1.1.3 Fuel Support Spacers

Upper and lower fuel support spacers may be used to position the active fuel region of the spent fuel within the neutron absorber regions of the fuel basket. The spacers must not yield or buckle under the maximum compressive loads due to the 30-ft hypothetical end drops. The end drops produced the most limiting loads on the fuel support spacers. Therefore, the fuel support spacers are evaluated only for the end drop impact conditions. The fuel support spacers are analyzed for an equivalent 60 g axial impact load. Detailed analysis and stress calculations are provided in Appendix 2.O of the SAR. The analysis results showed that the maximum axial stress is less than the allowable stress of 36,950 psi with a minimum safety margin of 1.1 for the heavier PWR assembly. The safety factor against elastic buckling is 5.7 for the upper spacers and larger than 10 for the lower spacers.

2.6.1.1.4 Closure Bolt Analysis

Stresses develop in the overpack closure bolts due to pre-load, pressure load, thermal loads, and drop impact loads. The analysis of closure bolts are presented in Appendix 2.U of the SAR. The analysis follows the methodology of NUREG/CR-6007 and is conservatively assuming a nearly vertical drop orientation so that the closure bolts will resist the full impact loads of a loaded MPC. The loading condition chosen is bounding for the closure bolts. As a result, only one evaluation is presented for the closure bolts. The analysis results showed that the calculated bolt stresses from the combined loads are less than the allowable bolt stresses and the minimum safety factor is 1.30.

2.6.1.2 Side Drop

2.6.1.2.1 Stress analysis

The overpack and the MPC were analyzed for the 30-ft side drop conditions. Although both the 30-ft drop test and the analysis results showed a smaller g-load under the side drop condition, the analysis is conservatively based on an equivalent impact load of 60 g. A three- dimensional, half-symmetry finite element model of the overpack was constructed using the ANSYS Code. The applied loads included thermal loads (hot or cold), internal pressure, bolt pre-loads, fabrication loads, and inertia loads from the MPC and the overpack weights amplified by the 60 g impact. The applied loads are counter-balanced by the impact forces (e.g., reactions) from the impact limiters. The combined stresses in the overpack are presented in Appendix 2.AE of the SAR. Table 2.7.5 of the SAR shows the calculated safety factors of the overpack components for the hot ambient condition and Table 2.7.6 of the SAR shows the calculated safety factors for the cold ambient condition.

The MPC enclosure shell and the basket structure are evaluated by the finite element analysis for the 30-ft side drop accident conditions. The finite element model is a 1-inch thick cross section of the MPC. It is a two-dimensional structural model that includes the fuel basket, support structures, and the MPC enclosure shell. For the side drop condition, the analysis considered both 0° and 45° circumferential impact orientations of the fuel basket. The analysis is performed for a deceleration load of 60 g and an internal pressure of 125 psi. Analyses are performed only for the hot ambient temperature condition since this is the bounding case for the pressure loads. The results are presented in the Appendix 2.AC of the SAR. The calculated stresses are compared with the allowable stresses and the corresponding safety factors for the MPC components are shown in Table 2.7.4 of the SAR.

2.6.1.2.2 Buckling Analysis

The overpack inner shell is supported by five-layers of intermediate steel shells. Based on the stresses and safety margins shown in Tables 2.7.5 and 2.7.6 of the SAR, it can be concluded that overpack inner shell buckling is unlikely under the 30-ft side drop condition. The most vulnerable fuel assemblies can withstand 63 g under the side drop orientation as shown in the UCID-21246, Lawrence Livermore National Laboratory Report. Since the impact load under the 30-ft side drop is calculated and also validated by scale-model testing to be about 45 g (design is based on 60 g to be conservative), fuel rod buckling is not likely. Therefore, buckling analysis under the 30-ft side drop condition only need be evaluated for the MPC and the fuel basket panels.

The buckling analysis of the MPC shell and the fuel basket for the side drop condition is performed by finite element analysis using ANSYS's large deformation capability. The finite element model is identical to the stress analysis, e.g., a 1-inch thick slice of the MPC including the shell, fuel basket, and basket supports. The analysis results showed that the collapse load of the MPC finite element model is greater than 1.5 times the design basis inertial load of 60 g. Therefore, buckling of the MPC is unlikely. The local buckling analysis is performed to show that the fuel basket panel will not buckle under the most unfavorable lateral loadings during the 30-ft side drop. The analysis follows the simplified plate buckling critical stress calculations in NUREG/CR-6322. The critical buckling stress for a long plate with two clamped edges is calculated to be 49.22 ksi which is 3.588 times greater than the maximum stress in the fuel panel by the finite element analysis. The safety factor against buckling, as recommended by NUREG/CR-6322, is 2.12 for stainless steel under accident loads. Since the ratio of the critical buckling stress to calculated stress is 3.588, buckling of the fuel panel will not occur.

2.6.1.3 Corner Drop

The corner drop is evaluated for a bottom center of gravity over the corner (CGOC) drop and the top CGOC drop for both hot and cold ambient temperatures. From the geometry of the overpack, with impact limiters in place, the angle of impact is 67.5° from the horizontal plane. Although the analytical and test results showed the g-loads are less than 40 g, the design basis 60 g impact load is conservatively used in the evaluation. The 60 g is applied vertically. Thus, a 55 g load is applied along the longitudinal axis of the overpack, and a 23 g load is applied perpendicular to the overpack longitudinal axis. The lateral load from the MPC, amplified by the 23 g, is applied to the overpack as shown in Figures 2.7.11 and 2.7.12. The longitudinal component of the load from the MPC, amplified by 55 g, is conservatively applied as uniform pressure over the inside surface of the closure plate as shown in Figure 2.7.8. The overpack weight is applied by imposing amplified gravitation accelerations in the axial and radial directions. The impact limiter reaction force is applied as pressure loads acting on the axial and circumferential directions to balance the inertia loads. Stress analysis results are presented in Appendix 2.AE for both hot and cold ambient conditions. Factors of safety are listed in Tables 2.7.5 and 2.7.6 of the SAR.

For failed fuel shipments, the MPC is the inner container of the package. Section 71.63 requires that the inner container meets the containment requirements specified in 10 CFR Part 71. The failed fuel is placed inside an MPC-68F canister for transport. Therefore, the applicant must demonstrate that the MPC-68F canister lid-to-shell welds will not fail during the 30-ft free drops. The applicant performed hand calculations to evaluate the stresses at the closure lid region for an oblique drop on the top end corner. The analysis assumed the lid to shell welds will withstand the impact loads from the content (both the fuel and basket), the lid, and the net pressure loads. For this limiting analysis condition, the analysis assumed the drop orientation to be 19.792° measured from the vertical axis; impact g-load to be 39 g which was the maximum g-load obtained from the 1/4-scale 30-ft drop test and analysis; and the actual weights of the damaged fuels (i.e., 550 lb x 68 assemblies). The applicant found that it is necessary to increase the MPC shell thickness from 0.5 inches to 1.0 inches at the closure lid region and also the size of the closure weld from 0.75 inches to 1.25 inches as shown in Figure 2.7.22 of the SAR. Based on the modified closure design, the maximum lid weld stress is shown to be 25.0 ksi which is smaller than the ASME allowable stress of 29.8 ksi. The factor of safety is approximately equal to 1.2. The maximum enclosure shell stress is 31.5 ksi which is well below the allowable stress of 72.0 ksi. Thus, it is concluded that the modified MPC-68F canister lid welds, as shown in Figure 2.7.22 of the SAR, will not fail under the hypothetical 30-ft drop accident condition.

2.6.1.4 Oblique Drops

The applicant performed impact analysis for drop angles at 30°, 45°, and 60° from the horizontal impact surface. The three oblique drops, however, produced approximately the same vertical decelerations as shown in Table 2.H.5 of Appendix 2.H of the SAR. It is thus concluded that the 1/4-scale model shallow angle drop test experienced the largest secondary impact and should subject to detailed stress analysis. It was noted that although the initial release of the test specimen was set at an angle of 15° from the horizontal, the actual impact occurred with the overpack longitudinal axis at an angle of approximately 7.2° from the horizontal surface. The measured peak deceleration at the slap-down impact limiter is 59 g, and this is also validated by analysis. The applicant stated, that based on numerical analysis that the peak deceleration from secondary impact is not sensitive to impact angles between 5° and 12° from the horizontal and decreases as the angle goes above 12°.

The overpack stress analysis was performed by using the half-symmetry, three-dimensional finite element model and ANSYS computer code. The 60 g design deceleration force is used in the analysis. The impact force, equal to 60 g, is applied to one end of the overpack as pressure loads with distributions similar to those of a side drop. The weight of the overpack and the MPC is amplified by the 60 g factor as shown in Figure 2.7.18. The analysis showed that the minimum factor of safety in the cask inner shell is 2.14 for primary membrane and 2.78 for local membrane, plus primary bending, in Table 2.7.5 of the SAR.

The package is near horizontal during the secondary impact. The MPC has been analyzed for 60 g side drops in the finite element analysis. The critical load (e.g., 60 g) for the MPC during secondary impact is identical to impact loads applied to the MPC in a side drop. Therefore, the stresses for the secondary dropare bounded by the side drop stresses, no new stress analysis for secondary impact of the MPC is needed.

2.6.2 Crush

Because the package has a mass greater than 500 kg (1,100 lb) and a density greater than water, this test is not applicable.

2.6.3 Puncture

The package was evaluated for the 40-inch puncture test under the hypothetical accident conditions. The puncture pin was conservatively assumed to impinge directly on (1) center of overpack top closure plate; (2) center of overpack bottom end plate; and (3) overpack outer shell, beneath the package CG. The impact limiters and the neutron shield tank provide protection to the overpack but are conservatively neglected in the evaluations.

The test, vent, and drain ports are small and located in solid steel forgings. They are protected by the impact limiters and a solid steel plug. Closure of the ports is accomplished by bolts with seals under the bolt head and tightened to a prescribed torque to maintain the seal. The steel plug is also equipped with an O-ring seal to retain any leakage. Thus, the ports are adequately protected from the puncture bar and no structural evaluation is performed.

An estimate of local puncture resistance is determined by Nelm's equation which is generally applicable for lead-backed shells. Nevertheless, it is used to check for potential penetrations for the HI-STAR 100 packages. Using an ultimate strength of 70,000 psi for the steel, Nelm's equation predicts a minimum thickness of 2.65 inches to preclude penetration by the puncture pin. The overpack has substantially more material thickness in the closure plate, the end plate, and the total thickness (e.g., inner shell plus intermediate shells) of the body. It can be concluded that the overpack will not be punctured through by the specified puncture tests.

The puncture stress analysis is performed by the finite element method using the same ANSYS model of the overpack. Analyses are performed for the side, top, and bottom puncture events as shown in Figures 2.1.12 through 2.1.14 of the SAR. All puncture analysis assumes that the puncture pin is perpendicular to the impact surface. It is assumed that the pin has a flow stress equal to 48,000 psi and that the total puncture force is almost 1.4×10^6 lb and is applied to the finite element model as pressure loads. The results for all three puncture events are presented in Appendix 2.AE of the SAR. The local surface plastic stresses are neglected for those nodes directly under the impact pin in the stress intensity comparison with allowable values. Tables 2.7.5 and 2.7.6 of the SAR summarize the safety factors for the overpack components for both hot and cold ambient conditions.

2.6.4 Thermal

The applicant performed a thermal analysis to show the package can withstand the 30-minute fire test. The containment boundary temperature will remain below 700°F, and the maximum temperature of the ferritic steel materials in the body of the overpack is well below the material melting point. The shielding material, however, will experience temperatures above its design limit for shielding effectiveness. Consequently, the shielding analysis will conservatively assume that all shielding material is lost in post-fire shielding analysis.

Differential thermal expansion due to the fire accident is considered in Appendix 2.G of the SAR. Bounding temperatures in the cask and MPC are used in the evaluation for conservatism. The analysis showed that there is no structural restraint of free expansion in the axial or radial directions between a hot basket and the enclosure shell or between the MPC and the overpack.

The stresses in the MPC and the overpack due to accident fire condition were determined by finite element analysis. In the fire accident case, only primary stresses are of interest to demonstrate continued containment. The accident cask internal pressure is taken as 125 psi. The resulting stresses due to the combinations of pressure, bolt pre-load, and fabrication loads are small. The stresses resulting from the accident fire thermal conditions are classified as secondary stresses and as such they are evaluated in accordance with RG 7.6 for low-cycle fatigue. It was shown that both the MPC and the overpack maximum stress intensity due to the fire event are much less than the limits of stress range for 10 cycles from the design fatigue curves given in the ASME Code. Thus, the thermal accident criteria of RG 7.6 are satisfied.

2.6.5 Immersion-Fissile Material

Section 71.73(c)(5) specifies that fissile material packages in those cases where water in leakage has not been assumed for criticality analysis, must be evaluated for immersion under a head of water at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected. The criticality analysis of the package assumes the package is flooded with water at optimum moderation. Consequently, the immersion test for fissile materials is not applicable. However, immersion under 3-ft of water is equivalent to an external pressure of only 1.3 psig which has no effects on the package. The analysis presented in the SAR has demonstrated that both the package containment boundary and the MPC enclosure vessel meet the stress and stress-intensity allowable values for normal transport and hypothetical accident conditions. Therefore, the water in-leakage assumption made in the criticality analysis is conservative.

2.6.6 Immersion-All Material

The package design external pressure under the hypothetical accident conditions is 300 psi. The stress in the overpack inner shell (i.e., the containment boundary) is calculated in Appendix 2.J of the SAR. The analysis conservatively neglects the presence of the supporting intermediate shells and the external pressure is directly applied to the inner shell alone. The inner shell stresses are well below the yield stress of the material. The stability of the inner shell is also evaluated using the methodology of ASME Code Case N-284. It was shown that the inner shell satisfied all interaction equations and that the minimum factor of safety against elastic buckling is approximately equal to 4.0. Therefore, an external pressure of 21.7 psi, equivalent to immersion under 50 ft of water, would have no significant effects on the package.

2.6.7 Special Requirement for Irradiated Nuclear Fuel Shipments

As described in the section above, the applicant performed a conservative stress and buckling analysis of the overpack inner shell, based on a design external pressure of 300 psi which envelops the required water pressure of 290 psi. Thus, the regulatory requirement has been met.

2.6.8 Internal Pressure Test

The HI-STAR 100 system maximum normal operating pressure (MNOP) is calculated for a postulated 100% fuel rod failure and the release of fill and fission gases from the rods. The internal pressure of 100 psi was used in the stress evaluations. 10 CFR 71.85 requires that the licensee test the containment system at an internal pressure which is, at least, 50% higher than MNOP to verify the capability of that system to maintain its structural integrity. The resulting stress intensities in the overpack body for the internal pressure load are very small. Therefore, it can also be concluded that the package containment will not yield under the 150% MNOP test pressure load and the resulting stresses will be within the allowable stress limits of the Design ASME Code.

2.7 Evaluation Findings

2.7.1 Description of Structural Design

The staff reviewed the package design descriptions and concluded that the contents of the application meet the requirements of 10 CFR 71.31. The staff reviewed the codes and standards used in the package design and found that they are acceptable.

2.7.2 Material Properties

To the maximum credible extent, there are no known chemical, galvanic, or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry or wet environmental conditions. The effects of radiation on materials are considered in the evaluation of the containment components for preventing brittle fracture.

2.7.3 Lifting and Tie-down Standards for All Packages

The staff reviewed the lifting and tie-down systems for the package and concluded that they meet the 10 CFR 71.45 standards.

2.7.4 General Considerations for Structural Evaluation of Packaging

The staff reviewed the packaging structural evaluations and concluded that they meet 10 CFR 71.35 requirements.

2.7.5 Normal Conditions of Transport

The staff reviewed the packaging structural performances under the normal conditions of transport (as described in Section 2.5) and concluded that there will be no substantial reduction in the effectiveness of the packaging, no loss or dispersal of radioactive contents, and no significant increase in external surface radiation levels.

2.7.6 Hypothetical Accident Conditions

The staff reviewed the package's structural performance under the hypothetical accident conditions (as described in Section 2.6) and concluded that the packaging has adequate structural integrity to satisfy the sub-criticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

2.7.7 Special Requirement for Irradiated Nuclear Fuel Shipments

The staff reviewed the containment structure and concluded that it will meet the 10 CFR 71.61 requirements for irradiated nuclear fuel shipments. In addition, the staff reviewed the MPC-68F Canister Closure System and concluded that it has adequate structural integrity to meet 10 CFR 71.63 requirements for inner containers.

2.7.8 Internal Pressure Test

The staff reviewed the containment structure and concluded that it will meet the 10 CFR 71.85(b) requirements for pressure tests without yielding.

2.8 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

3 THERMAL REVIEW

REVIEW OBJECTIVE

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

3.1.1 Description of the Thermal Design

3.1.1.1 Packaging Design Features

The design criteria for the HI-STAR 100 transportation cask have been formulated by the applicant to assure that public health and safety will be protected during the period that spent fuel is transported in the cask. These design criteria cover both the normal transport conditions and postulated accidents, such as a fire, that last a short time.

To provide adequate heat removal capability, the applicant designed the HI-STAR 100 system 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) minimal 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;

4) continuous metal heat conduction axially provided by the basket structure;

5) flexible aluminum heat conduction elements for heat transfer from the basket periphery to the MPC shell; and

6) high emissivity paint on the overpack exterior surface to maximize radiative heat transfer to the environment.

The staff verified that all methods of heat transfer internal and external to the MPC and overpack are appropriate. Drawings in Section 1.4 of the SAR along with the material properties in Tables 3.2.2 - 3.2.9 of Section 3 provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package.

3.1.1.2 Codes and Standards

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant.

3.1.1.3 Content Heat Load Specification

The design basis fuel to be transported in the HI-STAR 100 cask is described in Table 1.2.13 of the SAR for both the BWR and PWR fuels. Based on ORIGEN-2 computer analyses, for the PWR fuels (loaded in the MPC-24 basket), the initial maximum average decay heat per assembly is 706 W. In the MPC-24, this allows for a total package decay heat of 16.9 kW. For the BWR fuels (loaded in the MPC-68 basket), the initial maximum average decay heat per assembly is 238 W. The MPC-68 allows for a total package decay heat of 16.2 kW. Both fuels vary in maximum burnup and initial enrichments due to the optimization of the allowable total heat load per basket. The burnup and cooling time limits are also specified in Table 1.2.13. The axial profiles for the design basis fuels are listed in Table 1.2.15 of the SAR. The peak axial power in the BWR assemblies is a factor of 1.2 times the average power and for the PWR assemblies is a factor of 1.1 times the average power.

The staff reviewed and confirmed the design basis decay heat based on assembly type. The staff also verified that the design basis decay heats are the bounding decay heats through independent analysis providing reasonable assurance that the decay heats were determined properly.

The thermal loads are different for the normal transportation conditions than for the accident conditions, such as fire. The difference with the thermal loads occurs at the surface of the cask. The application of the surface thermal loads will be for a short time during an accident, while the surface thermal loads are applied continuously during normal transport conditions. The decay heat load during an accident will be the same as for the normal transportation condition at the time of the accident.

3.1.1.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, SAR Tables 3.4.10, 3.4.11, and 3.5.4, were verified to include the impact limiters, containment vessel, seals, shielding, and neutron absorbers and were consistent with the temperatures presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions. For the hypothetical accident conditions, the applicant accounted for the pre-fire, during-fire, and post-fire component temperatures. With the exception of the impact limiters, which are not critical to containment during the fire, all components remain below their material property limits. The temperatures and design temperature limit criteria for the package components were reviewed and found to be consistent throughout the SAR.

3.1.1.5 Summary Tables of Pressures in the Containment System

Summary tables of the pressure in the containment system under the normal conditions of transport and hypothetical accident conditions (SAR Tables 3.4.15 through 3.4.33) were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation SAR sections. These tables reported the MNOP for each of the canister baskets. The accident condition pressure was reported for the bounding canister.

3.1.2 Material Properties and Component Specifications

3.1.2.1 Material Properties

The package application provided material properties in the form of thermal conductivities, densities, and specific heats for all modeled components of the cask. Conservative thermal emissivities were used to model the radiative heat transfer to and away from the transportation cask. Materials that did not have a readily determinable thermal emissivity relied on a value of 0.9 for hypothetical accident conditions, bounding 10 CFR 71.73(c)(4) requirements. The thermal properties used for the analysis of the package were appropriate for the materials specified and for the conditions of the cask required by 10 CFR Part 71 during normal and accident conditions.

3.1.2.2 Technical Specifications of Components

References for the TS of pre-fabricated package components for O-rings, impact limiters, and the neutron shield were provided by the applicant. All components were shown to perform without fail under normal conditions with an ambient temperature of -40°F.

3.1.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified. The staff verified that the design basis fuel cladding temperature of 1058°F for accident conditions was reasonable and justified on the basis of the Pacific Northwest National Laboratory (PNNL) report, PNL-4835, which is a methodology accepted by the staff.

3.1.3 Thermal Evaluation Methods

3.1.3.1 Evaluation by Analyses

A detailed analytical model for thermal design of the HI-STAR 100 system was developed using the FLUENT finite volume Computational Fluid Dynamics code and the industry standard ANSYS modeling package. Transport of heat from the fuel assemblies to the outside environment is analyzed in terms of three interdependent thermal models. The first model considers transport of heat from the fuel assembly to the basket cell walls. The second model considers heat transport within an MPC cross section by conduction and radiation. The third model deals with the transmission of heat from the MPC exterior surface to the external environment through the overpack. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the vertical and top cover surfaces.

For normal conditions, the steady-state analysis produced a maximum cladding temperature of 701°F for the MPC-24. For the normal conditions of transport with the MPC-68, the steady-state analysis produced a maximum cladding temperature of 713°F. This temperature is below the limit of 1058°F. For accident conditions, the analysis revealed a maximum cladding temperature of 751 °F which occurred during the post-fire cooldown. This is below the limit of 1058°F.

3.1.3.2 Evaluation by Tests

The thermal acceptance test required before the first use of the cask is described in SAR Section 8.1.6.

3.1.3.3 Temperatures

See Safety Evaluation Report (SER) Section 3.1.6.3.

3.1.3.4 Pressures

See SER Section 3.1.6.3.

3.1.3.5 Thermal Stresses

Thermal stresses were evaluated in SAR Sections 2.6.1, 2.6.2, and 2.7.3, using the temperatures generated by the thermal evaluation. The applicant evaluated the effects of differential thermal expansion on gaps and the stresses resulting from component interactions. The applicant developed a two-dimensional finite element model of the fuel basket and the MPC enclosure shell to evaluate the effect of pressure, radial temperature gradients, and lateral deceleration induced inertia loads. The applicant also developed a three-dimensional model of the overpack to assess performance of the overpack under all load cases. The applicant considered thermal stresses for the hot-normal, cold-normal, and transient accident conditions.

The application included a thermal analysis to show the package can withstand the 30-minute hypothetical fire accident. The same analytical models used for the normal conditions of transport were used to assess package performance under the hypothetical accident conditions. The results of that analysis are discussed in Section 3.1.6.4 of the SER.

For the accident analysis, the applicant used the maximum internal pressures and containment boundary temperatures to calculate the thermal stresses and displacements. The gap analysis, supported by calculations performed in SAR Section 4.5, provided reasonable assurance that the cask response, including containment integrity, is acceptable. The resulting and allowable stresses for the closure bolts, seal surface, and containment boundary are presented in SAR Table 2.7-21 and are acceptable.

3.1.3.6 Confirmatory Analyses

The staff reviewed the models used by the applicant in the thermal analyses. The code inputs in the calculation packages were checked for consistency to confirm that the applicant used the appropriate material properties and boundary conditions where required. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the code model. The material properties presented in the SAR were reviewed to verify that they were appropriately referenced and used conservatively. A detailed model of the fuel regions and basket geometry was developed using the ANSYS finite element code to ensure that the SAR results were realistic and conservative. Independent homogenized thermal resistances were determined for the confirmatory calculation and employed in the model. The temperature distributions generated by the staff's model displayed agreement with those values determined by the applicant.

The staff further concluded that the design of the heat removal system of the HI-STAR 100 cask is in compliance with 10 CFR Part 71 and that the applicable design and acceptance criteria are been satisfied. The evaluation of the thermal system design provides reasonable assurance that the HI-STAR 100 will enable safe transportation of spent fuel. This finding is based on a review which considered the requirements of 10 CFR Part 71, appropriate RGs, applicable codes and standards, and accepted practices.

3.1.3.7 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and in analytical methods. Based on the results of a confirmatory bounding thermal finite-element analysis, the staff found reasonable assurance that the applicant used appropriate considerations throughout the application.

3.1.4 Evaluation of Accessible Surface Temperature

Under normal conditions, the package is designed and constructed such that the accessible surface temperature is 143°F for the MPC-24 (139°F for the MPC-68) with the design basis heat load and no solar insolation. This temperature complies with the 10 CFR 71.43(g) requirement, under the condition that the package will be shipped as exclusive-use.

3.1.5 Thermal Evaluation under Normal Conditions of Transport

3.1.5.1 Heat

Under normal conditions, all of the materials used remain below their respective failure temperatures. The applicant performed two steady-state calculations under normal conditions of transport. These calculations provided steady-state temperature distributions for the following combined boundary conditions: (1) an ambient temperature of 100°F, with solar insolation and maximum decay heat; (2) an ambient temperature of -40°F, with no solar insolation and maximum decay heat; and (3) an ambient temperature of -40°F, with no solar insolation and no decay heat.

3.1.5.2 Cold

With no decay heat and an ambient temperature of -40°F, the entire package will maintain a steady-state temperature of -40°F. Cask components, including the overpack system seals, would not be adversely affected by this low temperature.

3.1.6 Thermal Evaluation under Hypothetical Accident Conditions

3.1.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package under hypothetical accident conditions. The initial condition of the cask prior to the start of the fire accident is based on the bounding normal transport condition MPC basket temperature distribution. Thus the initial conditions of the cask are based on the 100°F ambient temperature and the solar insolation prescribed by 10 CFR 71.71(c)(1).

The thermal transient model used to determine the fire accident temperature response was developed on the FLUENT code. During the fire event, the impact limiters installed on both ends of the cask are assumed to be fully crushed. This is a reasonable assumption which results in an increased heat input to the overpack due to the higher thermal conductivity and reduced thickness of the crushed impact limiter. During the fire, the surface emissivity of the cask is assumed to be 0.9. After the 30-minute fire, the 100°F ambient temperature is restored and the damaged cask 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.

The peak temperatures of the key cask components due to the 30-minute fire with a maximum decay heat based on the MPC-24 are shown in the Table 3.1.4 of the SER. During the fire, an upper bound material thermal conductivity for the neutron shield is assumed to maximize heat input to the cask. During the post-fire cooldown phase, no credit is taken for conduction through the neutron shield. As a result, all of the fire accident temperatures were below the short-term design basis temperatures with the exception of the neutron shield material. However, the accident condition dose rate limits are shown to remain below the regulatory limit of a total dose of 1 rem/hr at one meter and are acceptable to the staff. 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.

3.1.6.2 Fire Test

See SER Section 3.1.6.1.

3.1.6.3 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are given in Table 3.1-1 below. The accident temperatures in the table reflect the peak temperature of a specified component from the time the fire was extinguished to the time the package reached steady-state conditions. For both normal and accident conditions, the inner cavity was assumed to be filled with helium.

Table 3.1-1
Maximum Calculated Temperatures (°F)

HI-STAR 100 Cask Component	Normal Conditions [°F]		Accident Conditions [°F]	
	MPC-24 Normal Conditions	MPC-68 Normal Conditions	During 30-Minute Fire	Post Fire Cooldown
Fuel Cladding	701	713	708	751
MPC Basket	667	697	N/A	N/A
MPC Outer Shell Surface	315	306	319	419
Overpack Inner Surface	291	282	292	328
Neutron Shield Inner Surface	271	264	604	604
Overpack Outer Surface	222	217	1348	1348
Impact Limiter Surface	143	139	983	983

Under normal conditions, all of the materials remain below their respective melting temperatures. For the accident conditions, all of the materials, with the exception of the aluminum honeycomb impact limiter and the neutron shield, remain below their respective melting temperatures. Although the impact limiter was shown to exceed its melting temperature, the applicant assumed the material did not melt during the fire. By doing this, the applicant maximized the amount of heat to have entered the package. If the material had been allowed to melt, this process would have resulted in a lower maximum fuel cladding temperature during the fire accident. Even though the neutron shield fails during the fire accident, the accident condition dose rate limits are shown to remain below the regulatory limit of a total dose of 1rem/hr at one meter which is accepted by the staff. 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 MNOP assuming that 100% of the fuel rods fail and that 30% of the gaseous fission products are available for release. The total gas volume considered the gaseous fission products, the helium fill gas, and the cavity back-fill gas. The gaseous fission products were based on a fuel burnup of 45,000 MWd/MTU.

The average gas temperature was calculated to be 483°F. Based on this gas temperature, the MNOP was determined to be 98.9 psig (for the bounding MPC-24 basket). The MNOP under hypothetical accident conditions is 114.7 psig, based on the average cavity gas temperature of 614°F.

3.1.6.4 Maximum Thermal Stresses

The applicant performed a thermal analysis to show the package can withstand the 30-minute hypothetical fire accident. The containment boundary temperature will remain below 700°F. and the maximum temperature of the ferritic steel materials in the body of the overpack is well below the material melting point. The shielding material, however, will experience temperature above its design limit for shielding effectiveness. Consequently, the shielding analysis will conservatively assume that all shielding material is lost in a post-fire shielding analysis.

Differential thermal expansion due to the fire accident is considered in Appendix 2.G of the SAR. Bounding temperatures in the cask and MPC are used in the evaluation for conservatism. The analysis showed that there is no structural restraint of free expansion in the axial or radial directions between a hot basket and the enclosure shell or between the MPC and the overpack.

The stresses in the MPC and the overpack due to accident fire condition were determined by finite element analysis. In the fire accident case, only primary stresses are of interest to demonstrate continued containment. The accident cask internal pressure is taken as 125 psi. The resulting stresses due to the combinations of pressure, bolt pre-load, and fabrication loads are small. The stresses resulting from the accident fire thermal conditions are classified as secondary stresses and as such they are evaluated in accordance with RG 7.6 for low-cycle fatigue. It was shown that both the MPC and the overpack maximum stress intensity due to the fire event are much less than the limits of stress range for 10 cycles from the design fatigue curves given in the ASME Code. Thus, the accident thermal requirements of RG 7.6 are satisfied.

3.2 EVALUATION FINDINGS

3.2.1 Description of the Thermal Design

The staff reviewed the package description and evaluation and found reasonable assurance that they satisfy the thermal requirements of 10 CFR Part 71.

3.2.2 Material Properties and Component Specifications

The staff reviewed the material properties and component specifications 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.

3.2.3 Thermal Evaluation Methods

The staff reviewed the methods used in the thermal evaluation and found reasonable assurance that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.

3.2.4 Evaluation of Accessible Surface Temperature

The staff reviewed the accessible surface temperatures of the package as it will be prepared for shipment and found reasonable assurance that they satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle.

3.2.5 Evaluation under Normal Conditions of Transport

The staff reviewed the package design, construction, and 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.

3.2.6 Evaluation under Hypothetical Accident Conditions

The staff reviewed the package design, construction, and preparations for shipment and 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.

3.3 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

4 CONTAINMENT REVIEW

Review Objective

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

4.1 Description of the Containment System

The staff reviewed the SAR's description of the containment boundary to ensure that all components are shown in the drawings and confirmed that the following information regarding the components is consistent with that presented in the Structural Evaluation and Thermal Evaluation sections of the SAR:

- materials of construction
- welds
- applicable codes and standards
- bolt torque required to maintain positive closure
- maximum allowable temperatures of components, including seals
- temperatures of components under normal conditions of transport and hypothetical accident conditions
- all containment boundary penetrations and their method of closure are described in detail
- no device may allow continuous venting
- any valve or similar device on the package must be protected against unauthorized operation and must be provided with an enclosure to retain any leakage
- cover plates and lids should be recessed or otherwise protected
- compliance with release limits do not depend on filters
- confirm that all closure devices can be leak tested
- confirm that the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package
- confirm that the spent nuclear fuel contents are fully described, including fuel type, fuel amount, percent enrichment, burnup, cool time, decay heat, etc.
- verify that the SAR identifies the constituents which comprise the releasable source term, including radioactive gases, volatiles, and powders
- verify that the maximum permissible leakage rate under normal transport conditions is converted into a reference air leakage rate under standard test conditions according to ANSI N14.5 and NUREG/CR-6487

4.2 Evaluations and Findings

Holtec performed detailed analyses to illustrate that the design basis leakage rate for the HI-STAR 100 multi-purpose cask will not be exceeded during normal and hypothetical accident conditions of transport. The design rating of the standard leakage rate for the primary (overpack) containment is identified in Table 4.2-1, below. The detailed regulatory analyses that support the acceptability of the design specification leak rates were performed in accordance with the guidance provided in NUREG/CR-6487 and the 1997 edition of ANSI N14.5.

Table 4.2-1
Design Specifications

Design Attribute	Primary (Overpack) per 10 CFR 71.51	Secondary (MPC-68F) per 10 CFR 71.63(b)
Standard Leakage Rate	4.3×10^{-6} std cm^3/s, Helium	5.0×10^{-6} std cm^3/s, Helium
Sensitivity-Standard Leakage Rate	2.15×10^{-6} std cm^3/s, Helium	2.5×10^{-6} std cm^3/s, Helium

The staff's review of Holtec's analyses was in accordance with the guidance provided in the Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-6887). The design of the MPC consists of a primary and secondary containment system boundary. The primary containment boundary incorporates an overpack inner shell, a bottom plate, a top flange, a top closure plate, closure bolts, overpack vent and drain port plugs, and their respective mechanical seals. The secondary containment system boundary is only required for the MPC-68F (contains fuel debris). No fuel debris will be loaded in the MPC-24 or MPC-68. The secondary containment system consists of an enclosure vessel including the MPC shell, the MPC bottom plate, the MPC lid, closure ring, and vent and drain port cover plates. The MPC-68F provides a separate inner container (per 10 CFR 71.63(b)) for the HI-STAR 100 System transporting fuel classified as fuel debris. The other MPC designs are not evaluated for secondary containment requirements.

The staff reviewed the analytic assumptions used by Holtec for its containment analyses. These assumptions were consistent with the guidance provide in NUREG/CR-6487 and ANSI N14.5. Table 4.2-2 summarizes some of the assumptions made in the analysis.

Table 4.2-2
Analytic Assumptions

Assumption	PWR		BWR	
	Normal	Accident	Normal	Accident
Fraction of crud that spalls, f_c	0.15	1.0	0.15	1.0
Crud surface activity (Ci/cm^2) (Assumed to be Cobalt-60)	$140x10^{-6}$	$140x10^{-6}$	$1254x10^{-6}$	$1254x10^{-6}$
Fraction of rods that develop cladding breach, f_B	0.03	1.0	0.03	1.0
Fraction of fines that are released, f_f	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$
Fraction of gases that are released due to cladding breach, f_g	0.3	0.3	0.3	0.3
Fraction of volatiles that are released due to a cladding breach, f_v	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-4}$	$2x10^{-4}$
Time of crud spallation and cladding breaches	Instantaneously after fuel loading and container closure	Instantaneously after fuel loading and container closure	Instantaneously after fuel loading and container closure	Instantaneously after fuel loading and container closure
Internal pressure assumption for leakage rate calculations	Set to MPC internal design pressure	Set to MPC internal design pressure	Set to MPC internal design pressure	Set to MPC internal design pressure
Average cavity temperature	Set to design-basis peak cladding temperature	Set to design-basis peak cladding temperature	Set to design-basis peak cladding temperature	Set to design-basis peak cladding temperature

In the analysis of the primary containment boundary, the MPC is assumed to rupture and the flow is assumed not to be limited by choked conditions.

Damaged Fuel Assemblies

For normal transport conditions of the MPC-68F (containing fuel debris), 100% of the rods in the fuel debris are assumed to be breached.

For the MPCs with damaged fuel and/or fuel debris, the analytic assumptions for normal conditions included a weighted average source term between the intact and defective fuel assemblies:

$$f_x = 0.03 \times \frac{\text{\# of intact assemblies}}{\text{total \# of assemblies}} + 1.0 \times \frac{\text{\# of damaged assemblies}}{\text{total \# of assemblies}}$$

This weighed average source term was applied to the calculation of fines, gases, and volatiles. The staff found this assumption acceptable.

The evaluations assumed up to four DFCs with specified fuel debris may be placed in the MPC-68F.

In the analysis of the MPC-68F separate inner container, the applicant used the bounding assumption that the primary containment fails.

By performing independent fuel decay calculations, the staff confirmed Holtec's assumptions on isotope inventory used in the containment analyses, as described in Table 4.2.2 of the SAR. The staff's results were comparable with those performed by Holtec.

In addition, the staff performed independent audit calculations for each MPC addressed in the SAR (e.g., MPC-24, MPC-68, and MPC-68F). The design and test conditions are identified in Table 4.2-3. The staff's independent evaluations confirmed Holtec's conclusions that the limiting allowable leakage rate (1.90E-05 cm^3/s at upstream pressure) occurred for MPC-68 under normal transport conditions.

The staff's analyses also confirmed that the leak rates were insensitive to the assumed capillary length (1.9 cm [0.75 inch weld size for the MPC-24 and MPC-68] versus 3.175 cm [1.25 inch weld size for the MPC-68F]). Holtec conservatively set the design rating standard leakage rate to 4.3E-06 std cm3/s helium for the primary containment (MPC overpack) and 5.0E-06 std cm3/s helium for the separate inner container (MPC-68F). The MPC-24 and MPC-68 are not required to provide a containment function during transport as only intact fuel assemblies are allowed to be loaded. For shipment of fuel debris, only the MPC-68F may be used. The MPC-68F provides a separate inner container in accordance with 10 CFR 71.63(b). The closure welds on all MPC's, however, will be leak tested to 5.0E-06 std cm^3/s helium in accordance with ANSI N14.5 and be subject to volumetric or multiple pass dye penetrant examination to assure weld integrity. The staff has reasonable assurance that all MPC's will maintain structural and leak integrity. These design limits conform to 10 CFR 71 requirements.

Table 4.2-3
Parameters for Normal, Hypothetical Accident and Standard Conditions

Parameter	Normal	Hypothetical Accident (Helium)	Standard Test Conditions (Helium)
P_u	7.8 ATM	9.5 ATM	Primary: 1.68 ATM
			Secondary: 2:0 ATM
P_d	1 ATM	1 ATM	1 ATM
T	673 K	1058 K	373 K
M	4 g/mole	4 g/mole	4 g/mole
μ	0.0341 cP	0.0397 cP	0.0231 cP
a	Primary: 0.25 cm	Primary: 0.25 cm	Primary: 0.25 cm
	Secondary: 1.9 cm [1]	Secondary: 1.9 cm[1]	Secondary: 1.9 cm[1]

[1]Only used for MPC-68F, where damaged fuel assemblies require a separate inner container, in accordance with 10 CFR 71.63(b). A value of 1.9 cm (0.75 inches, the size of the closure weld on the MPC-24 and MPC-68) or 3.175 cm (1.25 inches, the size of the closure weld on the MPC-68F) has a negligible impact on the calculated leak rate.

4.3 Evaluation Findings

The staff reviewed the description and evaluation of the containment system and concluded that: (1) the SAR identifies established codes and standards for the containment system; (2) the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package; (3) the package is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction; (4) a package valve or similar device, if present, is protected against unauthorized operation and is provided with an enclosure to retain any leakage; (5) a package designed for the transport of damaged spent nuclear fuel includes packaging of the damaged fuel in a separate inner container that meets the requirements of 10 CFR 71.63(c).

4.4 Containment Under Normal Conditions of Transport

The staff reviewed the evaluation of the containment system under normal conditions of transport and concluded that the package is designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71 (normal conditions of transport) the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system.

4.5 Containment Under Hypothetical Accident Conditions

The staff reviewed the evaluation of the containment system under hypothetical accident conditions and concluded that the package satisfies the containment requirements of

In summary, the staff reviewed the Containment Evaluation section of the SAR and concluded that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71 and that the package meets the containment criteria of ANSI N14.5.

4.6 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

5 SHIELDING REVIEW

Shielding Review Objective

The objective of this review is to verify that the package design satisfies the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.1 Description of Shielding Design

5.1.1 Packaging Design Features

The HI-STAR 100 is designed to provide both gamma and neutron shielding during normal and hypothetical accident conditions specified in 10 CFR Part 71. The principal components of the radial gamma shielding are the 0.5-inch thick steel MPC shell, the 2.5-inch thick steel inner shell of the overpack, and the five intermediate steel shells welded to the inner shell with an equivalent thickness of 6 inches. Gamma shielding at the bottom of the cask is provided by the 2.5-inch thick steel MPC baseplate and the 6-inch thick steel overpack bottom plate. The gamma shielding at the top of the cask consists of a 6-inch thick steel overpack closure plate and a 9.5-inch thick steel MPC lid for the MPC-24 or a 10-inch thick steel lid for the MPC-68 and MPC-68F.

In addition to the steel components discussed above, neutron shielding on the side of the package is provided by a solid borated polyester resin poured into enclosed radial channels that surround the cask body. The thickness of the radial neutron shield is approximately 4.3 inches. The neutron shield material is Holtite-A (NS-4-FR) with a density of 1.68 g/cm^3, a nominal hydrogen content of 6.0 weight percent (wt%) and a minimum B_4C content of 1.0 wt%. Neutron shielding is also provided on the top and bottom of the package by a 2.5-inch layer of Holtite-A poured inside each impact limiter.

5.1.2 Summary Table of Maximum Radiation Levels

The summary tables of maximum radiation levels for both normal and hypothetical accident conditions outside the package for the MPC-24 and MPC-68 configurations show values within the regulatory limits for an exclusive-use shipment. The maximum radiation levels are shown in Tables 5.4.3-1 and 5.4.3-2.

5.1.3 Staff Evaluation

The staff reviewed the General Information chapter, the Shielding chapter, and the Design Drawings in the application for completeness of information and consistency. The information, parameters, and dimensions provided were sufficient to perform a review. However, some general descriptions of the design basis fuels for shielding that are listed in Chapter 1 were inconsistent with the parameters specified in Chapter 5. The staff only reviewed and evaluated the fuel parameters specified in the Chapter 5 shielding analysis. The staff has informed the applicant of the discrepancy in Chapter 1. It will be corrected in the next update of the SAR. All other information was consistent throughout the chapters and drawings. As appropriate, codes and standards are identified and used.

5.2 Source Specifications

5.2.1 Gamma and Neutron Source

Gamma and neutron source terms are generated with the SAS2H and the ORIGEN-S modules of SCALE 4.3, using the 44-group cross-section library (see Reference 1). Four design basis fuel types are identified for the PWR and BWR Zircaloy and stainless-steel-clad fuels loaded in the MPC-24 and MPC-68 package configurations. Source terms for each design basis fuel type are calculated for various specific burnup, cooling time, and enrichment combinations. Source terms include the gamma and neutron radiation from the spent fuel pellets and the activated Cobalt-60 in fuel grid-spacers and discrete hardware regions on the top and bottom of the fuel assemblies. The source term analyses assume initial Cobalt-59 impurities of 0.08 wt% in steel and 0.47 wt% in Inconel. An analysis demonstrates that fuel gammas with energies from 0.45 Mev to 3.0 Mev comprise more than 99% of the external gamma dose. Therefore, only this gamma energy range is applied in the gamma shielding evaluation.

The Babcock & Wilcox 15x15 assembly is identified as the design basis fuel for Zircaloy-clad PWR fuel (MPC-24). The burnup and cooling times range from 24,500 MWD/MTU for 10 years to 37,500 MWD/MTU for 15 years. The source terms assume a conservative initial uranium loading that is 20 kgU higher than the maximum PWR uranium loading specified for the package contents. A second set of values is determined assuming only Zircaloy grid spacers in the active fuel region. The burnup and cooling times range from 24,500 MWD/MTU for 7 years to 34,500 MWD/MTU for 15 years.

The Westinghouse 15x15 stainless-steel-clad fuel assembly is identified as the design basis fuel for the stainless-steel-clad PWR fuel (MPC-24). The two burnup and cooling times are 30,000 MWD/MTU for 9 years and 40,000 MWD/MTU for 24 years.

The General Electric 7x7 assembly is identified as the design basis fuel for the Zircaloy-clad BWR fuel (MPC-68). The burnup and cooling times range from 24,500 MWD/MTU for 8 years to 39,500 MWD/MTU for 15 years. The fuel assembly is also shown to have bounding source characteristics for the Zircaloy-clad 6x6 MOX fuel, the Zircaloy-clad damaged BWR fuel, and the Zircaloy-clad BWR fuel debris (MPC-68F) for normal and hypothetical accident conditions. The burnup and cooling time for these are 30,000 MWD/MTU for 18 years.

The Allis Chalmers 10x10 assembly is identified as the design basis fuel for the stainless-steel-clad BWR fuel (MPC-68). The burnup and cooling time are 22,500 MWD/MTU for 16 years.

5.2.2 Staff Evaluation

The staff performed an independent confirmatory analysis of selected gamma and neutron source terms for the Zircaloy-clad and stainless-steel-clad PWR and BWR fuels. The staff used SAS2H and ORIGEN-S modules of SCALE 4.4 (see Reference 1). The staff found acceptable agreement with the application's reported values and has reasonable assurance that the source terms for the PWR and BWR fuels are adequate for the shielding analysis. The source term fuel parameters of maximum burnup, minimum cooling time, minimum enrichment, and maximum initial uranium loading (with maximum tolerance) are included as conditions of the license for each fuel type.

5.3 Model Specifications

The shielding models for normal and hypothetical accident conditions consist of three-dimensional representations of the package based on HI-STAR design drawings. Separate models were developed for the MPC-24 and MPC-68 configurations. The shielding models also include explicit representation of the streaming paths through the package's radial fins and pocket trunnions. The models assume the package is transported on a flat-bed style vehicle with a width equal to the diameter of the impact limiters (7.1 ft), and a distance of 6 ft between the bottom of the package and the end of the transport vehicle.

5.3.1 Configuration of Source and Shielding

The fuel is assumed to maintain a fixed axial position during normal and hypothetical accident conditions with the use of fuel spacers in the MPC cavity. The axial distribution of the gamma source is assumed to linearly follow the relative axial burnup profiles for PWR and BWR fuels. The axial distribution of the neutron source is assumed to follow the relative axial burnup profiles raised to the power of 4.2.

5.3.2 Material Properties

The composition and densities of the materials used in the shielding analysis are the same or similar to shielding materials specified in the design drawings. Applicant analyses demonstrate that all shielding components used in the HI-STAR 100 remain below the design temperatures during normal conditions. The neutron shield material (Holtite-A) is expected to experience minor water evaporation during normal conditions. Therefore, the neutron shield density is reduced in the shielding models to account for potential reduction in hydrogen weight. The result of hypothetical accident conditions for shielding assumes complete loss of the radial neutron shield and the impact limiters. Therefore, the radial neutron shield and the impact limiters are not included in the hypothetical accident shielding analysis.

5.3.3 Staff Evaluation

The staff evaluated the shielding models for normal and hypothetical accident conditions and found them to be consistent with the drawings and appropriate or bounding for the analyses presented in the structural and thermal analyses. The staff evaluation of the structural integrity and thermal performance of the shielding components are presented in Sections 2 and 3 of the SER.

The occupied space dose requirements of 10 CFR 71.47(b)(4) is not evaluated for a specific vehicle configuration. Conformance with this requirement will be the responsibility of the package-user for each shipment and associated transport vehicle. In addition, the low dose rates calculated at the top of the package provide reasonable assurance that the package-user will be able to demonstrate compliance with 10 CFR 71.47(b)(4).

5.4 Evaluation

5.4.1 Methods

The shielding analyses of the HI-STAR 100 are performed with MCNP-4A (see Reference 2), a three-dimensional Monte Carlo transport code. The individual cross-section libraries used for each nuclide are based on ENDF/B-V cross-section data, and the flux-to-dose-rate conversion factors are based on ANSI/ANS Standard 6.1.1-1977. Subcritical multiplication of the neutron flux and secondary gamma rays produced by neutron capture are included in the analyses. Azimuthal dose peaking on the side of the package shield due to basket geometry is also considered in the analyses.

Dose values are calculated for both the MPC-24 and MPC-68 configurations containing both Zircaloy and stainless-steel-clad fuel. Separate dose values for the MPC-24 assuming no Inconnel in the active fuel regions is also calculated. Analyzed dose locations include the top, bottom, side, accessible-flange, and impact limiter areas on the package. Streaming dose rates through pocket trunnions and radial fins are also evaluated.

5.4.2 Key Input and Output Data

Sample input decks for the SAS2H and MCNP-4A models are provided in the application.

5.4.3 Radiation Levels

External dose rates are calculated on the exterior surfaces of the package and at locations 2 meters from the outer lateral surfaces of the assumed transport vehicle during normal conditions of transport. External dose rates are calculated at 1 meter from the package surface after hypothetical accident conditions. The maximum dose rate values calculated for the MPC-24 and MPC-68 configurations are shown in Tables 5.4.3-1 and 5.4.3-2.

Table 5.4.3-1
Maximum External Dose Rates for the HI-STAR MPC-24 Configuration

Normal Conditions of Transport			
	External Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.51	34.89	106.11
Neutron	4.06	156.37	12.35
Total	4.57	191.26	118.46
10 CFR 71.47(b)(1) Limit	200	200	200
	Two Meters from Vehicle Outer Surface		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.04	8.94	9.10
Neutron	0.43	0.78	0.29
Total	0.47	9.72	9.39
10 CFR 71.47(b)(3) Limit	10	10	10
Hypothetical Accident Conditions			
	One Meter from Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.19	25.28	582.54
Neutron	17.60	471.80	47.49
Total	17.79	497.08	630.03
10 CFR 71.51(a)(2) Limit	1000	1000	1000

Table 5.4.3-2
Maximum External Dose Rates for the HI-STAR MPC-68 Configuration

Normal Conditions of Transport			
	External Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.24	109.36	93.05
Neutron	1.75	16.43	7.98
Total	1.99	125.79	101.03
10 CFR 71.47(b)(1) Limit	200	200	200
	Two Meters from Vehicle Outer Surface		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.02	7.59	8.07
Neutron	0.19	1.73	0.31
Total	0.21	9.32	8.38
10 CFR 71.47(b)(3) Limit	10	10	10
Hypothetical Accident Conditions			
	One Meter from Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma	0.11	26.64	534.51
Neutron	7.24	458.55	29.83
Total	7.35	485.19	564.34
10 CFR 71.51(a)(2) Limit	1000	1000	1000

5.4.4 Staff Evaluation

The staff performed a confirmatory analysis of selected cases based on the design features, source terms, and model specifications as discussed above and in SER Sections 5.1 through 5.3. The analysis included dose rate values reported for both the MPC-24 and MPC-68 during normal and hypothetical accident conditions. The calculations were performed with MCBEND-9D, a three-dimensional Monte Carlo transport program with continuous point cross-section data from UKNDL and JEF2.2 cross-section libraries (see Reference 3). The staff found acceptable agreement with the selected cases and found all radiation levels reported in the application to be within regulatory limits.

Based on (1) the review of the information and analyses reported by the applicant, (2) its own confirmatory calculations, and (3) the limits placed on fuel design, maximum burnup, minimum cooling time, minimum enrichment, and maximum initial uranium loading, the staff believes there is reasonable assurance that the package, with approved contents, will meet the radiation shielding requirements in 10 CFR Part 71.

5.5 Evaluation Findings

5.5.1 Description of the Shielding Design

The staff reviewed the description of the packaging design and found reasonable assurance that it provides an adequate basis for the shielding evaluation.

5.5.2 Source Specification

The staff reviewed the source specifications used in the shielding evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package under 10 CFR Part 71 radiation shielding requirements.

5.5.3 Model Specification

The staff reviewed the models used in the shielding evaluation and found reasonable assurance that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package shielding design.

5.5.4 Evaluation

The staff reviewed the external radiation levels calculated for the package and vehicle as it will be prepared for shipment, and found reasonable assurance that they satisfy 10 CFR 71.47(b) for packages transported by exclusive-use vehicle.

The staff reviewed the package design and contents specified for shipment and found reasonable assurance that the external radiation levels will not significantly increase during normal conditions of transport consistent with the tests specified in 10 CFR 71.71.

The staff reviewed the package design and contents specified for shipment and found reasonable assurance that the external radiation level at 1 meter from the external surface of the package will not exceed 10 mSv/hr (1 rem/hr) during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73.

5.6 References

1. L.M. Petrie, et al. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-4, Revision 4, 1995.

2. J.F. Briesmeister, Ed. "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A," Los Alamos National Laboratory, LA-12625-M, 1993.

3. AEA Technology, "MCBEND - A Monte Carlo Program for General Radiation Transport Solutions, User Guide for Version 9," 1997.

4. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

5. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

6 CRITICALITY REVIEW

Review Objective

The objective of this review is to verify that the package design satisfies the criticality safety requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

The applicant performed a criticality analysis to show that the package remains subcritical under normal conditions of transport and hypothetical accident conditions. The analysis shows that the package meets the requirements of 10 CFR Part 71.

The staff's criticality review and confirmatory analysis are based on the information provided in Revision 8 of the SAR. The staff's criticality review is summarized below.

6.1 Description of the Criticality Design

6.1.1 Packaging Design Features

Criticality control is provided by the geometry of the fuel basket and the use of permanent neutron-absorbing panels (Boral). In all MPC designs, the fuel assemblies are placed in baskets with square fuel cells and Boral panels fixed to the fuel cell walls. In the MPC-24 basket, the primary design features that ensure subcriticality are the minimum size of the flux trap (1.09 inches) and the minimum ^{10}B content of the Boral panels (0.0267 g/cm^2). In the MPC-68 and MPC-68F baskets, the primary design features that ensure subcriticality are the minimum pitch of the fuel cells (6.43 inches) and the minimum ^{10}B content of the Boral panels (0.0372 g/cm^2 in MPC-68 and 0.01 g/cm^2 in the MPC-68F). In addition, the fuel cells have semicircular cutouts at the bottom to allow the volume inside and outside the fuel cells to flood and drain at the same rate.

Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the test specified in 10 CFR 71.71 and 71.73.

The staff reviewed Chapters 1 and 6 of the SAR for completeness of information and consistency. The design features important to criticality safety are clearly identified and adequately described. The engineering drawings and other information in these chapters are sufficiently detailed to support an in-depth criticality evaluation by the staff.

6.1.2 Codes and Standards

The criticality evaluation is consistent with the appropriate codes and standards for nuclear criticality safety. The criticality evaluation is also consistent with the recommendations provided in NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages."

6.1.3 Summary Table of Criticality Evaluations

Table 6.1.4 of the SAR contains a summary of the final criticality analysis results. This table addresses results for a single package and for arrays of damaged and undamaged packages, as required by 10 CFR 71.55 and 71.59. The summary table illustrates that the package meets the criticality criteria of 10 CFR Part 71 and that the package would remain subcritical under normal conditions of transport and hypothetical accident conditions.

The maximum k_{eff} for each MPC design, as calculated by the applicant, are summarized in the table below. The results of the staff's confirmatory calculations are in close agreement with the applicant's results.

Table 6.1.3-1
Maximum k_{eff} for each MPC Design
(MCNP4a Code Results, with Bias and Uncertainty)

Condition	MPC-24	MPC-68	MPC-68F
Single Package, Flooded 10 CFR 71.55(b), (d), and (e)	0.9478	0.9457	0.8033
Infinite Array of Undamaged Packages, Dry 10 CFR 71.59(a)(1)	0.3924	0.3665	0.3034
Infinite Array of Damaged Packages, Flooded 10 CFR 71.59(a)(2)	0.9473	0.9447	0.8026

6.1.4 Transport Index

Results of the criticality analysis show that any number of undamaged or damaged packages will remain subcritical in any arrangement with close full-water reflection and optimum interspersed hydrogenous moderation. Therefore, per 10 CFR 71.59(b), the transport index for the package is 0.

6.2 Spent Nuclear Fuel Contents

The fuel assemblies that can be transported in the HI-STAR 100 package must fit into 1 of 36 PWR or BWR fuel assembly classes defined by the applicant. The approved fuel assembly classes, and their specifications, are listed in Tables 1.2.10 and 1.2.11 of the SAR and in the CoC.

The MPC-24 contents are intact PWR fuel assemblies with maximum initial enrichments of 4.0 to 4.6 wt% ^{235}U. The MPC-68 contents are intact or damaged BWR fuel assemblies with maximum planar average initial enrichments of 2.7 to 4.2 wt% ^{235}U. The MPC-68F contents are intact or damaged BWR fuel assemblies and BWR fuel debris from the 6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A classes with a maximum planar average initial enrichment of 2.7 wt% ^{235}U. The BWR fuel assemblies and fuel debris may be shipped with or without the channels. DFAs and fuel debris must be placed in DFCs, which are designed to confine gross fuel particulates to a known, subcritical geometry.

The staff reviewed the description of the spent nuclear fuel contents and agrees that all specifications relevant to the criticality analysis have been provided. The staff also verified that the specifications used in the criticality evaluation are consistent with or bound those given in Tables 1.2.10 and 1.2.11 and in the CoC.

6.3 General Considerations for Evaluations

6.3.1 Model Configuration

The applicant used three-dimensional calculational models in its criticality analysis. Sketches of the models are given in Sections 6.3 and 6.4 of the SAR. The models are based on the engineering drawings in Section 1.4 of the SAR and take into consideration the dimensional worst-case tolerance values. The hypothetical accident conditions do not affect the design of the cask from a criticality standpoint. Therefore, the calculational models for the normal and accident conditions are the same.

The calculational models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit)
- 75% credit for the ^{10}B loading in the Boral panels (only 67% credit for the MPC-68F)
- absence of the Holtite neutron shield
- the Boral panels are only as long as the fuel assembly active length, 150 inches maximum (the engineering drawings specify them to be 156-inches long)
- the Boral panels located on the periphery of the MPC-24 are only 5-inches wide (the engineering drawings specify 12 of the peripheral panels to be 6.25 inches and all other panels to be 7.5-inches wide)
- flooding of the fuel rod gap regions with pure water, even the intact fuel assemblies
- a maximum planar average enrichment of 3.0 wt% ^{235}U for all fuel assemblies in the MPC-68F (the CoC permits a maximum planar average enrichment of 2.7 wt% ^{235}U)
- no credit for burnable absorbers (e.g., gadolinia in BWR fuel)

The fuel assemblies were modeled explicitly. For BWR fuel assemblies, the water channels were appropriately included in the model. The models for DFAs and fuel debris considered lost or missing fuel rods, collapsed fuel assemblies, and powdered fuel.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Chapter 1, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances were incorporated into the calculational models.

6.3.2 Material Properties

The composition and densities of the materials considered in the calculational models are provided in Table 6.3.4 of SAR. The staff reviewed these material properties and found them to be reasonable. A sampling of the mass and atom densities was checked and found to be correct. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Within the applicant's criticality analysis, 75% credit is taken for the ^{10}B content in the Boral panels. The minimum required ^{10}B content is verified through the acceptance testing program described in Section 8.1.5.3. The staff reviewed the neutron absorber acceptance test and found it acceptable based, in part, on the fact that the criticality analysis took only 75% credit for the ^{10}B. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant's principal criticality analysis code was MCNP4a, a three-dimensional, continuous-energy, Monte Carlo N-Particle code. The MCNP4a calculations used the continuous-energy cross-section data distributed with the code. This cross-section data is based on the ENDF/B-V cross-section library. The applicant also performed independent verification of its MCNP4a calculations using the KENO-Va code in the SCALE 4.3 system. The KENO-Va calculations used the 238-group cross-section library.

CASMO-3, a two-dimensional transport theory code, was used to assess the incremental reactivity effects of manufacturing tolerances. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances. Based on the results of the CASMO-3 calculations, the worst-case combination of manufacturing tolerances was determined and incorporated into the three-dimensional MCNP4a and KENO-Va models.

The staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

6.3.4 Demonstration of Maximum Reactivity

Most Reactive Fuel Assemblies and Fuel Assembly Parameters. As stated in Section 6.2 above, the classes of approved fuel assemblies and their specifications are listed in Tables 1.2.10 and 1.2.11 of the SAR. In those tables, the fuel specifications that are important to criticality safety are:

- maximum initial enrichment (PWR)
- maximum planar average initial enrichment (BWR)
- number of fuel rods, including number of partial length rods (BWR)
- minimum clad outer diameter
- maximum clad inner diameter
- maximum pellet diameter

- fuel rod pitch
- maximum active fuel length
- number of guide tubes (PWR)
- number of water rods (BWR)
- minimum guide tube thickness (PWR)
- minimum water rod thickness (BWR)
- maximum channel thickness (BWR)

The parameters listed above represent the limiting or bounding parameters for the fuel assemblies. In Tables 6.2.3 to 6.2.38 of the SAR, the applicant presented results showing that a fuel assembly having these actual specifications would be the most reactive assembly or bounding assembly in that class. Based on this information, the staff agrees that the most reactive fuel assemblies have been evaluated.

The applicant's criticality analysis for BWR fuel assumed that the entire fuel assembly was at the maximum planar average initial enrichment. The maximum planar average enrichment is the simple average of the distributed fuel rod enrichments within a given axial plane of the assembly lattice. The applicant performed calculations to show that this is more conservative than explicitly modeling the assembly's pin-by-pin enrichments. The calculations considered real assembly designs and hypothetical assembly designs. These calculations are presented in Appendix 6.B of the SAR. Based on the results of these calculations and the information in Appendix 6.B of the SAR, the staff agrees that using the maximum planar average initial enrichment in the criticality analyses of BWR fuel assemblies is appropriate.

Optimum Moderation. The applicant considered various levels of external (interspersed) and internal moderation to determine the most reactive moderating conditions (optimum moderation). For both a single package and an array of packages, and for all MPC designs, the applicant determined that optimum internal moderation occurs when the cask is fully flooded with 100% density water. The applicant also determined that the reactivity of a fully flooded single package or an array of packages is insensitive to degree of interspersed moderation.

Normally, preferential or uneven flooding within the MPC is not a concern because the MPC baskets are designed to allow the volume inside and outside the fuel cells to flood and drain at the same rate. For damaged fuel in DFCs, however, uneven draining may be possible. The drainage holes on the DFCs are covered with 250 mesh debris screens. The staff has learned that the water surface tension in the screen may be capable of supporting water. Thus, the DFCs may hold water or may not drain at the same rate as the rest of the MPC cavity. The applicant did not consider a case in which the DFCs retained water while the rest of the MPC cavity was drained. However, the staff performed independent analysis considering this scenario. In this analysis, the staff assumed that the entire internal volumes of the DFCs were filled with water while the rest of the MPC cavity was dry. This analysis resulted in a k_{eff} of approximately 0.9 for the most reactive damage fuel assembly class (the 6x6 C). In comparison, the applicant's analysis show a k_{eff} of approximately 0.8 for the 6x6 C assembly when the entire MPC cavity is fully and evenly flooded. Although there is a significant increase in k_{eff}, it still remains well below 0.95. Thus, the staff concludes that even if preferential or uneven flooding is possible with the DFCs, it does not present a criticality concern because the fuel assemblies that may be placed in the DFCs are limited to low reactivity fuel (6x6 A, 6x6 B, 6x6 C, 7x7 A, and 8x8 A with a maximum planar average initial enrichment of 2.7 wt% ^{235}U).

Based on the results of the applicant's evaluation and the staff's independent confirmatory calculations, the staff concludes that the most reactive moderating conditions have been considered.

Most Reactive Packaging Configuration. To determine the most reactive basket dimensions, considering manufacturing tolerances, the applicant performed two-dimensional CASMO-3 and three-dimensional MCNP4a calculations. These calculations were used to determine the reactivity effect of manufacturing tolerances and the worst-case combination of basket dimensions. Based on the results of these calculations, the MPC-24 was modeled using the nominal fuel cell pitch (10.777 inches), the minimum box inner dimension (8.81 inches), the nominal box wall thickness (5/16 inch), and the maximum flux trap size (1.09 inches). The MPC-68 was modeled using the minimum fuel cell pitch (6.43 inches), the minimum box inner dimension (5.993 inches), and the nominal box wall thickness (1/4 inch). Based on the results of the applicant's parametric calculations, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances has been considered.

6.3.5 Confirmatory Analyses

For its confirmatory analysis, the staff independently modeled the cask using the engineering drawings and bills of material presented in Section 1.4 of the SAR. Specifically, the staff used Drawing Numbers 5014-1395, 5014-1396, 5014-1397, 5014-1401, 5014-1402, and 5014-1784 and Bill of Material Numbers BM-1478, BM-1479, and BM-1819. The staff's fuel assembly models were based on the fuel assembly parameters given in Chapters 1 and 6 of the SAR. The staff found its models of the cask and contents to be compatible with those of the applicant.

The staff used the CSAS/KENO-Va codes and the 44-group cross-section library in the SCALE 4.3 system in its confirmatory analysis. The staff performed criticality calculations for the most reactive fuel assembly class for each fuel assembly array size. The results of the staff's confirmatory calculations are in close agreement with the applicant's results for the corresponding fuel assembly class.

Based on the applicant's criticality evaluation and the staff's confirmatory analysis, the staff has reasonable assurance that the HI-STAR 100 package will remain subcritical, with an adequate safety margin, under normal conditions of transport and hypothetical accident conditions.

6.4 Single Package Evaluation

The single package evaluation is discussed in Section 6.4.2.1.1 of the SAR. Using the package model described in Section 6.3.1 above, the applicant performed calculations which show that a single package remains subcritical when optimally moderated (fully flooded with 100% density water) and fully reflected by water. The applicant also performed additional calculations to demonstrate that an optimally moderated, single package is subcritical when just the containment system (the 2.5-inch thick inner shell) is fully reflected by water.

Based on the results of the applicant's single package evaluation and the staff's own confirmatory analysis, the staff has reasonable assurance that the HI-STAR 100 package satisfies the requirements of 10 CFR 71.55(b), (d), and (e).

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

See SER Section 6.6 below.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The analysis for an array of packages is discussed in Section 6.4.2.1.2 of the SAR. The transport index for criticality control is 0; therefore, an infinite array of packages was considered in the array evaluation. Using the package model described in Section 6.3.1 above, the applicant performed calculations for an infinite square-pitched array and an infinite, triangular-pitched array. As discussed in Section 6.3.1, the package models for normal and hypothetical accident conditions are the same; therefore, separate array models for undamaged and damaged packages were not necessary. Also, the array of packages under normal conditions of transport would have no internal or interspersed moderation and would, therefore, be bounded by the array for hypothetical accident conditions.

The applicant's array calculations show that an infinite array of dry packages, with nothing between the packages, is subcritical. The array calculations also show that an infinite array of packages is subcritical with optimum internal and interspersed moderation. Optimum internal moderation occurs with the packages fully flooded with 100% density water. Interspersed moderation had no significant impact on reactivity because the thick wall of the overpack precludes neutron coupling between packages.

Based on the results of the applicant's array evaluation and the staff's own confirmatory analysis, the staff has reasonable assurance that the HI-STAR 100 package satisfies the requirements of 10 CFR 71.59(a)(1), 71.59(a)(2), and 71.59(b).

6.7 Benchmark Evaluations

6.7.1 Experiments and Applicability

The benchmark evaluation is presented in Appendix 6.A of the SAR. The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of variables in the HI-STAR 100 design. The three most important parameters are the fuel enrichment, the ^{10}B loading of the neutron absorbers, and the fuel cell spacing (MPC-68) or flux trap size (MPC-24). Parameters such as reflector material and spacing, fuel pellet diameter and fuel rod pitch, soluble boron concentration, and MOX fuel, have a smaller effect but were also considered in selecting the critical experiments.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design.

6.7.2 Bias Determination

Results of the benchmark calculations show that there are no trends in the bias. The benchmark analysis yielded the following calculational biases: 0.0021 ± 0.0006 for MCNP4a and 0.0036 ± 0.0009 for KENO-Va. These biases were determined by truncating to 1.000 any calculated k_{eff} that exceed unity. The uncertainty associated with each bias has been multiplied by the one-sided K-factor for 95% probability at the 95% confidence level (~2.05 for the number of cases analyzed).

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

The staff found the applicant's method for determining the calculational bias acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.8 Appendix

All supportive information has been provided in the SAR, primarily in Chapters 1 and 6.

6.9 Evaluation Findings

Based on the information and criticality evaluation presented in the SAR and the staff's confirmatory analysis, the staff concluded that the HI-STAR 100 package satisfies the criticality safety requirements of 10 CFR Part 71.

6.9.1 Description of Criticality Design

The staff reviewed the description of the packaging design and concluded that it provides an adequate basis for the criticality evaluation.

The staff reviewed the summary information of the criticality design and concluded that it indicates the package is in compliance with the requirements of 10 CFR Part 71.

6.9.2 Spent Nuclear Fuel Contents

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

6.9.3 General Considerations for Evaluations

The staff reviewed the criticality description and evaluation of the package and concluded that it addresses the criticality safety requirements of 10 CFR Part 71.

6.9.4 Single Package Evaluation

The staff reviewed the criticality evaluation of a single package and concluded that it is subcritical under the most reactive credible conditions.

6.9.5 Evaluation of Package Arrays under Normal Conditions of Transport

The staff reviewed the criticality evaluation of the most reactive array of 5N (infinite) packages and concluded that it is subcritical under normal conditions of transport.

6.9.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The staff reviewed the criticality evaluation of the most reactive array of 2N (infinite) packages and concluded that it is subcritical under hypothetical accident conditions.

6.9.7 Benchmark Evaluations

The staff reviewed the benchmark evaluation of the calculations and concluded that it is sufficient to determine an appropriate bias and uncertainties for the criticality evaluation of the package.

6.10 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

7 OPERATING PROCEDURES REVIEW

Review Objective

The objective of this review is to verify that the operating procedures comply with the requirements of 10 CFR Part 71 and ensure that the package will be operated in a manner consistent with the conditions assumed in its evaluation for approval.

The CoC has been conditioned to specify that the package shall be both prepared for shipment and operated in accordance with detailed written operating procedures to be prepared by the licensee. Procedures for preparation and operation, shall be developed in accordance the guidance presented within the application and shall include those tests and inspections detailed within the CoC.

7.1 Package Loading

Section 7.1 of the SAR specifies the loading procedures. The loading procedures include receipt inspections, preparation of the overpack and MPC for loading, loading and closure of the MPC, closure of the overpack, and preparation of the package for transport. Since the package may be used for storage, as well as for shipment, procedures for preparing the package for transport after long-term storage are also included. Section 7.4 of the SAR addresses the procedures for preparing the package for shipment after a period of storage (which is considered to be a period of more than 1 year from the date that the overpack is closed and the containment boundary seals are leak tested).

The loading procedures were reviewed by the staff and found to contain sufficient detail to allow the applicant or shipper to develop detailed loading procedures. The staff agrees that the loading procedures adequately specify the steps, tests, and determinations that must be made before each shipment to ensure that the package is operated, and will operate, in a manner consistent with its evaluation for approval. The CoC has been conditioned to require that the package be loaded and prepared for shipment in accordance with detailed written operating procedures and that those procedures shall include the following critical provisions:

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

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

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

(a) before the first shipment;
(b) within the 12-month period prior to each successive shipment;
(c) after detensioning one or more overpack lid bolts and the vent port plug; and
(d) after each seal replacement.

(4) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} std cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a (3)(a) above.

(5) Each containment boundary seal must be replaced after each use of the seal.

(6) The rupture discs on the neutron seal must be replaced after each use of the seal.

(7) The MPC-68F shall be leak tested to show a leak rate of no greater than 5×10^{-6} std cm^3/sec prior to its shipment. The leak test shall have a minimum sensitivity of 2.5×10^{-6} std cm^3/sec. This leak test is required only if the MPC-68F contains fuel debris and the fuel debris plutonium content exceeds 20 Ci per package.

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

(a) The MPC shall be evacuated, through a stepped evacuation process, to pressure of less than 3 torr.
(b) The MPC cavity shall hold a stable pressure of less than 3 torr for at least 30 minutes.

(9) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium.

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

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2895 ± 90**
Overpack Vent and Drain Port Plugs	22 +2/-0
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0
Tie-down Bolts	250 +20/-0
Transport Frame Bolts	250 +20/-0

**Tighten closure plate bolts in 5 passes and in a criss-cross pattern.

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

7.2 Package Unloading

Section 7.2 of the SAR specifies the unloading procedures. The loading procedures include receipt inspections, preparation of the overpack and MPC for unloading, unloading of the MPC, and post-unloading operations. Receipt inspections include examination of the package for visible external damage, inspection of the tamper-indicating seals, and radiation and contamination surveys. The unloading procedure includes the special precautions that must be taken to ensure personnel safety during unloading operations, to prevent over-pressurization of the MPC, and to prevent thermal shock to the spent fuel assemblies.

The unloading procedures were reviewed by the staff and found to contain sufficient detail to allow the licensee or shipper to develop detailed unloading procedures.

7.3 Preparation of Empty Package for Transport

The procedures for preparing an empty HI-STAR overpack for transport are designed to meet the requirements of 49 CFR 173.428 and 10 CFR 71.87(i). To prevent interior contamination, the overpack will not be used to ship an unsealed and previously used MPC. The staff found the procedures to have sufficient detail to allow a licensee the basis for the development of a detailed site-specific procedure for transport of an empty overpack.

7.4 Evaluation Findings

7.4.1 Package Loading

The staff reviewed the proposed special controls and precautions for transport, loading, and handling and any proposed special controls in case of accident or delay, and concluded that they satisfy 10 CFR 71.35(c).

The staff reviewed the description of the radiation survey requirements of the package exterior and concluded that the limits specified in 10 CFR 71.47 will be met.

The staff reviewed the description of the temperature survey requirements of the package exterior and concluded that the limits specified in 10 CFR 71.43(g) will be met.

The staff reviewed the description of the routine determinations for package use prior to transport and concluded that the requirements of 10 CFR 71.87 will be met.

The staff reviewed the description of the special instructions needed to safely open a package and concluded that the procedures for providing the special instruction to the consignee are in accordance with the requirements of 10 CFR 71.89.

7.4.2 Package Unloading

The staff reviewed the proposed special controls and precautions for unloading and handling and concluded that they satisfy 10 CFR 71.35(c).

7.4.3 Preparation of Empty Package for Transport

The procedures for preparing an empty HI-STAR overpack for transport are designed to meet the requirements of 49 CFR 173.428 and 10 CFR 71.87. To prevent interior contamination, the overpack will not be used to ship an unsealed and previously used MPC. The staff found the procedures to have sufficient detail to allow a licensee the basis for the development of a detailed site-specific procedure for transport of an empty overpack.

7.5 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

8 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

Review Objective

The objectives of this review are to verify that the acceptance tests for the packaging comply with the requirements of 10 CFR Part 71 for the package design and that a maintenance program will ensure acceptable packaging performance throughout its service life.

Acceptance Tests

Those acceptance tests and inspections considered critical to the safe operation of the Holtec HI-STAR 100 were captured within the CoC.

8.1.1 Visual Inspections and Measurements

The applicant has committed that the HI-STAR 100 cask materials of construction and welds shall be examined in accordance with specifications which are delineated in the appropriate HI-STAR 100 drawings and in the requirements specified in Table 8.1.1 of the SAR. The staff reviewed the commitments and concluded that, if met, there is reasonable assurance that the packaging will be fabricated and assembled in accordance with drawings and other requirements specified in the SAR.

8.1.2 Weld Inspections

Section 8.8.1 of the SAR describes the fabrication and NDE requirements for the HI-STAR 100 system. Detailed instructions for the inspections, applicable ASME Code sections, and acceptance criteria are contained in SAR Tables 8.1.1 through 8.1.3. The staff verified that weld inspections are performed to verify fabrication in accordance with the drawings, codes, and standards specified in the SAR to control weld quality. The location, type, and size of the welds are confirmed by measurements as described in the tables listed above. The staff concluded that the weld inspections are acceptable.

8.1.3 Structural and Pressure Tests

8.1.3.1 Lifting Trunnions

The lifting trunnions of the package are designed, inspected, and tested in accordance with ANSI N14.6. The maximum design lifting load of the package is 250,000 lb. Thus, the two lifting trunnions are tested for a load of 750,000 lb (i.e., 300% of the maximum design lifting load) for a minimum of 10 minutes. The accessible parts of the trunnions (e.g., outside the HI-STAR overpack) and the local shell areas are then visually examined to verify that no deformations, distortion, or cracking have occurred. Testing will be performed in accordance with written and approved procedures. Test results will be documented and will become part of the final quality documentation package.

8.1.3.2 Hydrostatic Testing

8.1.3.2.1 Overpack Containment Boundary

The containment boundary of the package is hydrostatically tested to 150 psig +10,-0 psig, in accordance with the requirements of ASME Code Section III, Subsection NB, Article NB-6000. The test pressure of 150 psig is 150% of the design MNOP. The test will be performed in accordance with written and approved procedures. Accessible weld and material inspections will be performed after the hydrostatic testing to verify maintenance of structural integrity and absence of any permanent deformations. Test results will be documented and will become part of the final quality documentation package.

8.1.3.2.2 MPC Separate Inner Container Boundary

Hydrostatic testing of the MPC secondary containment boundary will be performed in accordance with the requirements of ASME Code Section III, Subsection NB, Article NB-6000, when field welding of the MPC lid-to-shell weld is completed. The hydrostatic pressure for the test will be 125+5,-0 psig, which is 125% of the design pressure of 100 psig. Following completion of the 10-minute hold period at the test pressure and while maintaining a minimum test pressure of 125 psig, the surface of the MPC lid-to-shell weld will be visually examined for leakage and then reexamined by liquid penetrant examination performed in accordance with ASME Code, Section V, Article 6, with acceptance criteria per Section III, Subsection NB, Article NB-5350. The test will be performed in accordance with written and approved procedures. Test results will be documented and will become part of the loaded MPC final quality document package.

8.1.4 Component Tests

There are two rupture discs installed in the upper ledge surface of the neutron shield enclosure vessel of the package. The rupture discs are designed to relieve at 30 ± 5 psig. Each manufactured lot of rupture discs will be sample tested in accordance with written and approved procedures to verify their point of rupture. The sample test program and the test results will be documented and become part of the quality record documentation package.

The welds on the HI-STAR impact limiter will be visually examined in accordance with the requirements of ASME Code, Section V, Article 9, with acceptance criteria per Section III, Subsection NF, Article NF-5360. The aluminum honeycomb material will be crush tested by the material supplier in accordance with an approved procedure and the certified test results will be submitted to Holtec with each shipment of aluminum honeycomb.

8.1.1.4 Leakage Tests

The staff confirmed that the containment system of the packaging is subjected to the fabrication leakage tests as specified in ANSI N14.5, and in accordance with written and approved procedures. The acceptable leakage criterion as defined in the CoC are consistent with that identified in the containment evaluation section of the SAR (Section 4). The staff concluded these leakage tests are acceptable.

For the MPC-68F, the only MPC required to provide a separate inner container function for transport in accordance with 10 CFR 71.63(b), the robust design, flaw tolerant materials, and helium leakage test and closure weld NDE provide reasonable assurance that the containment boundary will be maintained for transportation. Since the HI-STAR 100 system is a dual purpose (storage and transport) design, an extended storage period may exist before transport. The staff concluded no credible mechanism could affect the confinement and structural integrity of the closure weld and remain undetected; thus the continued integrity of the closure weld is assured for transport after storage. The staff concluded these leakage tests are acceptable.

8.1.1.5 Component and Materials Tests

8.1.1.6 Shielding Tests

The Holtite-A neutron shield material is a poured mixture. The radial neutron shield will have a minimum thickness of 4.3 inches and the impact limiter neutron shields will have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity will be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Fabrication process control includes verification, testing, and maintenance of Holtite-A sample test lots. Measurements will be performed over the entire surface of the radial neutron shield and the impact limiters using, at a maximum, a 6 x 6 inch test grid. The measurement tests are included as a condition of the license.

8.1.1.7 Neutron Absorber Tests

After manufacturing, a statistical sample of each lot of Boral is tested using wet chemistry and/or neutron attenuation techniques to verify the minimum ^{10}B content at the ends of the Boral panel. The minimum allowable ^{10}B content is 0.0267 g/cm^2 for the MPC-24 Boral panels, 0.0372 g/cm^2 for the MPC-68 Boral panels, and 0.01 g/cm^2 for the MPC-68F Boral panels. Any panel with a ^{10}B loading less than the minimum allowed will be rejected. Tests will be performed using written and approved procedures. Results will be documented and become part of the HI-STAR 100 system quality records documentation package.

The staff's acceptance of the neutron absorber test described above is based, in part, on the fact that the criticality analyses assumed only 75% of the minimum required ^{10}B content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

Installation of the Boral panels into the fuel basket shall be performed in accordance with written and approved procedures. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a Boral panel in accordance with the engineering drawings in Chapter 1 of the SAR.

8.1.1.8 Thermal Tests

Each fabricated HI-STAR overpack shall be subjected to a thermal acceptance test to verify the heat rejection capability of the packaging. The test will be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces have been painted per the design drawings in Section 1.4 of SAR. A test cover plate will be used to seal the overpack cavity. Testing will be performed in accordance with written and approved procedures. The staff reviewed the proposed methods and acceptance criteria and has reasonable assurance that they can be carried out in a satisfactory manner.

Maintenance Tests

Section 8.2 of the application specifies a maintenance program for the package. The maintenance program includes: (1) verification leak testing of the package per ANSI 14.5, (2) replacement of metallic O-rings after each use, and (3) visual inspection of various package components prior to loading and shipment.

8.2.1 Structural and Pressure Tests

No periodic structural or pressure tests are required.

8.2.2 Leakage Tests

The staff verified that the containment system of the packaging is subjected to periodic leakage tests as specified in ANSI N14.5. The maintenance and inspection program schedule is contained in Table 8.2.1. The staff concluded these leakage tests are acceptable.

8.2.3 Component and Material Tests

8.2.3.1 Component Tests

Prior to each fuel loading, a visual examination in accordance with a written and approved procedure will be performed for the lifting trunnions (areas outside of the overpack), pocket trunnion recesses, overpack internals and externals, and the impact limiters. The examination will look for indications of overstress such as cracking, deformation, wear and tear, or damages. Repair or replacement in accordance with written and approved procedures is required if unacceptable conditions are identified.

8.2.3.2 Neutron Absorber Tests

The inert helium environment inside the MPC cavity where the Boral is located ensures that the poisons will remain effective for the life of the canister. Given the design and service conditions, there are no credible means to lose the fixed neutron poisons or for their condition to deteriorate to the extent that they could not perform their intended function. Therefore, neutron absorber maintenance tests are not necessary.

8.2.4 Thermal Tests

A thermal performance test shall be performed on each HI-STAR 100 package prior to commencing transportation operations to verify that its heat rejection capabilities are consistent with the thermal analysis. This test shall be performed after a HI-STAR package is loaded with spent nuclear fuel prior to transport or when a previously loaded HI-STAR 100 system is prepared for transport if the test has not been successfully performed in the preceding 5 years. Acceptable performance under test conditions ensures that design basis fuel cladding temperature limits which the HI-STAR package is qualified under design basis heat loads will not be exceeded during transport. No special further testing and maintenance are required.

8.2.5 Shielding Tests

The neutron shield material may experience minor water loss over time at normal condition temperatures. Periodic verification of the neutron shield integrity will be performed every 5 years that the package is in transport service. The periodic verification will be performed by radiation measurements taken at three cross-sectional planes and at four points along each plane's circumference. The measurement tests, including five additional dose measurement points for each impact limiter imposed by the staff, are included as a condition of the license.

8.3 References

1. *U. S. Code of Federal Regulations*, "Packaging and Transportation of Radioactive Material," Part 71, Title 10, "Energy."

2. U. S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (Draft Report for Comment) NUREG-1617, March 1998.

9 CONCLUSIONS

Based upon the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concluded that the Holtec HI-STAR 100 package meets the requirements of 10 CFR Part 71.

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9 CONCLUSIONS

Based upon the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Holtec HI-STAR 100 package meets the requirements of 10 CFR 71.

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