71-9226

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 27, 1998

Mr. J. Neal Blue, President, Chairman
and Chief Executive Officer
General Atomics
P.O. Box 85608
San Diego, CA 92186-5608

SUBJECT: ISSUANCE OF CERTIFICATE OF COMPLIANCE NO. 9226 FOR THE GA-4 PACKAGE (TAC NO. L22363)

Dear Mr. Blue:

In accordance with the application dated August 31, 1994, as supplemented, and pursuant to Part 71 to Title 10 of the Code of Federal Regulations, enclosed is Certificate of Compliance No. 9226, Revision 0, for the Model No. GA-4 package and the Nuclear Regulatory Commission (NRC) staff's Safety Evaluation Report.

It is important to note that the Certificate has been conditioned in two areas that are different from your request. First, authorized fuel assemblies with missing fuel pins shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the assembly. Second, based on your shielding analysis and the NRC staff's confirmatory calculations, the minimum initial enrichment is 3.0 wt. percent uranium-235 for authorized fuel types with a maximum burnup of 35,000 MWd/MTU and cooled for a minimum of 10 years. There is no minimum initial enrichment restriction for the authorized fuel types which have been cooled for 15 years or longer.

General Atomics has been registered as the holder of the Certificate of Compliance for this package.

There are no registered users of the package under the general license provisions of 10 CFR 71.12 or 49 CFR 173.471.

This approval constitutes authority to use this package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471.

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If you have any questions regarding issuance of this certificate, please contact the Project Manager, Tim McGinty, at (301) 415-8580.

Sincerely,

original /s/ by

William F. Kane, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No.: 71-9226

Enclosures: 1. Certificate of Compliance No. 9226
2. Safety Evaluation Report

cc: Mr. James K. O'Steen
Department of Transportation

Dr. Keith E. Asmussen
General Atomics

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SAFETY EVALUATION REPORT

Model No. GA-4
Legal Weight Truck Spent Fuel Shipping Cask
Certificate of Compliance No. 9226
Revision No. 0

Summary

By application dated August 31, 1994, as supplemented, General Atomics (GA) requested approval of the Model No. GA-4 Legal Weight Truck Spent Fuel Shipping Cask as a Type B(U)F package. Based on the statements and representations in the application as supplemented, and the conditions listed in the Certificate of Compliance (CoC), the Nuclear Regulatory Commission (NRC) staff has concluded that the Model No. GA-4 package meets the requirements of 10 CFR Part 71.

References

GA-4 Legal Weight Truck Spent Fuel Shipping Cask Safety Analysis Report (SAR), dated August 5, 1998.

Background

GA application dated August 31, 1994.

GA supplements dated October 7, 1996; January 31, and November 4, 1997; and August 5, 1998.

NOTE: The section/paragraph numbering in this Safety Evaluation Report (SER) follows the Standard Review Plan format.

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List of Acronyms

AEF	average energy group causing fission
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
CoC	Certificate of Compliance
DU	depleted uranium
FSS	fuel support structure
GA	General Atomics
ILSS	impact limiter support structure
MNOP	maximum normal operating pressure
NFAH	non-fuel assembly hardware
NRC	Nuclear Regulatory Commission
PWR	pressurized-water reactor
QA	Quality Assurance
RG	Regulatory Guide
SAR	Safety Analysis Report
SER	Safety Evaluation Report
2-D	two-dimensional
3-D	three-dimensional

1 General Information Review

REVIEW OBJECTIVE

The objective of this chapter is to document that the application contains sufficient depth, for consideration by the staff in the licensing process, by (1) demonstrating an overview of relevant package information, including intended use; and (2) including a summary description of the packaging, operational features, and contents adequate to provide reasonable assurance that the package can meet the regulations and operating objectives.

1.5.1 General SAR Format

The application was prepared in accordance with Regulatory Guide (RG) 7.9, "Standard Format and Content for Part 71 Applications for Approval of Packaging for Radioactive Material."

1.5.2 Package Design Information

After the NRC staff received the initial application for a CoC, dated August 31, 1994, it conducted an initial acceptance review. The NRC staff determined that the application contained sufficient information to begin its review.

1.5.2.1 Purpose of Application

The application was for the approval to transport up to four intact pressurized-water reactor (PWR) spent fuel assemblies as an exclusive-use package.

1.5.2.2 Quality Assurance (QA) Program

As documented in NUREG-0383, Volume 3, Revision 17, GA has an NRC-approved QA program that satisfies the requirements of Appendix B of 10 CFR Part 50. The previously approved Part 50 QA program satisfies the requirements of Part 71, Subpart H. The approval covered design, fabrication, assembly, testing, procurement, maintenance, repair, modification, and use. The approval was issued July 19, 1979, and the current expiration date is June 30, 2001.

1.5.2.3 Proposed Use/Contents

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask (package) will be used for truck transport of up to four intact PWR spent fuel assemblies as an exclusive-use package, in accordance with Part 71 and 49 CFR Part 173.

(1) Type and Form of Material:

a. Intact fuel assemblies. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

b. The fuel authorized for shipment in the GA-4 package is irradiated 14x14 and 15x15 PWR fuel assemblies with uranium oxide fuel pellets. Before irradiation, the maximum enrichment of any assembly to be transported is 3.15 percent by weight of uranium-235 (^{235}U). The total initial uranium content is not to exceed 407 Kg per assembly for 14x14 arrays and 469 Kg per assembly for 15x15 arrays.

c. Fuel assemblies are authorized to be transported with or without control rods or other non-fuel assembly hardware (NFAH). Spacers shall be used for the specific fuel types, as shown on sheet 17 of the Drawings.

d. The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U or 45,000 MWd/MTU with a minimum cooling time of 15 years (no minimum enrichment).

e. The maximum assembly decay heat of an individual assembly is 0.617 kW. The maximum total allowable cask heat load is 2.468 kW (including control components and other NFAH when present).

f. The PWR fuel assembly types authorized for transport are listed in Table 1.1 below. All parameters are design nominal values.

Table 1.1 - PWR Fuel Assembly Characteristics

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-15x15 (Std/ZC)	469	204	0.563	0.3659	0.0242	144
W-15x15 (OFA)	463	204	0.563	0.3659	0.0242	144
BW-15x15 (Mk.B,BZ,BGD)	464	208	0.568	0.3686	0.0265	142
Exx/A-15x15 (WE)	432	204	0.563	0.3565	0.030	144
CE-15x15 (Palisades)	413	204	0.550	0.358	0.026	144
CE-14x14 (Ft.Calhoun)	376	176	0.580	0.3765	0.028	128
W-14x14 (Model C)	397	176	0.580	0.3805	0.026	137
CE-14x14 (Std/Gen.)	386	176	0.580	0.3765	0.028	137
Exx/A-14x14 (CE)	381	176	0.580	0.370	0.031	137
W-14x14 (OFA)	358	179	0.556	0.3444	0.0243	144
W-14x14 (Std/ZCA/ZCB)	407	179	0.556	0.3674	0.0225	145.5
Exx/A-14x14 (WE)	379	179	0.556	0.3505	0.030	142

(2) Maximum Quantity of Material per Package

a. For material described in SER Section 1.5.2.3(1), four (4) PWR fuel assemblies.

b. For material described in SER Section 1.5.2.3(1), the maximum assembly weight (including control components or other NFAH when present) is 1,662 lbs. The maximum weight of the cask contents (including control components or other NFAH when present) is 6,648 lbs., and the maximum gross weight of the package is 55,000 lbs.

.4 Package Type and Model Number

USA/9226/B(U)F-85

1.5.2.5 Package Category and Maximum Activity

Package Category: Category 1

Maximum Activity:

The maximum activity of the package is controlled by the fuel authorized for shipment (see SER Section 1.5.2.3), the maximum burnup, minimum cooling time, minimum enrichment, and maximum heat load. The fuel will have: (1) a maximum burnup for each fuel assembly of 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U, or 45,000 MWd/MTU with a minimum cooling time of 15 years; (2) been stored in an approved facility for a length of time sufficient to meet the thermal criteria defined below, but not less than 10 years; and (3) a cask heat load, under any conditions of use, of no more than 2.468 kW (including control components and other NFAH when present) with a maximum fuel assembly decay heat of 0.617 kW.

1.5.2.6 Fabrication and Welding Criteria

The applicant proposed to design and construct the GA-4 in compliance with the American Society of Mechanical Engineers (ASME) Code, Section III, Division 1, which, in general, is more restrictive than the Section III, Division 3, rules for the containment systems of transport packagings. However, as discussed in Section 3.3 of NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," access limitations often hinder the ability of the fabricator to inspect multi-wall vessels in strict compliance with the ASME Code requirements. This is the case regarding the radiography of the final fabrication weld of the containment system. Because of the interface with the depleted uranium (DU) shield, this weld cannot be radiographed. This weld, however, is both volumetrically examined using ultrasonic testing and inspected using progressive liquid penetrant, after each weld pass, as discussed in NUREG/CR-3019. GA retains the ultimate responsibility for ensuring that satisfactory in-service performance of the GA-4 cask is achieved and has provisions extending the QA requirements, including inspection and audit functions, to all sub-tier suppliers.

1.5.2.7 Transport Index and Maximum Number of Packages

Transport Index for Nuclear Criticality Control

The applicant analyzed the GA-4 cask for criticality in accordance with the provisions of 10 CFR 71.55, 71.59, and 71.73. The applicant derived the dimensionless number "N" equal to 0.5 for the performance of this analysis. Therefore, the transport index for criticality control is equal to 100 (50 divided by N) per 10 CFR 71.59(b).

The transport index for shipment is determined in accordance with the loading procedures in SAR Chapter 7. The maximum radiation dose measurement taken at 1 meter is verified to yield a transport index (via dose measurement) of 100 or less in accordance with 10 CFR 71.4. The operating procedures specify using the greater of the measured dose transport index or the nuclear criticality control transport index of 100.

Maximum Number of Packages: One

The transport index for shipment cannot exceed the nuclear criticality transport index of 100, yielding a maximum of one package per exclusive-use shipment in accordance with the provisions of 10 CFR 71.4, 71.47, and 71.59 and the operating procedures in SAR Chapter 7.

1.5.3 Package Description

1.5.3.1 Packaging

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the packaging (cask and impact limiters) and the radioactive contents. The packaging is designed to transport up to four intact PWR irradiated spent fuel assemblies as authorized contents. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The overall dimensions of the packaging are approximately 90 inches in diameter and 234 inches long.

The containment system includes the cask body (cask body wall, flange, and bottom plate); cask closure; closure bolts; gas sample valve body; drain valve; and primary O-ring seals for the closure, gas sample valve, and drain valve.

Cask Assembly

The cask assembly includes the cask, the closure, and the closure bolts. Fuel spacers are also provided when shipping specified short fuel assemblies, to limit the movement of the fuel. The cask is constructed of stainless steel, DU, and a hydrogenous neutron shield. The cask external dimensions are approximately 188 inches long and 40 inches in diameter. A fixed fuel support structure (FSS) divides the cask cavity into four spent fuel compartments, each approximately 8.8 inches square and 167 inches long. The closure is recessed into the cask body and is attached to the cask flange with 12 1-inch diameter bolts.

The cask has two ports allowing access to the cask cavity. The closure lid has an integral half-inch diameter port (hereafter referred to as the gas sample valve) for gas sampling, venting, pressurizing, vacuum drying, leakage testing, or inerting. A 1-inch diameter port in the bottom plate allows draining, leakage testing, or filling the cavity with water. A separate drain valve opens and closes the port. The primary seals for the gas sample valve and drain valve are recessed from the outside cask surface as protection from punctures. The gas sample valve and the drain valve also have covers to protect them during transport.

Cask

The cask includes the containment (flange, cask body, bottom plate and drain valve seals); the cavity liner and FSS; the impact limiter support structure (ILSS); the trunnions and redundant lift sockets; the DU gamma shield; and the neutron shield and its outer shell. The cask body is square, with rounded corners and a transition to a round outer shell for the neutron shield. The cask has approximately a 1.5 inch thick stainless steel body wall, a 2.6 inch thick DU shield (reduced at the corners), and 0.4 inch thick stainless steel fuel cavity liner.

The cruciform FSS consists of stainless steel panels with boron-carbide (B_4C) pellets for criticality control. A continuous series of holes in each panel, at right angles with the FSS axis, provides cavities for the B_4C pellets. The FSS is welded to the cavity liner and is approximately 18 inches square by 166 inches long and weighs about 750 lbs.

The flange connects the cask body wall and fuel cavity liner at the cask top, and the bottom plate connects them at the bottom. The gamma shield is made up of five rings, which are assembled with zero axial tolerance clearance within the DU cavity, to minimize gaps. The ILSS is a slightly tapered 0.4 inch thick shell on each end of the cask. The shell mates with the impact limiter's cavity and is connected to the cask body by 36 ribs. Under drop conditions, loads are transferred from the cask, through the ILSS, to the impact limiters.

The neutron shield is located between the cask body and the outer shell. The neutron shield design maintains continuous shielding immediately adjacent to the cask body under normal conditions of transport. The details of the design are proprietary. The design, in conjunction with the operating procedures, ensures the availability of the neutron shield to perform its function under normal conditions of transport.

Two lifting and tie-down trunnions are located about 34 inches from the top of the cask body, and another pair is located about the same distance from the bottom. The trunnion outside diameter is 10 inches, increasing to 11.5 inches at the cask interface. Two redundant lift sockets are located about 26 inches from the top of the cask body and are flush with the outer skin. The lifting trunnions and redundant lift sockets have been designed with a safety factor of 6 against yield, and 10 against ultimate, with a dynamic load factor of 1.2.

All containment boundary welds, with the exception of the final fabrication weld, are full penetration and are radiographed and liquid penetrant examined to the requirements of the ASME Code, Section III, Subsection NB. The final fabrication weld is both volumetrically examined using ultrasonic testing and inspected using progressive liquid penetrant.

Closure and Closure Bolts

The closure is square and stepped to snugly fit inside the cask cavity. It accommodates 12 Inconel 718 closure bolts and the gas sample valve, and captures the primary and secondary seals with dove-tailed grooves. The closure is approximately 26 inches square, 11 inches thick, and weighs about 1510 lbs.

Materials

All major cask components are stainless steel, except the neutron shield, the DU gamma shield, and the B_4C pellets contained in the FSS. Most cask steel components are of Type XM-19 austenitic stainless steel. Certain miscellaneous components and trunnion wear surfaces are Nitronic 60 or 300 series stainless steels, with bolts fabricated of solution-treated Inconel 718. All O-ring seals are fabricated of ethylene propylene.

Impact Limiters

The impact limiters are fabricated of aluminum honeycomb, completely enclosed by an all-welded austenitic stainless steel skin. Each of the two identical impact limiters is attached to the cask with eight bolts. Each impact limiter weighs approximately 2,000 lbs.

Drawings

The package shall be constructed and assembled in accordance with the following GA Drawing Number:

Drawing No. 031348,
sheets 1 through 19, Revision D
GA-4 Spent Fuel Shipping Cask Packaging Assembly

1.5.3.2 Operational Features

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask incorporates several important features to facilitate operations. The cask's four trunnions are designed for both lifting and trailer tie-down loads. The lower trunnions are offset with the cask centerline, biasing the cask to tilt in the proper direction when lowered onto the trailer front supports. Quick-connect features and access to the drain port, drain valve, and gas sample valve minimize operational exposures. The gas sample valve is used to: (1) sample the cask cavity, (2) pressurize the cavity to facilitate draining, (3) evacuate the cask during drying operations, and (4) inert the cask with helium for leakage testing of the drain valve's primary seal, and for transport. A DU cavity port is used to backfill the DU cavity with helium during fabrication and is subsequently plugged and welded to seal-in the helium.

1.5.3.3 Contents

See SER Section 1.5.2.3.

1.5.4 Compliance with 10 CFR Part 71

1.5.4.1 General Requirements of 10 CFR 71.43

The applicant provided summary statements indicating that the GA-4 shipping cask was in compliance with the general standards for all packages. These statements were verified during the review process of the specific SAR chapters and found to be accurate.

Minimum Package Size

No dimension of the package is less than 4 inches. The package meets the requirement of 10 CFR 71.43(a) for minimum size.

Tamper-Indicating Feature

A wire tamper-indicating device is incorporated between the cask and each impact limiter. An intact seal will be positive evidence that the containment vessel has not been opened by unauthorized persons and satisfies the requirements of 10 CFR 71.43(b).

Positive Closure

The closure and penetrations on the cask cannot be opened unintentionally without unbolting and removing a 2,000-pound impact limiter, and either the 12 1-inch closure bolts, the gas sampling port and cap, or the drain valve and cover. The package design meets the requirements of 10 CFR 71.43(c).

Valves, Other Devices, and Continuous Venting

The cavity gas sample valve and drain valve are located in the closure lid and bottom plate, respectively, and are closed and covered during transport. Furthermore, the presence of the impact limiters prevents their inadvertent operation during transport. They are designed and analyzed with sufficient structural integrity to withstand both normal and hypothetical accident conditions of transport. The gas sample valve, drain valve, closure seals, and related passages are also designed with a secondary seal, beyond the primary containment seals. The package has no feature that would allow venting during transport. The GA-4 cask meets the requirements of 10 CFR 71.43(e) and (h).

1.5.4.2 Condition of Package after 10 CFR 71.71 and 10 CFR 71.73 Testing

Summary descriptions were provided within the SAR, and the references were verified for the physical condition of the package subsequent to the tests specified in 10 CFR 71.71 (normal conditions of transport) and 10 CFR 71.73 (hypothetical accident conditions). These statements were verified by the NRC staff and that verification is documented within the applicable sections of this SER.

1.5.4.3 Structural, Thermal, Containment, Shielding, and Criticality

Summary statements in the SAR attested to the adequacy of the package design to meet the structural, thermal, containment, shielding, and criticality requirements of Part 71. These statements were verified by the NRC staff and that verification is documented within the applicable sections of this SER.

1.5.4.4 Operational Procedures, Acceptance Tests, and Maintenance

Summary statements in the SAR, supporting the adequacy of the development of the operational procedures and acceptance tests and maintenance program to ensure compliance with the requirements of Part 71, were made by the applicant. These statements were verified by the NRC staff and that verification is documented within the applicable sections of this SER.

1.6 EVALUATION FINDINGS

1.6.1 General SAR Format

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

1.6.2 Package Design Information

The SAR drawings contain adequate detail to allow their evaluation by the NRC staff against the requirements of Part 71. Each drawing was reviewed and was found to be consistent with the SAR text. Furthermore, each drawing contains keys or annotations to explain and clarify information on the drawing.

1.6.3 Package Description

The application for package approval includes a reference to an approved QA program.

1.6.4 Compliance with 10 CFR Part 71

The application for package approval identified 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 sufficiently detailed descriptions of the package to be evaluated for compliance with Part 71.

2 Structural Review

REVIEW OBJECTIVE

Structural reviews are performed to ensure that the packaging design meets the requirements of Part 71. Loads and loading combinations are reviewed for the normal transport conditions and the hypothetical accident conditions specified in Part 71. Structural materials and material specifications are reviewed and compared with acceptable codes and standards. Design details, analysis assumptions, fabrication processes, examination procedures, and testing methods are evaluated to ensure the package is structurally adequate to meet the requirements of Part 71. This includes reviewing the structural performance in areas such as stress, buckling, fracture, and fatigue for all critical components of the package.

2.5.1 Description of Structural Design

2.5.1.1 Descriptive Information, Including Weights and Centers of Gravity

The applicant sufficiently described the function, geometry, and materials of construction for all structural components, including the lifting and tie-down devices, to allow evaluating the structural performance of the GA-4 cask under both normal and hypothetical accident conditions of transport. The packaging drawings adequately specify the materials of construction, dimensions, tolerances, and fabrication methods of the packaging.

The applicant performed structural analyses, engineering evaluations, and physical tests to demonstrate that the package is structurally adequate for meeting the Part 71 requirements. Load combinations, as specified in RG 7.8, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," were properly considered.

The structural analyses involved both manual calculations and the application of computer analysis codes, such as ANSYS, to perform finite element analysis of the trunnions, cask body and cavity liner, and FSS. These calculations were used to demonstrate the structural adequacy of the package by showing that the cask containment boundary meets the design criteria as specified in RG 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels."

The engineering evaluations showed that the choice of materials for the containment boundary, under the effects of the test conditions specified in Part 71, would preclude: (1) failure by brittle fracture, and (2) buckling of the cask body and components.

The physical tests performed by the applicant involved: (1) the crushing of quarter-scale models of the impact limiter to measure load-deflection characteristics and (2) the 30-foot free drop and the 40-inch puncture tests of a half-scale model of the packaging. The tests were performed to support and confirm the assumptions and results of the structural analyses and engineering evaluations.

2.5.1.2 Codes and Standards

The GA-4 containment boundary components were designed in accordance with RG 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NB, "Class 1 Components," and Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits." The applicant reviewed Section III, Division 3, of the ASME Code, and determined that the current containment system design meets this standard, with the exception of the final fabrication weld.

As discussed in Section 3.3 of NUREG/CR-3019, "Recommended Welding Criteria for use in the Fabrication of Shipping Containers for Radioactive Materials," access limitations often hinder the ability of the fabricator to inspect multi-wall vessels in strict compliance with the ASME Code requirements. This is the case regarding the radiography of the final fabrication weld of the containment system. Because of the interface with the DU shield, this weld cannot be radiographed. This weld, however, is examined both volumetrically, using ultrasonic testing, and by progressive liquid-penetrant inspection after each weld pass, as discussed in NUREG/CR-3019.

The ILSS and the ILSS bolt anchors are fabricated to ASME Code Section III, Subsection NG, and the impact limiter housings are fabricated to Subsection NF requirements. The neutron shield skin and ILSS bolt anchor tubes are fabricated to meet the requirements of ASME Code Section VIII, Subsection UW.

2.5.2 Material Properties

2.5.2.1 Materials and Material Specifications

The GA-4 cask body, closure, fuel cavity liner, FSS, valve bodies, lifting sockets, neutron shield outer skin, impact limiter inner housing, and trunnions are designed to be fabricated of Type XM-19 austenitic stainless steel. This material combines high strength, excellent impact resistance at low temperatures, good corrosion resistance, and good ductility. The exterior impact limiter skin is made of Type XM-11 austenitic stainless steel. These steels are approved by the ASME B&PV Code for use in Class 1 components.

The impact limiter and cask closure bolts are SB-637 alloy N07718 (solution-treated Inconel 718), with materials properties obtained from Part D, Section II, of the ASME Code (and from Code Case N-47-23 for temperatures >800° F).

The gamma shield is constructed of DU. The integrity of the casks shielding will be determined during fabrication by the performance of a continuous gamma scan, as specified in SAR Chapter 8. The composition of the neutron shield will be verified by both supplier certification and an independent analysis by the applicant, as specified in SAR Chapter 8.

Aluminum honeycomb is used as the energy-absorbing material in the impact limiters. B_4C pellets are placed within the FSS for criticality control. The crush strengths of the aluminum honeycomb to be used in the impact limiters are to be established by bench tests, with both the acceptance tests and acceptance criteria for the materials specified in SAR Chapter 8.

The B_4C pellets to be placed in the FSS will be tested, as discussed in Chapter 6 of this evaluation, and as specified in SAR Chapter 8, to verify that specifications on enrichment, theoretical density, and total stack weight are met.

The material specifications for the cask structural components are shown in SAR Section 2.3 and Table 2.3-1.

2.5.2.2 Prevention of Chemical, Galvanic, or Other Reactions

The cask is constructed from stainless steel and uses hydrogenous and DU shielding materials. The system has been designed to avoid chemical or galvanic corrosion between individual components, between the contents and components, and between the environment and components. The cask environmental conditions include air, water, helium, and the boric acid solution of the fuel pool. No paints are used anywhere on the cask, and in particular, there are no zinc-coated surfaces, which may liberate hydrogen gas from the boric acid in the spent fuel pool water. The components of the system, both metallic and non-metallic, should not give rise to significant unfavorable chemical or galvanic reactions. The NRC staff agrees that buildup of hydrogen gas from radiolysis under service conditions will not be significant. The NRC staff agrees that the neutron shield is expected to have a negligible rate of corrosion and that liberation of hydrogen gas should not be a concern. The NRC staff evaluated the cask design for the issues raised in NRC Bulletin 96-04 and finds it acceptable.

2.5.2.3 Effects of Radiation on Materials

Radiation has no known damaging effects on the packaging material properties. The applicant determined that the containment boundary O-ring seals are designed to function properly, remain below allowable temperatures, and maintain sufficient compression under normal and hypothetical accident conditions. The applicant's calculation of the maximum accumulated annual exposure of $6.8x10^5$ rads indicates that the radiation exposure on elastomer material of the ethylene propylene O-ring seal is within the radiation levels considered to be acceptable under the manufacturer's guidelines, so that radiation is expected to have only a minor influence on performance of the material. To ensure the package performance throughout its service life, SAR Chapter 8 describes the acceptance tests and maintenance program for the package.

2.5.3 Lifting and Tie-Down Standards for All Packages

2.5.3.1 Lifting Devices

There are four lifting trunnions on the cask. The front two near the cask closure are welded diagonally to the corners of the cask cross-section and are used for vertical lifting. They were evaluated for critical-load lifting of the fully loaded cask for the safety factors of 6 and 10, respectively, against the trunnion yield and ultimate strengths. For horizontal lifting of the cask with impact limiters in place, all four trunnions were considered in stress analyses. The results show that the stresses in both the trunnion body and adjacent cask wall meet the allowable limits.

2.5.3.2 Tie-Down Devices

The cask tie-down system consists of four hinged pillow block assemblies with lateral, vertical, and longitudinal braces attached to the truck trailer. The system is designed to resist the longitudinal load through the two bottom-end trunnions, the vertical load through all four trunnions, and the transverse loads through one of each of the top and bottom trunnions.

The applicant analyzed the effects of the trunnion loads on the cask wall, the intersection of the cask wall and trunnion, and the junction of the trunnion and its gussets for meeting the tie-down device requirements of 10 CFR 71.45(b). The analysis results show that maximum trunnion stresses are within the allowable limits, and that the minimum design margins exist on the trunnion side, and not on the cask wall. This demonstrates that, under excessive load, trunnion failures would not impair the ability of the package to meet other requirements of Part 71.

2.5.4 General Considerations for Structural Evaluation of Packaging

2.5.4.1 Evaluation by Analysis

The structural components of the packaging were analyzed by closed-form calculations, using well-developed theory, or by finite element analysis using the GACAP and ANSYS computer codes. The GACAP computer code was also used to evaluate the impact limiter crush force and crush depth as the result of drops. The force-deflection characteristics of the impact limiters are based on test data and computed results, using GA's ILMOD computer code. SAR Section 2.10.1 describes the analysis, methods, and verifications of the GACAP and ILMOD codes. The specific evaluations performed by the NRC staff are discussed in SER Sections 2.5.5 and 2.5.6 for both normal conditions of transportation and hypothetical accident conditions, respectively. The NRC staff concluded that the analysis results have demonstrated adequate margins of safety for the structural design.

2.5.4.2 Evaluation by Test

The packaging components evaluated by test are the impact limiters and the cask. The physical tests involved crushing the quarter-scale models of the impact limiter to measure load-deflection characteristics. A half-scale model of the packaging was subject to the 30-foot free drop and the 40-inch puncture tests, to support and confirm the assumptions and results of the structural analyses and engineering evaluations. A 30-foot drop, followed by a 40-inch puncture test, was performed in side, slapdown, and corner drop orientations. The results of the tests are discussed in SER Sections 2.5.5.7, 2.5.6.1, and 2.5.6.3.

2.5.5 Normal Conditions of Transport

A variety of evaluations were performed to demonstrate that the cask would meet the criteria specified in 10 CFR 71.43 and 71.51, when subjected to the conditions and tests specified in 10 CFR 71.71 for normal conditions of transport.

2.5.5.1 Heat

For the ambient temperature of 100°F, the applicant evaluated the effects of the heat test on the cask under maximum solar insolation, maximum decay heat, and maximum internal temperature, which result in a maximum normal operating pressure (MNOP).

The applicant evaluated the thermal growth effects on the minimum gaps between the various important-to-safety components of the cask. ANSYS analyses were also performed to determine the thermal stresses resulting from the interaction of components such as the FSS and cavity liner. The results show that the stresses in the package are acceptable and the normal-condition heat test will not adversely affect the structural performance of the package.

2.5.5.2 Cold

The applicant determined that, with no decay heat, a steady-state temperature of -40°F will have no detrimental thermal gradient effects on the cask. The evaluation focused on the effects of the difference in the shrinkage of the DU and the cask liner and the resulting thermal stresses. The results show that there will be axial contact between the DU and the steel flange and bottom plate, under the cold conditions. The thermal stresses in the cask and the DU were determined to be small compared with the allowable limits. This assures that the normal cold condition test will not adversely affect the structural performance of the package.

2.5.5.3 Reduced External Pressure

A decrease in external pressure to 3.5 psia will have no significant effect on the package. The applicant performed stress analysis by considering the external pressure on the cask wall, together with the applicable ambient temperature conditions. The analysis results show that acceptable design margins exist for both the cask containment boundary and the outer shell of the neutron shield.

2.5.5.4 Increased External Pressure

An increase in external pressure to 20 psia will have no significant effect on the package. The applicant performed stress analysis by considering the external pressure on the cask wall together with the applicable ambient temperature conditions. The analysis results show that acceptable design margins exist for both the cask containment boundary and the outer shell of the neutron shield. The neutron shield shell was shown not to buckle under the increased external design pressure.

2.5.5.5 Vibration/Fatigue

The applicant referenced a test report in considering design vibration loads of 0.9 g vertical, 0.3 g longitudinal, and 0.3 g transverse for evaluating the vibration effects on the tie-down systems during transportation. The evaluation results show that the stresses in the tie-down trunnions are well within the material endurance limits for fatigue considerations. The NRC staff concurs with the applicant's conclusion that the package is adequately designed for vibration incidental to normal conditions of transport.

Fatigue evaluations of the package were performed for the lifting and dead load cycles for a cask design life of 50 years with 50 one-way trips (25 shipments) per year, a total of 2,500 operating cycles. The NRC staff concurs with the applicant's conclusion that the fatigue life of the cask body, cavity liner, FSS, and neutron shield structure far exceeds 2,500 cycles, the expected total number of operating cycles for the package. The applicant calculated a load-and-unload fatigue life of about 1,600 cycles for the closure bolts and slightly greater than 1,400 cycles for the impact limiter attachment bolts. On this basis, the NRC staff concurs with the applicant's conclusion that the closure bolts should be replaced at least every 1,600 load-and-unload operating cycles, and the impact limiter attachment bolts should be replaced at least every 1,400 one-way cask transport trips. Replacement of the closure and impact limiter bolts every 20 years is included in the annual maintenance program, as shown in Table 8.2-1 of the SAR. It is unlikely that these bolts will approach their operating cycle limits over 20 years of service. However, their replacement frequencies should be considered, in conjunction with any available cask usage data, upon application for package recertification every 5 years.

2.5.5.6 Water Spray

The cask structure consists of metallic materials whose strength is unaffected by water spray.

2.5.5.7 Free Drop

Structural analyses were performed to demonstrate the adequacy of the package for the normal conditions of transportation involving a 1-foot free drop. The cask components considered in the analyses include the cask containment boundary, closure bolts, FSS/cavity liner, and neutron shield. The analyses show that all components have acceptable design margins when subjected to the combined effects of a 1-foot free drop under the applicable initial environment conditions of temperature, pressure, and fabrication stresses. This assures that the cask will maintain containment of its contents under normal conditions of transport, and the structural performance of the package will meet the requirements of 10 CFR 71.71.

The applicant used proprietary computer codes, GACAP and ILMOD, to determine cask deceleration g-loads in a 1-foot free drop event. GACAP calculates the cask impact response by modeling the cask as a two-dimensional (2-D), lumped-mass, single-axis beam crushed at the impact limiters, characterized with the load-deflection curves. ILMOD considers the footprint of the crushed impact limiter to generate the load-deflection curves. By varying the drop angles every 15 degrees from 0 to 90 degrees, and considering a range of impact limiter material properties and the maximum and minimum weights of the cask contents in the GACAP analysis, the applicant determined the following maximum g-loads to be used for evaluating the maximum stresses in cask components:

One-Foot Drop Test
(at cask center of gravity)

Orientation (Deg)	Max. Load for Containment, Closure Bolts, FSS, Cavity Liner (g)		Maximum Load for Neutron Shield (g)	
	Transverse	Axial	Transverse	Axial
0	15.6	0.0	16.6	0.0
15	7.4	2.0	8.2	2.2
30	4.7	2.7	5.4	3.1
45	3.7	3.7	4.1	4.1
60	6.5	11.2	6.9	11.9
75	2.6	9.9	2.7	11.4
90	0.0	14.9	0.0	16.8

To perform bounding stress analysis of the cask containment boundary, the applicant considered two ANSYS finite element models, the corner and flat models, to take advantage of the symmetric square cross-section, with rounded corners, of the cask body. Thirty-two basic load cases, 16 each for the flat and corner models, were analyzed using statically equivalent g-loads as input, with and without the MNOP. The resulting stresses were then combined with those for other normal condition loads, such as cask internal pressure, and evaluated against the allowable limits.

Four ANSYS finite element models, the frame and plate models, were considerad for the FSS/cavity liner. Two 2-D frame models, the flat and corner models, were used to evaluate the effects of the fuel assembly inertia g-loads assumed to be uniformly distributed along the length of the FSS/cavity liner. These analysis results were combined with the results of other cases, such as the MNOP and thermal conditions, and the drop-induced ovalization of the cask containment boundary. The mid-cavity and end-region plate models were used to evaluate local effects of non-uniform fuel assembly inertia g-loads by assuming that the concentrated inertia loads were applied to the FSS only at the fuel assembly support grids. The resulting stresses were superposed on the frame model effects involving the DU inertia loading, thermal conditions, and ovalization of the cask containment boundary.

The applicant performed hand stress analyses for the closure bolts and neutron shield structure. The closure bolt evaluation considered the bolt preload, differential thermal expansion between the closure and the flange, MNOP, and closure/contents impact loading for all drop angles. The neutron shield evaluation considered the effects of inertia loads and dynamic pressures of the neutron shield material.

The results of the above analyses demonstrate that stresses in the cask components are within the allowable limits. The NRC staff concurs with the applicant's conclusion that the package is adequately designed for the 1-foot free drop event.

2.5.5.8 Corner Drop

The corner drop test is not applicable because the package weight exceeds 100 kg (220 lbs.) and neither wood nor fiberboard is used as a material of construction.

2.5.5.9 Compression

The compression test is not applicable because the weight of the package exceeds 5,000 kg (11,000 lbs.).

2.5.5.10 Penetration

The applicant used a formula by the Ballistic Research Laboratory to evaluate the penetrability of the package, including the neutron shield outer skin and the impact limiter enclosure. The NRC staff concurs with the applicant's conclusion that a 13-lb. penetration bar dropping from a height of 40 inches, in accordance with 10 CFR 71.71(c)(10), will not adversely affect the ability of the cask to maintain containment of its contents.

2.5.6 Hypothetical Accident Conditions

The applicant evaluated the structural adequacy of the cask for hypothetical accident conditions through analysis and scaled model testing. The cask components considered included the cask containment boundary, closure bolts, FSS/cavity liner, neutron shield, and ILSS and its attachment bolts. The evaluations show that all cask components have acceptable design margins for the hypothetical accident conditions. This assures that the cask will maintain containment of its contents, and the structural performance of the package will meet the requirements of 10 CFR 71.73.

<u>Scaled Model Tests, Impact Limiters</u>

The applicant performed static crush tests of the 1/4-scale impact limiter models to validate the ILMOD methodology for calculating load-deflection curves. Seven tests were performed on four 1/4-scale impact limiters, three tested twice on opposite sides, at different crush angles ranging from the end to the side orientations. The NRC staff concurs with the applicant's conclusion that the test load-deflection curves correlate well with those calculated by ILMOD. The test load-deflection curves, after extrapolation, were also considered, together with the curves calculated by ILMOD, as input to GACAP, to determine the bounding free drop g-loads for the cask.

<u>Scaled Model Tests, Cask</u>

The applicant performed a series of 30-foot free drop tests and 40-inch drop puncture tests of a half-scale cask model to demonstrate cask performance and confirm the capabilities of GACAP to calculate conservative g-loads for the cask.

The half-scale model was constructed with the same cask and impact limiter materials as those for the prototype. Critical dimensions and tolerances of the cask design, including closure seals, were considered in model scaling. Steel blocks were attached to the cask model to satisfy the mass similitude requirements. The applicant conducted three sequences of drop tests: (1) side drop and puncture at side of closure; (2) 30-degree slapdown and puncture at cask body flat side; and (3) center of gravity-over-closure corner drop, puncture at closure bolt and gas sample port, and puncture at DU joint. The test program included recording cask deceleration time histories and leak testing the closure seals and gas sample port seals before and after each test sequence.

The results of the half-scale model drop tests are consistent with the analytical results. As further evaluated in the following section, there would be no permanent deformation of the cask components to compromise the leak tight containment boundary. The attachment bolts of the impact limiter would not fail. Therefore, the impact limiter would not be dislodged from the cask body. These conclusions were also demonstrated by the helium leakage tests performed during each sequence of the drop and puncture tests, which showed that the cask model performed satisfactorily in maintaining its leak tight containment boundary. The following comparison shows that GACAP was used conservatively to calculate g-loads for the cask:

Comparison between Test and Analysis Results
(at cask center of gravity)

Orientation	GACAP Analysis(g)	Model Test(g)
Side Drop		
Transverse	47.7	40-44
Slapdown, Primary		
Axial	12.3	7
Transverse	21.4	16
CG-Over-Corner Drop		
Axial	56.5	46-52

Note: Model test results have been adjusted to account for scaling effect by a factor of 2.

2.5.6.1 Free Drop

The applicant evaluated the effects of the 30-foot free drop tests on the cask design in accordance with 10 CFR 71.73(c)(1). The cask components considered included the containment boundary, FSS/cavity liner, neutron shield, ILSS, and ILSS attachment bolts.

Considering manufacturing tolerances, temperature variation, and strain loading rate of the impact limiter material, the applicant used ILMOD to generate bounding load-deflection curves for the impact limiters. GACAP was then used to obtain the following maximum g-loads to be used for evaluating maximum stresses in cask components:

Thirty-Foot Drop Test
(at cask center of gravity)

Orientation (Deg)	Max. Load for Containment, Closure Bolts, FSS, ILSS, and Cavity Liner (g)		Maximum Load for Neutron Shield (g)	
	Transverse	Axial	Transverse	Axial
0	47.7	0.0	53.0	0.0
15	21.5	5.8	23.6	6.3
30	21.4	12.3	23.2	13.4
45	23.1	23.1	24.4	24.4
60	21.8	37.8	23.0	39.8
75	14.9	55.4	16.4	61.3
78	11.9	56.5	13.1	62.9
90	0.0	61.0	0.0	69.4

The analysis models and approaches used for the 1-foot free drop were also used for the cask containment boundary, FSS/cavity liner, and neutron shield, for evaluating the accident conditions involving the 30-foot free drop tests. Additionally, a 2-D ILSS model was used to apply g-loads at four positions of the cask cross-section at 0-, 15-, 30-, and 45-degrees, with respect to the impact surface. This model allows stresses in the shells and ribs of the ILSS to be evaluated for the most damaging cask drop orientation.

The applicant analyzed 38 basic load cases by considering the MNOP and the 30-foot drops at different drop angles, including slapdown, for both the flat and corner cask models. For load combination consideration, these analysis results were evaluated to focus on only six basic cases, for which the cask was re-analyzed to also include the differential thermal expansion stresses.

Considering the 30-foot drop test g-loads, the closure bolts were evaluated using the same models and approaches as those for the normal conditions of transport.

The ILSS was evaluated for the effects of various cask drop orientations, including side, oblique, and end cask drops, as well as slapdown. In addition to the stresses in the ribs, shells, and associated weld joints, the ILSS was evaluated to ensure that the cask closure plate and bolts would not be subject to direct shear force resulting from local deformations of the ILSS.

The applicant calculated the maximum stresses in the impact limiter attachment bolts by assuming that the bolts would be activated to resist the moment produced by crushing the total unbacked area of the impact limiter against the cask. The drop test results of the half-scale cask model support the results of the attachment bolt analysis and show that the bolts will perform adequately without exceeding the stress allowables.

The applicant followed Subsection NF of the ASME Division 1, Section III, Code and NUREG/CR-6322 to evaluate buckling strength of the cask body, FSS/cavity liner, and ILSS ribs. For the neutron shield shell, the applicant states that its overall buckling is precluded by the stiffness of the cask body. The NRC staff concurs with the applicant's conclusion that the cask components are adequate in meeting the buckling design criteria.

These evaluations demonstrate that stresses in the cask components are within allowable limits and that the cask body, cavity liner, FSS, neutron shield shell, and ILSS ribs are acceptable in resisting buckling. This provides reasonable assurance that the package is adequately designed for the 30-foot free drop accident.

2.5.6.2 Crush

Because this package has a mass greater than 500 kg (1100 lbs.) and a density greater than water, this test is not applicable.

2.5.6.3 Puncture

The package was evaluated for the effects of the 40-inch drop puncture tests on meeting the requirements of 10 CFR 71.71(c)(3). The applicant used two methods, Nelm's equation and the formula by Larder and Arthur, to analyze the local behavior of the steel closure, bottom plate, and cask side wall to show that they would be of sufficient thickness to preclude punching shear failure. The results from both methods show significant margins against local punch shear failure. For the overall effects on the cask, the applicant applied a concentrated force which is equivalent to that developed in a 6-inch diameter mild steel punch, subject to a flow stress of 47,000 psi. The force was applied at either the mid-length of the package, the center of the closure, or the bottom plate. The calculated maximum primary membrane-plus-bending stresses in the cask body, closure, and bottom plate were shown to be within the allowable limits. This evaluation, as complemented by the results of the half-scale puncture tests, demonstrates the structural adequacy of the package in resisting the puncture drop accident.

2.5.6.4 Thermal

The applicant performed a fire accident analysis and calculated a maximum cask internal pressure of 90.2 psig and a maximum containment boundary temperature of 780°F. ANSYS thermal stress analysis of the cask was then performed to evaluate the stresses under the combined thermal and internal pressure conditions. On the basis of differential thermal expansions, maximum gaps between various cask components were calculated, including a maximum temporary gap of 0.024 in. between the closure and flange at the location of the primary closure seal. The applicant determined that this temporary gap would cause a reduction in seal pressure and produce axial and bending stresses in the closure bolts.

The NRC staff concurs with the applicant's conclusion that the fire-induced thermal stresses are within the allowable limits, and the differential thermal expansions will not cause a loss of cask containment integrity.

2.5.6.5 Immersion – Fissile Material

Water in-leakage is assumed for the criticality analysis of the package. Therefore, the immersion test for fissile materials is not applicable (10 CFR 71.73(c)(5)). However, immersion under 3 feet of water is equivalent to an external pressure of 1.3 psig, which has no effect on cask integrity.

2.5.6.6 Immersion – All Material

The effect of a 21.7-psig external pressure caused by immersion under 50 feet of water, as required by 10 CFR 71.73(c)(6), is of negligible consequence. The cask design has been shown to satisfy 10 CFR 71.61, which requires the containment boundary be designed to withstand an external water pressure of 290 psi (200 meters immersion pressure) without collapse, buckling, or in-leakage of water.

2.5.7 Special Requirement for Irradiated Nuclear Fuel Shipments

The applicant evaluated the cask containment boundary in meeting the requirements of 10 CFR 71.61 as follows:

Collapse Analysis

The stress state of the cask was analyzed to show that the containment boundary can withstand an external water pressure loading of 290 psi. The results show that there is a large stress margin to prevent breach or rupture of the cask containment boundary.

Buckling

The applicant performed a buckling evaluation and demonstrated adequate margin against buckling of the cask subject to an external pressure of 290 psi.

In-leakage of Water

Under an external water pressure of 290 psi, no permanent deformation of the containment boundary could result and, thus, would not affect the function of the primary seals in preventing water in-leakage.

2.5.8 Internal Pressure Test

The package MNOP is 74 psig. The cask was analyzed for a bounding internal pressure load of 80 psig. The containment boundary is pressure-tested before first use at 120 psig, which exceeds 1.5 x MNOP. The pressure test is conducted in accordance with the acceptance procedures, which are described and evaluated in SER Section 8. The calculation of MNOP is evaluated in SER Section 3 and is included in the structural evaluations for normal conditions of transport (SER Section 2.6).

2.6 EVALUATION FINDINGS

2.6.1 Description of Structural Design

The NRC staff has reviewed the package structural design description and concludes that the contents of the application meet the requirements of 10 CFR 71.31.

The NRC staff has reviewed the codes and standards used in package design and finds that they are acceptable.

2.6.2 Material Properties

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. The effects of radiation on materials were considered and package containment is constructed from materials meeting the guidelines of RGs 7.11 and 7.12. Therefore, the NRC staff concludes that the material properties are adequate to meet the requirements of 10 CFR 71.43(d).

2.6.3 Lifting and Tie-Down Standards for Package

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

2.6.4 General Considerations for Structural Evaluation of Packaging

The NRC staff has reviewed the packaging structural evaluation and concludes that the application meets the requirements of 10 CFR 71.35.

2.6.5 Normal Conditions of Transport

The NRC staff has reviewed the packaging structural performance under the normal conditions of transport and concludes that there will be no reduction in the effectiveness of the packaging.

2.6.6 Hypothetical Accident Conditions

The NRC staff has reviewed the packaging structural performance under the hypothetical accident conditions and concludes that the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of Part 71.

2.6.7 Special Requirement for Irradiated Nuclear Fuel Shipments

The NRC staff has reviewed the containment structure and concludes that it will meet the 10 CFR 71.61 requirements for irradiated nuclear fuel shipments.

2.6.8 Internal Pressure Test

The NRC staff has reviewed the containment structure and concludes that it will meet the 10 CFR 71.85(b) requirements of maintaining the structural integrity capability during the pressure test.

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.

3.5.1 Description of the Thermal Design

3.5.1.1 Packaging Design Features

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask is designed to transport up to four PWR fuel assemblies. The GA-4 cask provides for the containment of the radioactive materials under Part 71. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The packaging's overall dimensions are approximately 90 inches in diameter and 234 inches long. Several aspects of the cask design and operation provide significant thermal advantages. For normal conditions, the use of helium within the primary containment boundary and within the DU cavity enhances the transfer of decay heat from the fuel assemblies to the inner and outer shells of the cask, respectively. The FSS surrounding the fuel assemblies also aids the heat transfer by behaving like a set of internal fins to dissipate heat.

3.5.1.2 Codes and Standards

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

3.5.1.3 Content Heat Load Specification

The applicant analyzed the GA-4 cask for the transportation of up to four PWR fuel assemblies. The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years or 45,000 MWd/MTU with a minimum cooling time of 15 years. Each assembly in the package is allowed a maximum thermal load of 617 Watts, with an axial power profile that results in a peaking factor of 1.24. This gives a maximum package thermal load of 2.468 kW. The ORIGEN-S code was used to determine the assembly decay heat load using burnup, enrichment, and cooling time of the fuel. The method of determining heat load was reviewed and confirmed to be correct.

3.5.1.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, SAR Tables 3.1-1, 3.4-1, and 3.5-1, 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.

3.5.1.5 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 3.6.3-1 through 3.6.3-4) were reviewed and found consistent with the pressures presented in the "General Information," "Structural Evaluation," and "Containment Evaluation" SAR sections. These tables reported both the MNOP and the accident condition resultant pressures for each of the fuel types.

3.5.2 Material Properties and Component Specifications

3.5.2.1 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 0.8 for hypothetical accident conditions, pursuant to 10 CFR 71.73(c)(4). 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 Part 71, during normal and accident conditions.

3.5.2.2 Technical Specifications of Components

References for the technical specifications 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.

3.5.2.3 Thermal Design Limits of Package Materials and Components

The NRC 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 NRC staff verified that the design basis fuel cladding temperature of 716°F was reasonable and justified by the Pacific Northwest National Laboratory (PNL) report, PNL-4835, which is a methodology accepted by the NRC staff.

3.5.3 Thermal Evaluation Methods

3.5.3.1 Evaluation by Analyses

The NRC staff confirmed that the methods used for the thermal analysis were identified and sufficiently described to permit a complete and independent verification. The applicant used the TAC2D finite-difference and ANSYS finite-element code for its thermal evaluation. For normal conditions, the finite-difference code was used to perform a steady-state evaluation of the entire cask. For accident conditions, both the finite-difference and finite-element codes were used to model the end portions of the cask that sustained damage during the free drop and puncture tests before the fire. For normal conditions, the steady-state analysis produced a maximum cladding temperature of 313° F. This temperature is below the limit of 716° F. The maximum seal temperature under normal conditions is 143° F, which is below the extended exposure limit of 300° F.

For accident conditions, the analysis revealed a maximum cladding temperature of 442° F which is below the limit of 1058° F. Under these conditions, the maximum seal temperature was shown to be 300° F. This seal temperature for the 30-minute fire accident is below the 50-hr exposure limit of 350° F.

3.5.3.2 Evaluation by Tests

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

3.5.3.3 Temperatures

See SER Section 3.5.6.3.

3.5.3.4 Pressures

See SER Section 3.5.6.3.

3.5.3.5 Thermal Stresses

Thermal stresses were evaluated in SAR Sections 2.6.1, 2.7.3, and 2.10.12, 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. Stresses were calculated for the interaction of the cavity liner and containment boundary and for gaps associated with the DU and FSS. The gaps analyzed are depicted in SAR Figure 2.6-1. The applicant considered thermal stresses for the hot-normal, the cold-normal, and the transient accident conditions.

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.

3.5.3.6 Confirmatory Analyses

Confirmatory analyses were performed by the NRC staff by hand calculations and the ANSYS finite element code. Taking into account the unique inner geometry of the GA-4 cask with a smeared fuel mass, the NRC staff predicted cask temperatures that were in agreement with the applicant's results.

3.5.3.7 Effects of Uncertainties

The NRC 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. Because of significant design margins, the NRC staff found reasonable assurance that the applicant used appropriate considerations throughout the application.

3.5.4 Evaluation of Accessible Surface Temperature

Under normal conditions, the package is designed and constructed such that the accessible surface temperature is 170° F 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.

3.5.5 Thermal Evaluation under Normal Conditions of Transport

3.5.5.1 Heat

Under normal conditions, all of the materials used remain below their respective failure temperatures. The applicant performed three 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.

The applicant used one model to determine the temperatures under the conditions listed above. For this analysis the package was modeled using cylindrical coordinates to obtain an axial temperature distribution. A longitudinal cross-section, symmetric about the axis, employed an ambient temperature of 100° F and an adiabatic boundary along the axis. Detailed spent fuel assemblies and the FSS warranted the use of effective thermal conductivities, which combined conductive and radiative modes of heat transfer. Because the cask was designed with a square basket within a cylindrical enclosure, some gaps varied as a function of azimuthal position. For this situation, gaps were averaged around the circumference of the package, providing the effective gap thickness. The design basis decay heat used was 617 W per assembly.

3.5.5.2 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 containment system seals, would not be adversely affected by this low temperature.

3.5.6 Thermal Evaluation under Hypothetical Accident Conditions

3.5.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package under hypothetical accident conditions. Two models were used for these conditions: a 2-D, finite-difference model of the closure end with a damaged impact limiter, and a three-dimensional (3-D), finite-element analysis of the closure end with a damaged impact limiter. The 3-D model was used to provide temperature distribution for the thermal stress analysis.

The 2-D model assumed a cask in vertical orientation with the neutron shield absent and replaced with air. The outer shell is intact. Similar to the thermal evaluation for normal conditions, the model employed a longitudinal cross- section, axisymmetric about the axis, with the introduction of an adiabatic boundary at the midplane. Crushed and punctured impact limiters were modeled on the package ends since they were shown to remain attached after the drop tests. Damage to the impact limiter was exaggerated as an added conservatism to the model. A conservative convection coefficient bounded the exterior, to ensure the maximum heating of the package during the fire.

For the thermal stress analysis, the applicant modeled a 3-D cylindrical sector of the damaged cask. The model provides temperatures for the thermal stress analysis used in the structural evaluation. As an added conservatism caused by the thermal stresses introduced by the fire, the damage to the impact limiter is further exaggerated by increasing the exposed area of the closure lid. The neutron shield during the post-accident phase is assumed to be replaced with air.

Two separate groups of boundary conditions were used to determine the temperatures for maximum thermal stresses. For hot initial conditions, the applicant assumed a uniform temperature of 120° F with the design basis heat load and no solar insolation. For the cold initial conditions, the initial temperature was -20° F with zero decay heat.

For the post-accident steady-state analysis, the applicant used the model developed for normal conditions of transport but replaced the neutron shield material with air.

3.5.6.2 Fire Test

See SER Section 3.5.6.1.

3.5.6.3 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are given in Table 3.1 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
Maximum Calculated Temperatures (°F)

Location	Normal Conditions	Accident Conditions
Impact Limiter	135	1472
Outer Accessible Surface	188	1244
Neutron Shield	191	926
Cask Body	198	612
DU Shielding	202	453
Cavity Liner	222	437
Fuel Support Structure	294	426
Fuel Cladding	313	442

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, 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. Had the material been allowed to melt, this process would have resulted in a lower maximum fuel cladding temperature during the fire accident.

The NRC staff agrees with the applicant's assessment that burning of the aluminum would require temperatures and surface-to-volume ratios that are considerably higher than those for the aluminum honeycomb of the impact limiter and that the aluminum will not burn under the hypothetical accident conditions.

Potential interactions between molten aluminum and components of the cask system were also evaluated by the NRC staff. A 0.04-in-thick steel skin surrounding the aluminum honeycomb impact limiters could be at temperatures sufficient to melt the Aluminum 5052 alloy, whose melting point is about 1100°F. Any safety-related interaction would require molten aluminum to flow past the seal of the closure and into the body of the transportation cask. The stress analysis presented in SAR Section 2.7, "Hypothetical Accident Conditions," indicates that the closure bolts have sufficient margin to maintain tension on the closure under the conditions imposed by a 30-foot free drop. Further, the structural analysis indicates that the cask would have sufficient margins against the local punch shear failure in the 40-inch drop puncture test. In the "Hypothetical Accident Thermal Evaluation" of SAR Section 3.5, it is shown that the containment seals will remain functional at the temperatures and conditions of the hypothetical accident. The NRC staff further concludes that any molten aluminum would solidify in the gap and it would not contact the ethylene propylene seal.

The NRC staff also considered, qualitatively, the secondary effect of the thermal transient associated with the first 10 to 15 minutes of a hypothetical accident fire. The diameter of the cask is sufficiently small to conclude that temperatures from the center to edge of the closure would be nearly uniform. During the time that molten aluminum is likely to be present on the exterior of the cask, temperature differences through the thickness of the closure would likely be small and would promote closure (gap tightening). Therefore, differential heating is regarded as a secondary effect that could not play a significant role in distorting the lid in a manner that would aid the entry of aluminum into the cask. It is concluded that the cask containment boundary, including the fit-up of the closure, will be adequately maintained during and after the hypothetical accident conditions and that there is reasonable assurance that any molten aluminum could not penetrate the seal and enter the cask. It is concluded further that any molten aluminum that might form during the hypothetical accident would be present only outside the cask, that it would largely be contained within the impact limiter steel skin, and that it would not lead to safety-related consequences or unacceptable damage to the cask.

The applicant calculated the MNOP assuming that 100 percent of the fuel rods fail and that 30 percent 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 60,000 MWd/MTU.

The average gas temperature was calculated to be 233°F. Based on this gas temperature, the MNOP was determined to be 88.6 psia. The maximum pressure under hypothetical accident conditions is 104.9 psia, based on the average cavity gas temperature of 360°F.

3.5.6.4 Maximum Thermal Stresses

Thermal stresses as a result of the hypothetical accident condition fire are determined by finite element analysis and were evaluated in SAR Sections 2.7.3 and 2.10.12. 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, provides 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.

3.6 EVALUATION FINDINGS

3.6.1 Description of the Thermal Design

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

3.6.2 Material Properties and Component Specifications

The NRC staff has 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 Part 71.

3.6.3 Thermal Evaluation Methods

The NRC staff has 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.

3.6.4 Evaluation of Accessible Surface Temperature

The NRC staff has 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.

3.6.5 Evaluation under Normal Conditions of Transport

The NRC staff has 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.

3.6.6 Evaluation under Hypothetical Accident Conditions

The NRC staff has 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 71.73.

4 Containment Review

REVIEW OBJECTIVE

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

4.5.1 Description of the Containment System

4.5.1.1 Containment Boundary

The containment boundary of the GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the following components: (1) cask body wall, (2) bottom plate, (3) cask flange, (4) cask closure lid, (5) gas sample valve, (6) drain valve, and (7) the inner ethylene propylene elastomer O-rings on the closure lid, gas sample valve, and drain valve.

Table 4.1 lists all containment boundary components and their material of construction.

Table 4.1 GA-4 Containment Boundary Components

COMPONENT	MATERIAL	GA Drawing 031348, Rev. D, Item & Sht. No.
Cask Body Wall	ASME SA-240, Type XM-19	9, 2
Bottom Plate	ASME SA-182, Type FXM-19	8, 2
Cask Flange	ASME SA-182, Type FXM-19	10, 5
Cask Closure Lid	ASME SA-182, Type FXM-19	35, 6
Gas Sample Valve	ASME SA-479, S21800	43, 6
Drain Valve	ASME SA-479, S21800	84, 7
Elastomer O-ring (Closure Lid)	Ethylene Propylene Compound Parker E740-75	39, 5
Elastomer O-ring (Gas Sample Valve)	Ethylene Propylene Compound Parker E740-75	46, 6
Elastomer O-ring (Drain Valve)	Ethylene Propylene Compound Parker E740-75	93, 7

The containment system is designed to be leak tight, as defined in American National Standards Institute (ANSI) N14.5-1987 (i.e., a leakage rate of less than or equal to 1×10^{-7} std-cm^3/sec of air).

All containment seals are elastomer ethylene propylene O-ring seals. The cask closure is an 11 inch thick Type XM-19 stainless steel plate, which is attached to the cask body with twelve 1-inch bolts. Each bolt is torqued to 235 ± 15 ft-lbs. The closure lid is equipped with dual O-ring seals set in dovetail grooves. The dimensions of the dovetail grooves are designed to maintain sufficient squeeze and leak tightness of the primary seal during normal and hypothetical accident conditions. The gas sample valve and drain valve are seated on plugs that form part of the containment boundary in the closed position. The design of the gas sample valve and closure lid interface enables simultaneous leak testing of the closure lid and gas sample valve primary O-rings. The design of the drain valve also enables direct leak testing of its primary O-ring. The gas sample valve and drain valve are torqued to 20 ± 2 ft-lbs. According to the manufacturer's guidelines, the ethylene propylene elastomer O-ring containment seals are designed to properly operate within an accumulated radiation dose of 1.0×10^{8} rads.

4.5.1.2 Codes and Standards

All containment boundary welds, except the final fabrication weld joint connecting the cask body wall to the bottom plate, are full-penetration and are radiograph and liquid-penetrant examined, in accordance with ASME Code Section III, Division 1, Subsection NB. The final fabrication weld joint connecting the cask body wall to the bottom plate cannot be radiographed because of interference from the gamma shield. However, it is examined by both ultrasonic testing and progressive liquid-penetrant inspection, in accordance with note 20 on sheet 19 of GA Drawing No. 031348, Rev. D.

The NRC staff has reviewed the description of the containment system, as given in SAR Chapters 1 and 4. The NRC staff found reasonable assurance that: (1) the SAR describes the containment system in sufficient detail to provide an adequate basis for its evaluation; (2) the SAR identifies established codes and standards for the containment system; (3) 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; and (4) the containment system is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reactions.

4.5.1.3 Special Requirements for Damaged Spent Nuclear Fuel

Failed fuel is not considered in this review, therefore, this section is not applicable.

4.5.2 Containment Under Normal Conditions of Transport

4.5.2.1 Pressurization of Containment Vessel

Within the thermal and structural evaluations, the applicant demonstrated, and the NRC staff confirmed, that the MNOP and resulting stresses are within the structural allowables.

4.5.2.2 Containment Criteria

The containment system is designed to be leak tight (i.e., a leakage rate of less than or equal to 1×10^{-7} std-cm^3/sec), under both normal conditions of transport and hypothetical accident conditions. Therefore, it was not necessary for the applicant to calculate the releasable radiological source term or the maximum allowable leak rate. In accordance with ANSI 14.5-1987, fabrication verification, periodic verification, and assembly verification leak tests will be performed to verify the leak tightness of the containment system.

4.5.2.3 Compliance with Containment Criteria

Results of the applicant's structural and thermal evaluations show that the tests specified for normal transport and hypothetical accident conditions do not affect the integrity of the containment boundary. The applicant also performed a leak test on a full-scale closure lid and containment seal, in conditions similar to normal transport, and determined leak rates were below 1.0×10^{-7} std-cm^3/sec. The applicant calculated the maximum annual exposure to any elastomer containment seal to be 6.8×10^5 rads and stated the accumulated exposure is within the manufacturer's guidelines. Therefore, radiation is expected to have only a minor influence on performance of the O-ring material.

A fabrication verification leak test of all containment components will be performed before first use (shipment) of the package. A leak test will also be performed on containment seals after the third use of the package. All seals will be replaced annually and a periodic leak test of the containment seals will be performed annually or within 12 months before each shipment. Also, a maintenance leak test will be performed on any containment component that has been replaced or any portion of a containment component that has been repaired. Each leak test will be performed with a helium mass spectrometer and will have a test sensitivity of at least 5.0×10^{-8} std-cm^3/sec. The leak test acceptance criteria will be a leak rate no greater than 1.0×10^{-7} std-cm^3/sec. The fabrication verification, periodic, and maintenance leak test procedures are specified in the operating procedures and the acceptance tests and maintenance procedures.

After loading and final closure of the package, an assembly verification leak test will be performed before each shipment. The leak test will be performed with a pressure rise monitor and will have a test sensitivity of at least 1.0×10^{-3} std-cm^3/sec. The leak test acceptance criteria is no detectable leakage. Assembly verification leak test procedures are specified in the operating procedures.

Results of the applicant's structural and thermal analyses show that the containment system remains leak tight under the tests specified in 10 CFR 71.71. Therefore, the NRC staff has reasonable assurance that the loss or dispersal of radioactive material from the cask will be less than 10^{-6} A_2 per hour under normal conditions of transport, as required in 10 CFR 71.51(a)(1). The gas sample and drain valves are protected against unauthorized operation and are provided with leak tested O-ring seals to retain any leakage. Therefore, the package design meets the requirements of 10 CFR 71.43(e).

4.5.3 Containment Under Hypothetical Accident Conditions

4.5.3.1 Pressurization of Containment Vessel

See SER Section 4.5.2.1.

4.5.3.2 Containment Criteria

See SER Section 4.5.2.2.

4.5.3.3 Compliance with Containment Criteria

The containment boundary O-ring seals are designed to function properly, remain below allowable temperatures, and maintain sufficient compression under both normal transport and hypothetical accident conditions. The applicant also performed a leak test on a full-scale closure lid and containment seal, in conditions similar to the thermal hypothetical accident condition, and determined leak rates were below 1.0×10^{-7} std-cm^3/sec.

Results of the applicant's structural and thermal analyses show that the containment system remains leak tight under the tests specified in 10 CFR 71.73. Therefore, the NRC staff has reasonable assurance that the escape of krypton would not exceed 10 A_2 in 1 week, and the escape of other radioactive materials would not exceed A_2 in 1 week, under hypothetical accident conditions, as required by 10 CFR 71.51(a)(2).

4.6 EVALUATION FINDINGS

4.6.1 Description of Containment System

The NRC staff has reviewed the description and evaluation of the containment system and found reasonable assurance 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 constructed of materials that assure that there will be no significant chemical, galvanic, or other reaction; and (4) the gas sample and drain valves are protected against unauthorized operation and are provided with leak tested O-ring seals to retain any leakage.

4.6.2 Containment Under Normal Conditions of Transport

The NRC staff has reviewed the evaluation of the containment system under normal conditions of transport and found reasonable assurance that the package design, acceptance tests, operating procedures, and maintenance procedures satisfy 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.

4.6.3 Containment Under Hypothetical Accident Conditions

The NRC staff has reviewed the evaluation of the containment system under hypothetical accident conditions and found reasonable assurance that the package design, acceptance tests, operating procedures, and maintenance procedures satisfy the containment requirements of 10 CFR 71.51(a)(2), for hypothetical accident conditions, with no dependence on filters or a mechanical cooling system.

4.6.4 Containment Review Summary

The NRC staff reviewed the Containment Evaluation of the SAR and concludes that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of Part 71. Furthermore, the leak test methods and acceptance criteria specified in the SAR are adequate and meet the containment criteria of ANSI N14.5-1987.

5 Shielding Review

REVIEW OBJECTIVE

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

5.5.1 Description of Shielding Design

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask design is described in SAR Section 1.2 with additional details in Chapters 2, "Structural," and 5, "Shielding Evaluation." The cask is designed to transport up to four PWR assemblies with fuel burnups of 35,000 MWd/MTU and a cooling time of 10 years or 45,000 MWd/MTU and a 15-year cooling time.

Cask components include the containment (flange, cask body, bottom plate, and drain valve seals); the cavity liner and FSS; the DU gamma shield; and the neutron shield and its outer shell. With the exception of the DU gamma shield, the neutron shield, and the B_4C pellets in the FSS, all major components of the cask are stainless steel. The top and bottom ends of the cask are a solid stainless steel structure.

The cask body is square with rounded corners and transitions to a round outer shell for the neutron shield. The thickness of the neutron and gamma shields is reduced at the corners. Shielding in the non-fuel regions of the cask is tapered. The cask is designed to optimize the cask shielding configuration for minimum weights and maximum payloads.

The FSS is made up of four equal panels that contain B_4C pellets for criticality control and divide the cask cavity into four spent fuel compartments. The flange connects the cask body wall and fuel cavity liner at the top, and the bottom plate connects them at the bottom. These components contain and support the DU gamma shield. The gamma shield is made up of five rings that are assembled with zero axial clearance, within the DU cavity, to minimize the gaps at operating temperatures.

The neutron shield occupies the space between the cask body wall and the outer shell. The components of the neutron shield are proprietary and thus are not further described in this document. The proprietary version of the SAR contains a detailed description of the neutron shield, including dimensions and the specification of materials used for the neutron shield.

5.5.1.1 Packaging Design Features

See SER Section 5.5.1.3.

5.5.1.2 Codes and Standards

See SER Section 5.5.1.3.

5.5.1.3 Summary Table of Maximum Radiation Levels

The "General Information" Chapter, the "Shielding Chapter", and the Drawings in the application were reviewed for completeness of information and consistency. The information, parameters, and dimensions provided are sufficient to perform a review and were consistent among the chapters and drawings. Where appropriate, standards are identified and used. The summary table of maximum radiation levels for both normal and accident conditions outside the cask shows values within the regulatory limits for an exclusive-use shipment.

5.5.2 Source Specification

The cask is designed to transport up to four PWR spent fuel assemblies with burnups of 35,000 MWd/MTU and a cooling time of 10 years or 45,000 MWd/MTU and a 15-year cooling time. GA used a variety of codes to complete the shielding evaluation. The source term was generated using the SAS2 module of the SCALE-4.1 code. SAS2 computes gamma and neutron source terms.

5.5.2.1 Gamma Source

See SER Section 5.5.2.2.

5.5.2.2 Neutron Source

GA used the SAS2 module of SCALE-4.1 to generate the neutron and gamma source terms for the spent nuclear fuel to be transported in the GA-4 cask. A standard Westinghouse 15x15 fuel assembly with an initial enrichment of 3.0 wt. percent U-235 was used to calculate the source term. The 3.0 wt. percent enrichment produces a higher source term, because of larger neutron values than higher initial enrichments. The source specification for the shielding design assumes an axial distribution of relative burnup in the active fuel region. The SAS2 module computes gamma and neutron source terms using reactor history and cooling times for fuel assemblies.

Radiation sources in a spent fuel assembly come from four regions: the active fuel region, the bottom tie plate and skirt, the plenum and spring, and top tie plate. The active fuel region includes both gamma and neutron sources, whereas the three non-fuel regions include only gamma sources.

The gamma sources for the fuel region used in the shielding analysis include primary gammas, x-rays, conversion photons, alpha-neutron photons, prompt and fission-product gammas from spontaneous fission, and bremsstrahlung radiation. Generally, only gammas with energies from 0.8 MeV to 2.5 MeV contribute significantly to the external dose. GA used gamma energies from 0.7 MeV to 2.5 MeV in calculating the gamma dose rates outside the cask. Contributions for the neutron source term include the primary neutron source in the spent fuel, spontaneous fission, and alpha-neutron reactions.

The NRC staff performed confirmatory calculations using the SAS-2H module of SCALE 4.3 to generate the neutron and gamma source term. The information needed to develop the input deck for the SAS-2H computer run is located in SAR Chapter 5. Since the assembly was divided into six axial lengths, with each axial length having the appropriate relative power, six SAS-2H computer runs were performed. The gamma and neutron source terms determined from the confirmatory calculations do not differ significantly from the source terms determined by GA and presented in the SAR. The gamma and neutron source terms determined by the NRC staff were then used in the confirmatory calculations to verify GA's predicted dose rates.

5.5.3 Model Specification

The structural and thermal properties of the cask components were evaluated by the NRC staff. The NRC staff has determined that, based on the information provided, the cask components, including those necessary for shielding, have an acceptable design margin to maintain structural integrity during normal conditions of transport and hypothetical accident conditions. The thermal evaluation conducted by NRC staff concluded that the maximum normal operating and hypothetical accident condition temperatures would not adversely affect the casks components, including the shielding materials.

The models for normal and accident conditions were reviewed and found to be consistent with the drawings and appropriate or bounding for the analyses presented in the structural and thermal analyses.

5.5.3.1 Configuration of Source and Shielding

See SER Section 5.5.3.

5.5.3.2 Material Properties

The reported material properties were reviewed and a sample of mass and atom densities was checked and found to be correct.

5.5.4 Evaluation

5.5.4.1 Methods

The shielding analyses were performed using the PATH point kernal integration code, DORT transport code, the CSASN module of SCALE-4.3 for cross-section data, and MCNP Monte Carlo Code. The DORT, SCALE, and MCNP codes are well-established codes that are used extensively in industry. PATH was developed by GA as a gamma shielding computer program that uses the point-kernel integration technique to perform calculations of dose rates and shielding requirements for complex geometry and various source types. The PATH code has been validated against the QAD code, which is a multidimensional point kernal module of SCALE.

MCNP was developed by Los Alamos National Laboratory and is a complete shielding code with cross-section data for neutrons and gammas built into the code. MCNP is a 3-D code that can explicitly model the unusual geometry of this cask design. Using MCNP to calculate the gamma dose from the fuel region, the DU gamma shield and the neutron shield were subdivided into several subregions to determine the radial dependence of the dose rates on the material thicknesses. Dose rates were calculated over several azimuthal regions, to determine the azimuthal variation of the dose rate at the cask surface and at 2 meters from the edge of the transporter.

The PATH code was also used to calculate the gamma dose rates from the hardware region of the cask. This code calculates the exponential attenuation of gamma rays and applies single-medium buildup factors to determine the final dose rate. For the cask ends, which are made up of only stainless steel, the PATH code will generate reliable dose rate predictions. However, for the sections of the cask where the gammas from the hardware will interact with more than one material (the DU gamma shield and the stainless steel), a correction factor needs to be applied when determining the dose rate. MCNP was used to account for the differing scattering characteristics of the different shielding material. The correction factor was determined to be the ratio of the MCNP dose rate predication to the PATH dose rate prediction.

A gap analysis of the DU gamma shield was performed using the 2-D DORT code. The DU shield is divided into five pieces that are assembled with zero clearance at room temperature to minimize gaps at operating temperatures. The analysis assumed a 0.114-cm gap between two pieces of the shield in the region of the peak fuel source. This analysis yielded a dose rate increase of up to 8 percent at the cask surface and up to 1 percent at 2 meters. Based on this analysis, primary gamma dose rates on the side of the cask were increased 8 percent and 1 percent at 2 meters for conservatism.

MCNP was also used to calculate the neutron dose rate along the surface of the cask and at the cask ends. The MCNP radial model for the neutron dose, which was identical to the model used to determine the gamma dose, was used to determine the azimuthal variation of dose rate at the cask surface and at 2 meters from the edge of the transport vehicle.

An axial MCNP model was developed to describe the lower end of the cask bottom because the neutron source peaks at the lower end of the assembly. Also, the cask closure is thicker than the cask bottom plate, so the cask bottom plate will have a higher dose. A simple, cylindrical MCNP model was developed to determine the dose rate ratio between the top and bottom cask surfaces. The neutron dose rates at the top end of the cask were determined by multiplying the dose rates at equivalent locations at the bottom end by the dose rate ratio. To confirm that this resulted in a conservative dose, GA performed an axial MCNP model of the top end of the cask. This evaluation determined that the dose rates on the sides and conical surfaces of the upper impact limiter were 20 percent less than the corresponding dose rates for the lower end model.

MCNP was used to perform a ground scattering analysis, since ground scattering can be a significant component of the total external dose rate, especially at 2 meters from the transport. The ground scattering factor is generally higher for neutrons because they have a higher albedo than gammas. Three MCNP cases were run: (1) the cask without the ground, (2) a horizontal cask 3.5 feet above the ground, and (3) a cask lying on its side upon the ground. From these runs, a ground scattering factor of 1.4 for neutrons and 1.1 for gammas was applied to the results of the dose rates at 2 meters from the transport.

The applicant's calculations showed dose rates within the regulatory limits at the package surface, the vehicle surface, 2 meters from the vehicle, and at the underside of the vehicle. The NRC staff used the data generated from the confirmatory analysis of the source term as input to the MCBEND computer code to perform confirmatory shielding analyses. MCBEND uses the Monte Carlo methodology and quasi-continuous-energy cross-section data derived from UKNDL to solve radiation transport and shielding problems. With MCBEND, the NRC staff modeled the complex shield geometry of the GA-4 transportation cask and calculated the doses from direct neutrons, subcritical neutron multiplication, direct gammas, and neutron-induced gammas. The NRC staff also extended the MCBEND model to estimate the contributions of ground shine to the computed doses. The NRC staff's MCBEND results confirm that the dose rates on contact and at 2 meters, with the design basis fuel at 3.0 weight-percent ^{235}U enrichment, are within the limits specified in 10 CFR 71.47 and correspond to the dose rates calculated by GA. Although the calculated dose rates on the surface of the cask and at 2 meters from the vehicle have been determined to be within the regulations in 10 CFR 71.47, the end-user of the GA-4 transport cask is still responsible for performing a dose rate survey of the loaded cask to demonstrate compliance with the regulations before transportation.

Based on the information submitted by the applicant and from the results of the NRC staff's confirmatory analysis, the NRC staff has determined that for fuel with a burnup of 45,000 MWd/MTU and cooled for a minimum of 15 years, an initial enrichment of 3.0 wt. percent ^{235}U is bounding. Based upon information available from DOE, for fuel with burnups above 35,000 MWd/MTU, the initial enrichment of the fuel is generally not less than 3.0 wt. percent ^{235}U.

For fuel with a burnup of 35,000 MWd/MTU and cooled for a minimum of 10 years, the minimum initial enrichment authorized will be 3.0 wt. percent ^{235}U. This minimum enrichment is based upon the analyses performed by GA which identifies 3.0 wt. percent ^{235}U as the bounding initial enrichment. For enrichments less than 3.0 wt. percent ^{235}U, the neutron dose rate component of the total dose rate may be increased while the gamma dose component will essentially remain the same. To load fuels with initial enrichments less than 3.0 wt. percent ^{235}U in the GA-4 cask, GA will have to submit an amendment request with supporting analyses to demonstrate compliance with the dose rate limits specified in 10 CFR 71.47.

5.5.4.2 Key Input and Output Data

Key input and output data for the shielding calculations were identified and provided. The NRC staff reviewed the applicant's key input data and output files and found them appropriate.

5.5.4.3 Flux-to-Dose Rate Conversion

The flux-to-dose-rate conversion factors used in the shielding calculations are from ANSI/ANS-6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors."

5.5.4.4 Radiation Levels

GA performed an evaluation, using 3.0 wt. percent ^{235}U enrichments, to demonstrate the shielding evaluation of the cask is adequate to ensure compliance with the dose rate limits in

Part 71. The summary of maximum dose rates calculated by GA for the GA-4 transportation cask for normal conditions and the hypothetical accident condition is found in Table 5.1. The flux-to-dose-rate conversion factors used by GA in the shielding evaluation are from ANSI/ANS 6.1.1-1977, which is a standard accepted by the NRC.

TABLE 5.1
SUMMARY OF MAXIMUM REGULATORY DOSE RATES FOR GA-4 CASK in mR/hr (mSv/hr)

Burnup (GWd/MTU)	35			45			
Cooling Time (years)	10			15			
Number of Assemblies	4			4			
Normal Conditions							
Package Surface	**Gamma**	**Neutron**	**Total**	**Gamma**	**Neutron**	**Total**	**Reg.**
Side	105.5 (1.06)	57.7 (0.58)	163.2 (1.63)	81.6 (0.82)	116.2 (1.16)	197.8 (1.98)	200 (2)
Top End	20.8 (0.21)	2.0 (0.02)	22.8 (0.23)	11.3 (0.11)	12.0 (0.12)	23.3 (0.23)	200 (2)
Bottom End	48.1 (0.48)	6.0 (0.06)	54.1 (0.54)	33.8 (0.34)	12.0 (0.12)	45.8 (0.46)	200 (2)
2 m from Vehicle Surface	**Gamma**	**Neutron**	**Total**	**Gamma**	**Neutron**	**Total**	**Reg.**
Side	6.59 (0.07)	1.68 (0.02)	8.27 (0.08)	3.72 (0.04)	5.05 (0.05)	8.77 (0.09)	10 (0.1)
Rear	1.54 (0.02)	0.15 (0.002)	1.69 (0.02)	1.07 (0.01)	0.29 (0.003)	1.36 (0.014)	10 (0.1)
Back of Cab	0.278 (0.003)	0.042 (0.0004)	0.32 (0.003)	0.19 (0.002)	0.10 (0.001)	0.29 (0.003)	2 (0.02)
Hypothetical Accident Conditions							
1 m from Damaged Cask	**Gamma**	**Neutron**	**Total**	**Gamma**	**Neutron**	**Total**	**Reg.**
Side (peak)	103 (1.03)	194 (1.94)	297 (2.97)	75 (0.75)	398 (3.98)	473 (4.73)	1000 (10)

The methods used to determine the dose rates were described in the preceding section. The transporter used in the dose rate evaluations was an 8-foot-wide semitrailer with 19.6 feet between the top of the impact limiter and the rear of the tractor's cab. The cask is to be mounted on the trailer bed with the corners facing down. The dose rate evaluation for the hypothetical accident condition assumes a complete loss of neutron shield and the stainless steel outer skin. Dose rates for the accident condition were determined in the same manner as for normal transport conditions, except that the neutron shield and outer shell are missing.

The NRC staff performed calculations to confirm the applicant's dose rate results for normal conditions. The NRC staff used the SAS2H sequence in the SCALE system to generate the radiation source terms. The shielding calculations used this source term with the MCBEND code. The dose rates calculated by the NRC staff are consistent with those presented in the application. The operating procedures specify that the loaded package is to be surveyed prior to shipment to verify that the dose rates are within the limits specified in NRC and Department of Transportation regulations.

The NRC staff agrees, in conjunction with the minimum enrichment conditions discussed previously, with the applicant's conclusion that the package shielding, together with the radiation survey performed before each shipment, are adequate to assure that external dose rates are within allowable limits.

5.6 EVALUATION FINDINGS

5.6.1 Description of the Shielding Design

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

5.6.2 Source Specification

The NRC staff has reviewed the source specifications used in the shielding evaluation and, in conjunction with the minimum enrichment conditions discussed previously, found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against Part 71 shielding requirements.

5.6.3 Model Specification

The NRC staff has 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.

5.6.4 Evaluation

The NRC staff has reviewed the external radiation levels of 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 an exclusive-use vehicle.

The NRC staff has reviewed the package design, construction, and preparations 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 NRC staff has reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the maximum 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.

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 Part 71 for exclusive-use shipments with a transport index for criticality control of 100. The analysis and transport index limits the number of packages in a shipment to a single GA-4 cask.

6.5.1 Description of the Criticality Design

The applicant described the packaging in sufficient detail to provide an adequate basis for its evaluation. The descriptions in the SAR include the types and dimensions of materials of construction and materials specifically used as nonfissile neutron absorbers or moderators.

The structural analysis shows that the configuration of the fuel basket will be maintained under normal conditions of transport and hypothetical accident conditions.

6.5.1.1 Packaging Design Features

The GA-4 cask can transport up to four intact, irradiated PWR fuel assemblies. Each fuel assembly comprises a 14 X 14 or 15 X 15 pin array with Zircaloy-clad fuel rods. The fuel rods contain uranium dioxide (UO_2) pellets with a maximum initial uranium enrichment of 3.15 wt. percent ^{235}U. The detailed contents description is provided in SER Section 1.5.2.3., and is discussed further, later in this evaluation.

A cruciform FSS provides the spacing and neutron poison needed to prevent criticality. The FSS' four identical panels are made of stainless steel. Each panel contains pellets of 96 wt. percent boron-10 (^{10}B) enriched B_4C, stacked in a uniform series of 292 holes running perpendicular to the FSS axis. The FSS is welded to the stainless steel liner of the spent fuel cavity. Radially outside the cavity liner are a gamma shield of DU metal and a hydrogenous neutron shield. Material specifications, fabrication controls, and the acceptance measurements and tests described in SAR Section 8.1 ensure that each FSS hole contains no less than the specified minimum ^{10}B loading.

6.5.1.2 Codes and Standards

The applicant identified, where appropriate, the codes and standards used in all aspects of the criticality design and evaluation.

6.5.1.3 Summary Table of Criticality Evaluations

The applicant provided a summary of the final criticality results in SAR Table 6.4-4. The table addresses results for a single package, and arrays of damaged and undamaged packages, as required by 10 CFR 71.55 and 71.59. The summary table illustrates that the GA-4 cask meets the criticality criteria of Part 71.

6.5.1.4 Transport Index

The applicant specified a number "N" of 0.5 and analyzed the appropriate number of packages in arrays. The GA-4 cask, based on these analyses, has a transport index for criticality control (50/N) of 100, which was verified to be specified consistently throughout the SAR. The transport index thus limits to one the number of GA-4 casks that can be transported in a single shipment.

6.5.2 Spent Nuclear Fuel Contents

The specifications of the spent fuel used in the criticality evaluation and authorized for transport are consistent with those specified in other SAR sections. The GA-4 cask can transport up to four intact, irradiated PWR fuel assemblies. Each fuel assembly comprises a 14x14 or 15x15 pin array, with Zircaloy-clad fuel rods. The fuel rods contain UO_2 pellets, with a maximum initial uranium enrichment of 3.15 wt. percent ^{235}U. The detailed contents description is provided in SER Section 1.5.2.3. and, is discussed further, later in this evaluation.

6.5.3 General Considerations for Evaluations

6.5.3.1 Model Configuration

In the criticality calculations, the applicant conservatively assumed fresh fuel without burnable poisons. The boron carbide pellet stacks were approximated as continuous cylinders of B_4C, with a single axial gap located at either end or in the middle of the stack. Each B_4C cylinder was assumed to have the minimum pellet diameter and the minimum pellet-stack height. The size of the axial gap in the pellet-stack model was taken as the maximum-tolerance difference between the FSS hole depth and pellet-stack height at bounding operating temperatures. The poison material was modeled as 96 wt. percent ^{10}B enriched B_4C with its density adjusted to give 90 percent of the specified minimum ^{10}B loading in each pellet stack. The DU shielding material was modeled with a ^{235}U content greater than the specified maximum. The rounded, contoured DU shield was approximated with a square model geometry. Pure water conservatively replaced the actual neutron shielding material in the applicant's computational model.

Control components, if present, were ignored in the modeling. All fuel rods were modeled intact with no rods missing and the maximum enrichment throughout.

The models for normal and accident conditions were reviewed and found to be consistent with the drawings and in keeping with the structural and thermal analyses.

6.5.3.2 Material Properties

The reported material properties were reviewed and a sampling of mass and atom densities was checked and found to be correct. GA specifies that the boron in the pellets be 96 wt. percent ^{10}B enriched and the minimum density be 96 percent of theoretical. The boron carbide pellets are acceptance-tested, as described in SAR Section 8.1.4.4, to verify that these specifications are met. Additionally, the drawings specify a minimum diameter, length, and total weight of ^{10}B of an assembled pellet stack. The applicant's specifications and acceptance methods for the boron carbide pellets were reviewed by the NRC staff and determined to be acceptable.

The NRC staff verified that the B_4C is essentially inert and will not be attacked or degraded under expected service conditions. Only negligible amounts of radiation-induced swelling of the B_4C are expected in service and, in the unlikely event that a pellet is cracked in service, the neutron absorption characteristics will remain essentially unaltered during the service period.

The applicant's criticality calculations conservatively assume DU with 0.3 wt. percent ^{235}U, and GA Drawing 031348 specifies a maximum ^{235}U concentration of 0.2 wt. percent.

6.5.3.3 Computer Codes and Cross-Section Libraries

The applicant's criticality analysis used KENO-Va and 27BURNUPLIB within the CSAS25 sequence of the SCALE-4.3 code system. To validate the computational method for this application, the applicant performed benchmark calculations for a set of 27 fresh-fuel critical experiments. The input files for the KENO-Va runs were provided in SAR Section 6.6 and reviewed by the NRC staff. In addition, benchmarking of the analytic method by the applicant and independent calculations by the NRC staff provide reasonable assurance that the package meets regulatory requirements.

6.5.3.4 Demonstration of Maximum Reactivity

The applicant determined that the Westinghouse 15x15 OFA was the most reactive of the requested contents. Table 6.1 presents a summary of the relative computed reactivities of the requested contents, as modeled by the applicant. Using the most reactive contents, the applicant performed a series of calculations to determine the most reactive configurations under optimally flooded conditions. The calculations showed that the closest assembly pitch, with the assemblies touching the FSS, was most reactive. Although reactivity was shown to be only weakly sensitive to the assumed positioning of maximum gaps in the B_4C pellet stacks, the case with the gaps at the outsides of the pellet stacks did appear to be marginally more reactive than the cases with gaps at the insides or middle of the pellet stacks. All flooded cases assumed flooding within the fuel-clad gap.

Table 6.1 Criticality Safety Basis for Allowed Contents
(Assembly types listed in order of decreasing maximum k_{eff})

Fuel Assembly Type (Mfr.-Array-Version)	Ratio k_{eff} to k_{USL} *	Active Length (inches)		MTU (3.15 wt. % ^{235}U)	
		Modeled	Nominal**	Modeled	Nominal**
W-15x15-Std/ZC	0.9990	144.35	142-144	0.4660	0.4563-0.469
W-15x15-OFA	0.9990	144.35	144	0.4660	0.4627
BW-15x15-Mk.B,BZ,BGD	*0.995*	142.70	141.8	0.4765	0.4636
Exx/A-15x15-WE	*0.990*	144.35	144	0.4422	0.432
CE-15x15-Palisades	*0.987*	144.35	132.0 (144)	0.4459	0.413
CE-14x14-Ft.Calhoun	*0.977*	128.40	128	0.3784	0.376
W-14x14-Model C	*0.974*	137.75	136.7	0.4147	0.397
CE-14x14-Std/Gen.	*0.973*	137.75	137	0.4060	0.386
Exx/A-14x14-CE	*0.971*	137.75	134.1 (137)	0.3921	0.381
W-14x14-OFA	*0.965*	144.35	135.2-144	0.3641	0.336-0.358
W-14x14-Std/ZCA,/ZCB	*0.965*	146.00	141.2-145.2 (145.5)	0.4191	0.389-0.407
Exx/A-14x14-WE	*0.952*	142.70	142	0.3728	0.379

*k_{USL} = 0.9331 for the most limiting assembly types. Ratio values in italics are estimated from applicant's calculations.
**Nominal values are taken from DOE/RW-0184. Where different, nominal values quoted in the SAR are shown in parentheses.

The applicant's analysis demonstrates the most reactive case for the single package, array of undamaged packages and arrays of damaged packages, and is further evaluated in SER Section 6.5.4.

6.5.3.5 Confirmatory Analyses

See the Evaluation Findings in SER Section 6.6, below.

6.5.4 Single Package Evaluation

6.5.4.1 Configuration

To show that the package meets the single-package requirements of 10 CFR 71.55(b), the applicant analyzed a single water-reflected package fully flooded by water at various densities. Flooding with full-density water, conservatively modeled as 1.0 g/cm^3, was most reactive. In addressing the requirements of 10 CFR 71.59 for arrays of undamaged packages, the applicant modeled an infinite array of dry packages. Results of the structural and thermal analyses showed that the dimensions and arrangements of package internals would not be changed under accident conditions. Because the application is for a criticality transport index of 100, with N=0.5, the applicant's analysis for the damaged array was the same as that for the single package.

6.5.4.2 Results

Table 6.2 summarizes the reported results of the criticality analysis.

Table 6.2 Summary of Applicant's Criticality Safety Calculation Results

Analyzed Configuration	Ratio of maximum calculated k_{eff} to k_{USL} *
Single Package: Optimally moderated and reflected by full-density water	0.9990
Array of ≥ 5N Undamaged Packages: Bounded for N=0.5 by infinite array of dry undamaged packages	0.3164
Array of ≥ 2N Damaged Packages: For N=0.5, analyzed as a single package optimally moderated and reflected by full-density water	0.9990

*k_{USL} = 0.9331

6.5.5 Evaluation of Package Arrays Under Normal Conditions of Transport

6.5.5.1 Configuration

See SER Section 6.5.4.1

6.5.5.2 Results

See SER Section 6.5.4.2

6.5.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

6.5.6.1 Configuration

See SER Section 6.5.4.1

6.5.6.2 Results

See SER Section 6.5.4.2

6.5.7 Benchmark Evaluations

6.5.7.1 Experiments and Applicability

The applicant's criticality analysis used KENO-Va and 27BURNUPLIB within the CSAS25 sequence of the SCALE-4.3 code system. To validate the computational method for this application, the applicant performed benchmark calculations for a set of 27 fresh-fuel critical experiments. All of the experiments consisted of various configurations of UO_2 fuel-pin arrays moderated by water or borated water. Uranium enrichments ranged from 2.35 to 4.74 wt. percent ^{235}U. In nine of the benchmark experiments, fuel-pin arrays were separated by absorber curtains consisting of B_4C-pellet-loaded pins in two cases, boral in three cases, and borated stainless steel in four cases. Four other experiments included reflection by DU. Other important physical characteristics of the benchmarks, such as the pitch-to-diameter ratio of the fuel rods and the CSAS25-computed average energy group causing fission (AEF), were similar to and bracketed those of the package.

6.5.7.2 Bias Determination

The applicant used the methodology from NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," Section 4.1.1, Upper Subcritical Limit Method 1, to perform a statistical trending analysis of the 27 benchmark results. Considering AEF, rod pitch, fuel enrichment, hydrogen-to-uranium atom ratio, and Dancoff factor as potential bias-trending parameters, the analysis showed that rod pitch gave both the strongest correlation for linear variation of the computed critical eigenvalue and the lowest upper-safety-limit eigenvalue (k_{USL}) for the package analysis. Using an eigenvalue administrative margin of 0.05, the applicant invoked a uniform k_{USL} of 0.9326 even though the analysis supported a slightly higher k_{USL} of 0.9331 for the rod pitch of the limiting assembly type. Although the NRC staff has chosen to use the latter k_{USL} value in its evaluation of the analysis, this choice has not affected any conclusions drawn from the analysis.

The NRC staff noted that the reported benchmarking results for the two B_4C experiments showed greater-than-average underpredictions of the critical eigenvalue. Specifically, the two B_4C experiments had CSAS25-calculated critical eigenvalues of 0.9864 ± 0.0019 and 0.9875 ± 0.0017, whereas the corresponding trended average of the eigenvalue for all 27 benchmarks was 0.9916 (i.e., k(pitch) in SAR Figure 6.1-3). The NRC staff also noted, however, that the two B_4C benchmarks were part of a three-experiment set taken from a single facility (SAR Ref. 6.5-3), and that the CSAS25-computed critical eigenvalue for that set's third experiment, which had no B_4C pins or other absorber curtains, was even lower than the other two at 0.9844 ± 0.0018. It was further noted that the benchmark models assumed nominal ^{10}B loadings in the stacked-pellet B_4C absorber pins, whereas the package analysis method assumes 90 percent-of-minimum loadings. Based on the latter observations, the NRC staff believes that the applicant's analysis of the combined set of 27 benchmark results did produce a conservative estimate of bias for the package calculations.

6.6 EVALUATION FINDINGS

The NRC staff performed independent confirmatory calculations using the MONK7B Monte Carlo code with quasi-continuous-energy cross-section data derived primarily from UKNDL. A set of 22 benchmarks of thermal UO_2 rod lattices shows that MONK7B generally overpredicts the reactivity of such systems by approximately 0.5 percent. Without any bias allowances, the MONK7B-computed k_{eff} for the bounding case described above was 0.9381 ± 0.0008. Repeating the calculation with natural uranium (0.7 wt. percent ^{235}U) in place of DU (DU, modeled with 0.3 wt. percent ^{235}U) as the gamma shielding material resulted in a k_{eff} of 0.9512 ± 0.0010, thereby demonstrating a significant sensitivity to the ^{235}U content in the DU shielding material. MONK7B calculations with both exact- and square-geometry models of the DU shield showed statistically negligible differences in reactivity. Sensitivity calculations with the limiting-case fuel density increased by 0.8 percent, likewise showed insignificant reactivity effects. The NRC staff's calculations also confirmed the applicant's conclusions regarding the most reactive fuel type and the most-reactive conditions of fuel-assembly pitch, B_4C-gap location, and internal water density.

Table 6.3 summarizes key specifications for the package materials and components affecting criticality safety. These specifications ensure that the actual dimensions and compositions of the FSS and DU shield are consistent with the stated bases for the applicant's calculational models. In particular, the tabulated specifications establish the following relationships between the package and the analysis model:

- Each B_4C pellet stack has at least 111 percent (i.e., 1/0.90) of the ^{10}B content assumed in the applicant's computational model.
- The actual length and outer surface area of each B_4C pellet stack are no less than in the computational model.
- The sizes of B_4C pellet gaps within the FSS holes are no greater than those considered in the analysis model.
- The DU shielding material contains no more ^{235}U than is assumed in the analysis model.

Table 6.3 Key Package Component Specifications Affecting Criticality Safety

Specified Parameter	Minimum	Maximum
B_4C boron enrichment	96 wt. percent ^{10}B	N/A
Diameter of each B_4C pellet	0.426 in	0.430 in
Height of each B_4C pellet stack	7.986 in	8.046 in
Mass of ^{10}B in each B_4C pellet stack*	31.5 g	N/A
Mass of each B_4C pellet stack*	43.0 g	45.0 g
Diameter of each FSS hole	0.432 in	0.44 in
FSS nominal hole pitch	N/A	0.55 in
FSS hole depth minus B_4C pellet-stack height (at room temperature)	0.009 in	0.129 in
Thickness of each FSS panel	0.600 in	0.620 in
Fuel cavity width	N/A	9.135 in
^{235}U content in DU shielding material**	N/A	0.2 wt. percent

* The applicant specifies only the minimum mass of ^{10}B in each pellet stack, a quantity that is not readily measured. Assuming B_4C with 1.0 wt. percent chemical impurities (typical), the NRC staff notes that this corresponds to a minimum pellet stack mass of 43.0 grams. Pellet stack masses greater than 45.0 grams suggest that other B_4C parameters may be outside their specified acceptance limits (e.g., very low ^{10}B enrichment, leading to higher B_4C mass density).
** The applicant's criticality calculations conservatively assume DU with 0.3 wt. percent ^{235}U.

Based on its review of the information and analyses reported by the applicant and its own calculations, the NRC staff has determined that there is reasonable assurance that the package design meets the criticality safety requirements in Part 71.

6.6.1 Description of Criticality Design

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

The NRC staff has reviewed the summary information of the criticality design and found reasonable assurance that it indicates the package is in compliance with the requirements of Part 71.

6.6.2 Spent Nuclear Fuel Contents

The NRC staff has reviewed the description of the spent nuclear fuel contents and found reasonable assurance that it provides an adequate basis for the criticality evaluation.

6.6.3 General Considerations for Evaluations

The NRC staff has reviewed the criticality description and evaluation of the package and found reasonable assurance that it addresses the criticality safety requirements of Part 71.

6.6.4 Single Package Evaluation

The NRC staff has reviewed the criticality evaluation of a single package and found reasonable assurance that it is subcritical under the most reactive credible conditions.

6.6.5 Evaluation of Package Arrays under Normal Conditions of Transport

The NRC staff has reviewed the criticality evaluation of the most reactive array of 5N packages and found reasonable assurance that it is subcritical under normal conditions of transport.

6.6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The NRC staff has reviewed the criticality evaluation of the most reactive array of 2N packages and found reasonable assurance that it is subcritical under hypothetical accident conditions.

6.6.7 Benchmark Evaluations

The NRC staff has reviewed the benchmark evaluation of the calculations and found reasonable assurance that the calculations are sufficient to determine an appropriate bias and uncertainties for the criticality evaluation of the package.

7 Operating Procedures Review

REVIEW OBJECTIVE

The objective of this review is to verify that the operating procedures comply with the requirements of 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 applicant. Procedures for preparation and operation shall be developed in accordance with the guidance presented within the application and shall include those tests and inspections detailed within the CoC.

7.5.1 Package Loading

The loading procedures for the GA-4 spent fuel shipping cask provide for both the typical wet loading at reactor sites and for dry loading at facilities where a hot cell is available.

7.5.1.1 Preparation for Loading

The package preparation procedures specify a receipt inspection, radiation surveys, and contamination surveys of package-accessible surfaces and the trailer. A redundant lifting fixture is specified and used if required for the heavy-loads specification of certain utilities. The preparation procedures also specify visual inspection and replacement, if necessary, of seals that are part of the containment boundary. All containment boundary seals are verified to have been replaced within the previous 12 months, as specified in the maintenance procedures.

7.5.1.2 Loading

The loading procedures were reviewed by the NRC staff and found to contain sufficient detail to allow the applicant, as required by the CoC, to develop detailed loading procedures. The cask loading procedure requires specific identification and verification that the fuel to be loaded in the cask meets the specifications of the CoC. Critical requirements both for the closure of the cask body and for the leak testing and transport readiness of the GA-4 cask have been made a part of the CoC. Those requirements include the following:

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) That before shipment the licensee shall:
   a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.

b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates: 3.4 x (peak neutron dose rate at any point on cask surface at its midlength) + 1.0 x (gamma dose rate at that location) $\leq$ 1000 mR/hr.

c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.

d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1 x10^{-3} std-cm^3/sec and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in SAR Chapter 8.

3) Before leak testing, the following closure bolt and valve torque specifications:
   a) The cask lid bolts shall be torqued to 235 ± 15 ft-lbs.
   b) The gas sample valve and drain valve shall be torqued to 20 ± 2 ft-lbs.

4) During wet loading operations and before leak testing, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:
   a) Cask evacuation to a pressure of 0.2 psia (10 mm Hg) or less for a minimum of 1 hour.
   b) Verifying that the cask pressure rise is less than 0.1 psi in 10 minutes.

5) Before shipment, independent verification of the material condition of the neutron shield as described in SAR Section 7.1.1.4 or 7.1.2.4.

7.5.1.3 Preparation for Transport

The operating procedures specify decontamination of the accessible surfaces of the cask, before shipment, as required by 10 CFR 71.87(I). The procedures also specify providing written instructions to the carrier, as required by 10 CFR 71.47 for exclusive-use shipments. A thermal acceptance test will be performed on the first GA-4 cask to be fabricated, to verify the heat rejection capability of the packaging. This test is stipulated as a Condition of the Certificate. The test results will be correlated with the analytical predictions. Satisfactory performance of the thermal acceptance test will provide reasonable assurance that, before the first use of the package design, the external surface temperature requirements of 10 CFR 71.43(g) will be met for the design bases contents.

The NRC staff has previously evaluated, in this document, and determined acceptable, provisions of the design and loading procedures with respect to radiation surveys, leakage testing, a tamper-indicating feature, and tie-downs.

7.5.2 Package Unloading

7.5.2.1 Receipt of a Package from Carrier

Package receipt procedures were reviewed and found to have sufficient detail to allow a licensee a basis for the development of a detailed site-specific procedure for the receipt of a cask. The applicant specified a receipt inspection, cask and trailer inspection (including the integrity of the tamper-indicating seals), and the performance of dose and contamination surveys to ensure compliance with 10 CFR 71.87.

7.5.2.2 Preparation for Unloading

The procedures for wet and dry unloading preparations were reviewed by the NRC staff and contain sufficient detail to provide the basis for the development of detailed procedures by the licensee. Provisions are provided to check and depressurize the cask cavity to the facility's processing system. During wet unloading, provisions to monitor and control the cask refill rate, to prevent exceeding MNOP, are included.

7.5.2.3 Contents Removal

The procedures for wet and dry unloading procedures were reviewed by the NRC staff and, with consideration for sequencing, contain the same acceptable provisions of the loading procedures. They contain sufficient detail, as required by the CoC and in conjunction with the empty package procedures, to provide the basis for the development of detailed contents removal procedures by the licensee.

7.5.3 Preparation of Empty Package for Transport

The applicant's preparation procedures for empty package transport perform the appropriate radiation and contamination surveys, as required by 10 CFR 71.47, 71.87, and 49 CFR 173.428. The procedures contain sufficient detail to provide the basis for the development of detailed procedures by the licensee.

7.6 EVALUATION FINDINGS

The operating procedures review resulted in the following findings:

7.6.1 Package Loading

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

The NRC staff has reviewed the description of the radiation survey requirements of the package exterior and found reasonable assurance that the limits specified in 10 CFR 71.47 will be met.

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

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

The NRC staff has reviewed the description of the special instructions (if applicable) needed to safely open a package and found reasonable assurance that the procedures for providing the special instruction to the consignee are in accordance with the requirements of 10 CFR 71.89.

7.6.2 Package Unloading

The NRC staff has reviewed the proposed special controls and precautions for unloading and handling and found reasonable assurance that they satisfy 10 CFR 71.35(c).

7.6.3 Preparation of Empty Package for Transport

The NRC staff has reviewed the description of the routine determinations for package use before transport and found reasonable assurance that the requirements of 10 CFR 71.87 will be met.

7.6.4 Other Procedures

The NRC staff made no findings with respect to other procedures.

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 Part 71 for the package design and that a maintenance program will ensure acceptable packaging performance throughout its service life.

Acceptance Tests

Section 8.1 of the application specifies all testing required on the GA-4 cask before its first use. SAR Table 8.1-1 summarizes the acceptance tests. The acceptance tests and inspections considered critical to the safe operation of the GA-4 are captured within the CoC.

8.2.4.1 Visual Inspections and Measurements

The applicant has committed that the GA-4 cask materials of construction and welds shall be examined in accordance with the specifications delineated on the GA-4 Spent Fuel Shipping Cask Packaging Assembly Drawing No. 031348, sheets 1 through 19, Revision D. The NRC staff has reviewed the commitments and has 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.

8.2.4.2 Weld Inspections

The GA-4 containment boundary components were designed in accordance with RG 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," ASME B&PV Code, Section III, Subsection NB, "Class 1 Components," and Appendix F, "Rules for Evaluation of Service Loadings with Level D Service Limits." The applicant reviewed Section III, Division 3, of the ASME Code, and determined that the current containment system design meets this standard, with the exception of the final fabrication weld.

However, as discussed in Section 3.3 of NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," access limitations often hinder the ability of the fabricator to inspect multi-wall vessels in strict compliance with the ASME Code requirements. This is the case regarding the radiography of the final fabrication weld of the containment system. Because of the interface with the DU shield, this weld cannot be radiographed. This weld, however, is both volumetrically examined using ultrasonic testing, and by progressive liquid-penetrant inspection, after each weld pass, as discussed in NUREG/CR-3019.

8.2.4.3 Structural and Pressure Tests

The acceptance tests specify proof loading tests of the trunnions and redundant lifting sockets. The upper trunnions are tested with a vertical lift, the redundant lifting sockets are tested with a vertical lift, and all four trunnions are tested for a horizontal lift. The vertical lifts apply a load to each trunnion and redundant lifting socket, for 10 minutes, equal to 300 percent of one-half of the combined weight of the cask (without impact limiters), contents, and water in the cavity. The horizontal lift applies a load equal to 150 percent of one-quarter of the cask's design weight (55,000 lbs.) to each trunnion. Following the tests, all trunnions and lifting sockets are visually examined, and all welds are liquid-penetrant tested, following ASME Code, Subsection NG 5233.

The MNOP was determined to be 74 psig in SAR Section 3.4.4 "Thermal Evaluation." The applicant conservatively used a MNOP of 80 psig in the cask pressure and drop analyses. The cask containment boundary shall be pressure-tested before first use to 150 percent of the MNOP, per the requirements of 10 CFR 71.85(b), to verify structural integrity. The containment vessel will be tested to 120 psig, which is 150 percent of 80 psig, and is conducted in accordance with ASME Code Section III, Division 1, Subsection NB-6000. Accessible weld and material inspections will be performed during the pressure hold to detect leakage and pressure decay and to verify the maintenance of structural integrity and the absence of any permanent deformations.

The applicant also specifies acceptance testing of the neutron shield. The details of the testing are proprietary and are specified within the SAR. The NRC staff reviewed the testing specifications and acceptance criteria and determined them to be acceptable.

8.2.4.4 Leakage Tests

Fabrication leakage tests shall be performed on all containment components, including the O-ring seals, to verify that the containment boundary leakage rate does not exceed the maximum allowable design leakage rate of 1×10^{-7} std-cm^3/sec. The fabrication leakage tests shall have a test sensitivity equal to or greater than 5×10^{-8} std-cm^3/sec. The acceptance criterion specified in the test procedures are consistent with the containment evaluations.

The DU cavity is also tested for leakage through the cask cavity liner. A leakage rate of less than 1×10^{-7} std-cm^3/sec, with a test sensitivity of 5×10^{-8} std-cm^3/sec, is acceptable to show that the cavity liner is leak tight.

8.2.4.5 Component Tests

Tests are performed to verify the nominal crush strength of the aluminum honeycomb to be used in the impact limiters. Qualification tests of the honeycomb are performed to establish the temperature effects on the crush strength of each honeycomb type, and verifications tests of the crush strength of each honeycomb production lot are performed. The NRC staff has reviewed these tests and finds reasonable assurance that they will adequately demonstrate the performance of the aluminum honeycomb.

8.2.4.6 Shielding Tests

The integrity of the cask shielding will be determined during cask fabrication. The DU procurement specification requires a gamma scan to ensure there are no discontinuities and specifies a source strength, scanning rate, and grid spacing to provide 100 percent inspection coverage. Specifications are provided of the supplier to certify that the neutron shield material meets the minimum requirements specified in the shielding analysis. GA will perform an independent chemical analysis of the neutron shield for verification. GA also specifies detailed first-use gamma and neutron dose readings, scaled up to the design basis fuel source, to ensure that the dose readings will be below the limits of 10 CFR 71.47.

8.2.4.7 Neutron Absorber Tests

The B_4C criticality control pellets inserted into drilled holes within the FSS are tested for B-10 enrichment and total weight. The full extent of these specifications was described in Chapter 6 of this evaluation and were determined to be acceptable.

8.2.4.8 Thermal Tests

The first GA-4 cask shall be subjected to a thermal acceptance test, to verify the heat rejection capability of the packaging. The test will be conducted in the horizontal (transport) condition, with a heat source and dimensions approximating the design basis thermal spent fuel contents. The cask will be backfilled with air rather than helium, and an insulated lid will be used, instead of the closure, to allow penetrations for temperature instrumentation. The results will be correlated with the analytical predictions presented in the thermal evaluation. The NRC 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 GA-4 cask. The maintenance program includes: (1) annual testing of the neutron shielding, as discussed previously in SER Section 8.2.4.3; (2) annual containment system periodic verification leakage tests; (3) containment system assembly verification leakage tests, before shipment; (4) replacement of containment boundary O-ring seals after third use and before the annual leakage testing; (5) annual checks on the material condition of the neutron shield; and (6) visual inspection of various package components before loading and shipment.

8.3.4.1 Structural and Pressure Tests

The neutron shield will be tested annually, as previously evaluated in SER Section 8.2.4.3. Other than the tests required before first use, no other structural or pressure tests are necessary to ensure continued performance of the packaging.

8.3.4.2 Leakage Tests

The containment boundary O-ring seals are to be replaced after their third use and before the containment system's annual verification leakage test. The O-ring seals shall be tested to show a leak rate of less than 1×10^{-7} std-cm^3/sec. The leak test shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

If a containment component is replaced or repaired, the affected portion shall be tested to show a leak rate of less than 1×10^{-7} std-cm^3/sec. The leak test shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

8.3.4.3 Component Tests

All fasteners and threaded inserts are visually inspected, and replaced if necessary, before each shipment. The closure bolts and impact limiter bolts shall be replaced every 20 years. The impact limiters are visually inspected annually and before each shipment, to ensure that they are in an unimpaired physical condition. Lifting and tie-down trunnions are visually receipt inspected, including the trunnion wear surfaces.

8.3.4.4 Neutron Absorber Tests

After the initial fabrication inspections and verifications, no further special maintenance is required.

8.3.4.5 Thermal Tests

Before first use, each package will undergo a thermal acceptance test to verify that its heat rejection capabilities are consistent with the thermal analysis. The cask design and maintenance provisions prevent deterioration of the heat transfer mechanisms over time. No special further testing and maintenance are required.

Evaluation Findings

The NRC staff has reviewed the identification of the codes, standards, and provisions of the QA program applicable to maintenance of the packaging and found reasonable assurance that the requirements specified in 10 CFR 71.31(c) and 10 CFR 71.37 (b) will be met.

The NRC staff has reviewed the description of the routine determinations for package use before transport and found reasonable assurance that the requirements of 10 CFR 71.87(b) and 10 CFR 71.87(g) will be met.

CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. GA-4 package meets the requirements of Part 71.

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CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. GA-4 package meets the requirements of Part 71.

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CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, we have concluded that the Model No. GA-4 package meets the requirements of Part 71.

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