January 21, 2010

Mr. Charles J. Temus
Project Manager
AREVA Federal Services, LLC
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402-3526

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9341 FOR THE MODEL NO. BRR PACKAGE

Dear Mr. Temus:

As requested by your application dated March 25, 2009, as supplemented August 6 and November 5, 2009, enclosed is Certificate of Compliance No. 9341, Revision No. 0, for the Model No. BRR package. The staff's Safety Evaluation Report is also enclosed.

AREVA is registered as user of the package under the general license provisions of 10 CFR 71.17. This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me or Chris Staab of my staff at (301) 492-3321.

Sincerely,

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

Docket No. 71-9341
TAC No. L24332

Enclosures:
1. Certificate of Compliance No. 9341, Rev. No. 0
2. Safety Evaluation Report

cc w/encls.: R. Boyle, Department of Transportation
J. Shuler, Department of Energy

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SAFETY EVALUATION REPORT

Model No. BRR Package
Docket No. 71-9341
Certificate of Compliance No. 9341
Revision 0

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Safety Evaluation Report
Model No. BRR Package
Docket No. 71-9341
Certificate of Compliance No. 9341
Revision 0

SUMMARY

By application dated March 25, 2009, as supplemented August 6 and November 5, 2009, AREVA Federal Services, LLC (AREVA) submitted an application to the U.S. Nuclear Regulatory Commission for a Certificate of Compliance (CoC) for the BEA Research Reactor (BRR) package.

The package will be used to ship fuel from university research reactor facilities, will replace the BMI-1 package, and will be a Type B(U)F-96 package. The package is authorized by the certificate for use under the general license provisions of 10 CFR 71.17. Transport by air of fissile material is not authorized. The certificate will expire on December 31, 2014.

The staff has concluded the package meets the requirements of 10 CFR Part 71 by using the guidance in NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material” and guidance in NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel.”

REFERENCES

AREVA Federal Services LLC, application dated March 25, 2009.

Supplements dated August 6, 2009 and November 5, 2009.

1.0 GENERAL INFORMATION

The BRR package is used to transport fuel elements that have been irradiated in various test and research reactors, including the University of Missouri Research Reactor (MURR), the Massachusetts Institute of Technology Nuclear Research Reactor (MITR–II), Advanced Test Reactor (ATR), and Training, Research, Isotopes, General Atomics (TRIGA) reactors.

1.1 Packaging

The fuel is primarily of two basic types: highly enriched aluminum–uranium plate fuel, and TRIGA fuel of varying enrichments. Within the package, the fuel is contained in basket structures specifically designed for each fuel type that provide for optimum heat rejection and criticality control. The packaging consists of a payload basket, a lead–shielded cask body, an upper shield plug, a closure lid, and upper and lower impact limiters. The package is of conventional design and utilizes ASTM Type 304 stainless steel as its primary structural material. The package is designed to provide leaktight containment of the radioactive contents under all Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). The BRR package may be used in a pool or hot cell environment. The cask body is provided with a drain port, and is intended for use with a drying system to ensure that water is not present during transport. The package is designed to be transported as one package per conveyance, with its longitudinal axis vertical, by highway truck or by rail in exclusive use.When loaded and prepared for transport, the BRR package is 119.5 inches long, 78 inches in diameter (over the impact limiters), and weighs 32,000 lb.

The BRR cask body is a right circular cylinder 77.1 inches long and 38 inches in diameter (not including the impact limiter attachments and the thermal shield). It is composed of upper and lower massive end structures connected by inner and outer shells. Thick lead shielding is located between the two circular shells, in the lower end structure, and in the shield plug. The payload cavity has a diameter of 16 inches and a length of 54 inches.

1.2 Contents

The BRR package may contain up to 8 irradiated MURR fuel elements, up to 11 irradiated MITR–II fuel elements, up to 8 irradiated ATR fuel elements, and up to 19 irradiated TRIGA fuel elements. Only one fuel element is allowed per basket location. Details for each fuel type are provided below.

1.2.1 MURR

The MURR fuel element may be irradiated to a maximum burnup of 180 MWD or a U-235 depletion of 30.9%. The minimum cooling time is 180 days after reactor shutdown.

Each fresh MURR element contains 775.0 ± 7.8 g U–235, enriched up to 93 wt.%. The weight percents of the remaining uranium isotopes are 1.2 wt.% U–234, 0.7 wt.% U–236, and 5.0 – 7.0 wt.% U–238. The MURR fuel element fissile material is uranium aluminide (UAlx).

Each MURR fuel element contains 24 curved fuel plates. Fuel plate 1 has the smallest radius, while fuel plate 24 has the largest radius. The fuel “meat” is a mixture of uranium metal and aluminum, while the cladding and structural materials are an aluminum alloy. The fuel plates are rolled to shape and swaged into the two fuel element side plates. The fissile material (uranium aluminide) is nominally 0.02–in thick for all 24 plates. Fuel element side plates are fabricated of ASTM B 209, aluminum alloy 6061–T6 or 6061–T651 and are approximately 0.15–in thick. The fuel plates are typically spaced with a 0.08–in gap between plates.

The MURR element overall length, including irradiation growth, is 32.75 inches. The bounding weight of one assembly is 15 lb. The maximum decay heat per fuel element is 158 W. MURR fuel element dimensions are in Table 1.1

Table 1.1

MURR - Key Fuel Dimensions	
Item	Dimension (inches)
Maximum active fuel length	24.8
Overall length	32.75
Minimum cladding thickness	0.011 - 0.015
Nominal fuel matrix thickness	0.02
Maximum number of fuel plates	24
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	782.8
Maximum enrichment (wt.%)	93.0

1.2.2 MITR-II

The MITR–II fuel element may be irradiated to a maximum burnup of 225 MWD or a U–235 depletion of 59.3%. The minimum cooling time is 930 days after reactor shutdown.

Each fresh MITR–II element contains 510.0 +3.0/-10.0 g U–235, enriched up to 93 wt.%. The weight percents of the remaining uranium isotopes are 1.2 wt.% U–234, 0.7 wt.% U–236, and 5.0 – 7.0 wt.% U–238. Like the MURR fuel element, the MITR–II fuel element fissile material is uranium aluminide (UAlx).

Each MITR–II fuel element contains 15 flat fuel plates. The fuel plates are fabricated and swaged into the two fuel element side plates. The fuel "meat" is a mixture of uranium metal and aluminum, while the cladding and structural materials are an aluminum alloy. The fissile material (uranium aluminide) is nominally 0.03–in thick and the cladding is nominally 0.025–in thick. Fuel element side plates are fabricated of ASTM B 209, aluminum alloy 6061–T6 and are approximately 0.19–in thick. The fuel plates are nominally 0.08 inches apart.

The MITR–II element overall length, including irradiation growth, is 26.52 inches. The bounding weight of one assembly is 10 lb. The maximum decay heat per assembly is 30 W. MITR–II fuel element dimensions are in Table 1.2.

Table 1.2

MITR-II - Key Fuel Dimensions	
Item	Dimension (inches)
Maximum active fuel length	22.76
Overall length	26.52
Minimum cladding thickness	0.017
Nominal fuel matrix thickness	0.03
Maximum fuel matrix width	2.171
Maximum number of fuel plates	15
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	513

1.2.3 ATR

The ATR fuel element may be irradiated to a maximum burnup of 480 MWD or a U–235 depletion of 58.6%. The minimum cooling time is 1,670 days (4.6 years) after reactor shutdown.

There are two general classes of ATR fuel element, XA and YA. The XA fuel element has a fresh fuel loading of 1,075 ± 10 g U–235. The weight percents of the remaining uranium isotopes are 1.2 wt.% U–234 (max), 0.7 wt.% U–236 (max), and 5.0 – 7.0 wt.% U–238. Like the MURR and MITR–II fuel elements, the fuel element fissile material is uranium aluminide (UAlx).

The XA fuel element is further subdivided into fuel element types 7F, 7NB, 7NBH. In the 7F fuel element, all 19 fuel plates are loaded with enriched uranium in an aluminum matrix with the eight outer plates (1 through 4 and 16 through 19) containing boron as a burnable poison. The fuel element with the greatest reactivity is the 7NB that contains no burnable poison. The 7NBH fuel element is similar to the 7NB fuel element except that it contains one or two borated plates. The YA fuel element is identical to the 7F fuel element except that plate 19 of the YA fuel element is an aluminum alloy plate containing neither uranium fuel nor boron burnable poison. The YA fuel element has a fresh fuel loading of 1,022.4 ± 10 g U–235. A second YA fuel element design (YA–M) has the side plate width reduced by 15 mils.

The ATR fuel elements contain 19 curved fuel plates. Note that an intact ATR fuel element has end boxes although these end boxes are removed prior to insertion in the BRR package. The fuel plates are rolled to shape and swaged into the two fuel element side plates. Fuel plate 1 has the smallest radius, while fuel plate 19 has the largest radius. The fissile material (uranium aluminide) is nominally 0.02–in thick for all 19 plates. Fuel element side plates are fabricated of ASTM B 209, aluminum alloy 6061–T6 or 6061–T651 and are approximately 0.19–in thick. The fuel plates are typically spaced with a 0.08–in gap between plates. The ATR element overall length, after removal of the end box structures, 51.0 inches max.

The bounding weight of one assembly is 25 lb. The maximum decay heat per assembly is 30 W. ATR fuel element dimensions are in Table 1.3.

Table 1.3

ATR - Key Fuel Dimensions	
Item	Dimension (inches)
Maximum active fuel length	48.77
Overall length	51
Minimum cladding thickness	0.03/0.015/0.04
Maximum fuel matrix thickness	0.02
Maximum number of fuel plates	19
Fuel matrix	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	1,085

1.2.4 TRIGA

Many different types of TRIGA fuel elements have been fabricated over the past several decades. TRIGA fuel elements utilize a zirconium hydride fuel matrix. TRIGA fuel element dimensions are in Table 1.4. The BRR package is limited to five specific TRIGA fuel types:

- 8 wt% U - U/Zr with uranium aluminum clad element (General Atomics catalog number 101).
- 8.5 wt% U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 103).
- 8.5 wt% U - U/Zr with uranium stainless steel clad element, high enriched uranium (General Atomics catalog number 109).

This fuel element is sometimes referred to in the literature as a Fuel Life Improvement Program (FLIP) element.

- 20 wt% U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 117). This fuel element is sometimes referred to in the literature as a FLIP–LEU–I element.

- 8.5 wt% U - U/Zr with uranium stainless steel clad element, instrumented (General Atomics catalog number 203).

Table 1.4

TRIGA - Fresh Fuel Characteristics					
Parameter	GA Cat. # 101	GA Cat. # 103	GA Cat. # 109	GA Cat. # 117	GA Cat. # 203
Active Fuel Length (in)	14	15	15	15	15
Fuel Pellet OD (in)	1.41	1.44	1.44	1.44	1.44
Overall Element Length (in)	28.37	28.9	28.9	29.68	45.25
Cladding OD (in)	1.48	1.48	1.48	1.48	1.48
Cladding Thickness (in)	0.03	0.02	0.02	0.02	0.02
Graphite Reflector Length Top/Bottom (in)	4.0 / 4.0	2.6 / 3.7	2.6 / 3.7	2.6 / 3.7	3.1 / 3.4
Zr Fuel Matrix Mass (g)	2,070	2,088	2,060	2,060	2,088
U-235 (g)	36	39	137	101	39

The maximum length of an element, including irradiation growth, is 45.50 inches. Non-instrumented fuel elements are somewhat shorter. For all fuel elements, spacers are utilized within the TRIGA baskets.

The two FLIP elements have significantly higher U–235 loadings and hence much larger burnups and longer cooling times. The bounding weight of any TRIGA fuel element is 10 lb. The maximum decay heat per element is 20 W. TRIGA fuel parameters are in Table 1.5.

Table 1.5

TRIGA - Fuel Parameters			
Fuel Type	Maximum U-235 depletion (%)	Maximum Burnup (MWD/MTU)	Minimum Decay Time
GA Cat. # 101	22.42	36,953	28 days
GA Cat. # 103/203	20.72	34,111	28 days
GA Cat. # 109	59.74	339,368	1 year
GA Cat. # 117	43.81	75,415	1 year

1.3 Criticality Safety Index

The criticality safety index for the BRR package is zero, as an unlimited number of packages will remain subcritical under the procedures specified in 10 CFR 71.59(a).

1.4 Drawings

The packaging is constructed in accordance with AREVA Federal Services LLC drawings:

- 1910–01–01–SAR, BRR Package Assembly SAR Drawing, Rev. 2, 4 sheets

- 1910–01–02–SAR, BRR Package Impact Limiter SAR Drawing, Rev. 0, 2 sheets
- 1910–01–03–SAR, BRR Package Fuel Baskets SAR Drawing, Rev. 2, 3 sheets

2.0 STRUCTURAL AND MATERIALS EVALUATION

The objective of this review is to verify that the structural and materials performance of the package is adequately evaluated to meet the requirements of 10 CFR Part 71, including the tests and conditions specified under NCT and HAC.

2.1 Structural Design

2.1.1 Description of Structural Design

The BRR transportation package is made of three principal structural components: the cask body, the basket, and the impact limiters.

Cask Body: The cask body, as a right circular cylinder about 77-inches long and 38 inches in diameter, is composed of heavy upper and lower end castings connected by inner and outer shells. The cast-in-place lead shielding fills the annulus between the shells. Together with the removable 11.2-inch thick shield plug under the closure lid, the cask body assembly provides a 54-inch long by 16-inch diameter payload cavity. Drawing 1910-01-01-SAR of the application delineates structural design details, including stainless steel materials grades, for the cask body assembly. As presented in Section No. 1.2.2.1 of the Safety Analysis Report (SAR), a set of eight receptacles are attached to the outer shell at each end of the body to serve as impact limiter attachments. The 1-inch thick inner shell, the end castings, the closure lid, and the vent and drain ports constitute the metal part of the package containment boundary.

Basket: Drawing 1910-01-03-SAR of the application provides structural details for the four basket designs for shipping the MURR, MITR-II, ART, and TRIGA fuel. The Type 304 stainless steel basket weldments, which are made with plates, bars, pipes, and tubes, feature a number of cavities that fit the size and shape of the fuel. Each basket is 53.45-inches long and 15.36 inches in diameter to facilitate its emplacement in the cask body payload cavity.

Impact Limiters: Drawing 1910-01-02-SAR of the application depicts design details for the two identical impact limiters measuring 78 inches in diameter and 34.6 inches long overall with a conical section 15 inches long towards the outer end. The impact limiters are fabricated with pour-in-place polyurethane foam encased in the stainless steel shells, ¼-inch and ½-inch thick for the respective external and internal faces. They are attached at two ends of the cask body, each by means of eight blade-to-receptacle assemblages.

2.1.2 Design Criteria

The applicant demonstrates structural capabilities of the package primarily by tests and analyses. The ½-scale packaging drop tests are used to evaluate impact limiter performance. They are also considered together with analytical predictions to establish bounding deceleration g-loads for the package structural components evaluation by analysis. Section No. 2.1.2 of the SAR summarizes the structural design criteria for the package. Section No. 2.1.4 of the SAR identifies codes and standards as well as load combinations for package design. These design criteria are reviewed as follows.

Codes and Standards: Section No. 2.1.4 of the SAR identifies the applicable codes and standards for the structural design of the packaging. This includes ASME Code Division 1, Section III, Subsections NB and NG for the containment system and fuel baskets, respectively. This conforms to the NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," guidance for Category I shipping containers. The Subsection NB stress criteria are also considered for the non-containment outer shells, which is conservative. As noted in Section No. 2.1.2.3.3 of the SAR, the shell buckling analysis is performed using the methodology of ASME Code Case N-284-2.

Structural Performance and Miscellaneous Criteria: Table No. 2.1-1 of the SAR lists the stress allowables in the linear elastic analysis for demonstrating structural acceptability of the cask components under the NCT and HAC. Section No. 2.1.2.2 of the SAR notes that components of the impact limiters, including the stainless steel shells, energy-absorbing foam, and blade-to-receptacle attachments are expected to undergo permanent deformations under NCT and HAC. Additional impact limiter performance criteria include preventing hard contact of a rigid part of the cask with the ground due to excessive deformation of the foam. Miscellaneous structural performance criteria for brittle fracture and fatigue are described in Section Nos. 2.1.2.3.1 and 2.1.2.3.2 of the SAR, respectively.

Load Combinations: Section No. 2.1.4 of the SAR notes that load cases are applied and combined per the Regulatory Guide 7.8 guidance. The eleven load combination cases, as listed in Table No. 2.12.2-1 of the SAR, involve pressure, thermal, and free-drop impact loads.
This meets the intent of the NCT and HAC load combination provisions.

2.1.3 Fabrication and Examination

Section No. 2.3.1 of the SAR provides that the containment shell and outer shell fabrications shall comply with the tolerance requirements of the ASME Code, Subsection NE, Article NE-4220. To justify the departure from the Subsection NB standards, per NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," Article NE-4220 is selected because the package cylindrical shells are verified for buckling performance using ASME Code Case N-284-2, which is based on the fabrication requirements of paragraph NE-4222. This is acceptable.

Section No. 2.3.2 of the SAR provides that each of the materials performing significant safety function must meet the ASTM specifications delineated in the licensing drawings. The requirements and acceptance criteria for installation, inspection, and testing of the closed-cell, polyurethane foam for the impact limiters are presented in Section No. 8.1.5.1 of the SAR.

2.1.4 General Considerations for Package Structural Analysis

The applicant performs structural analyses of the package using both hand calculation and finite element modeling. The former approach by equations and formulas is used primarily for evaluating package lifting, stresses in the closure bolts and the baskets, as well as buckling interaction equations check for cylindrical shell components of the package. In the following, the staff reviews general considerations for structural analysis by the finite element modeling for the cask body stress analysis for a variety of loadings, including the NCT and HAC free drops, and load combinations thereof.

Finite Element Analysis Code and Analysis Model: As noted in Section No. 2.12.4.1 of the SAR, the general-purpose finite element analysis code, ANSYS, Revision 11.0 is used to perform structural analysis of the cask body. The half-symmetry cask body analysis model, which is constructed exclusively with the SOLID95, 20-node brick elements, includes the massive upper and lower end structures, the inner shell, and the outer shell. The free-drop inertia effects of the non-model entities, such as the lead shielding and closure lid, are simulated, however, with equivalent pressure loads. Section 2.12.4.2 of the SAR delineates modeling of the thermal and free drop impact loads. Section 2.12.4.3 of the SAR notes analysis details, including selection of material modeling properties and use of the at-temperature stress allowables. Section 2.12.4.4 of the SAR summarizes the loading conditions and load combinations analyzed. For the stress performance evaluation, stress margins of safety, each defined as the ratio of the allowable and the calculated stress minus unity, are reported. A margin of safety greater than zero is considered acceptable.

The staff reviewed the package structural design including the design criteria and structural evaluation approaches, and concludes that it is described in accordance with the NUREG-1609 guidelines and is, therefore, acceptable.

2.2 Weights and Centers of Gravity

The maximum gross weight of the package is 32,000 lb. Table No. 2-1.2 of the SAR summarizes information on weights of package components and corresponding centers of gravity for the MURR, MITR-II, ATR, and TRIGA fuel types with applicable baskets. Locations of the package centers of gravity, with individual baskets and contents, are calculated to range from 38.6” to 38.7” from the cask body base.

2.3 Material Evaluation

2.3.1 BRR Material Properties and Specifications

Materials used for fabrication of the BRR package can be organized into the following categories and material types (in parenthesis): 1) structural (Type 304, stainless steel (SS)); 2) impact limiter (polyurethane foam); 3) shielding (lead); and 4) payload (alloy 6061-T6/6061-0 aluminum, uranium-aluminide (UAlx), and uranium-zirconium hydride (UZrH)). The staff reviewed the information presented in the BRR package assembly drawings and found it to be consistent with the material specifications listed in Table Nos. 2.2-1 through Table 2.2-6 of the SAR.

2.3.2 BRR Structural Containment Materials

The BRR package is made up of the following structural components: a lead shielded cask body (i.e., inner/outer SS shells welded to cast end structures), a payload basket (specific to fuel type), upper shield plug (removable), a closure lid, closure bolts (twelve) and impact limiters (polyurethane foam filled). The containment system is classified as ASME Code, Section III, Division 1, Subsection NB.

Materials used in fabrication of the structural components are solely Type 304 austenitic stainless steel (SS) as specified by various American Society of Testing Materials (ASTM) specifications. Specifications and temperature dependent mechanical properties, including yield strength, tensile strength, allowable strength, modulus of elasticity, and coefficient of thermal expansion conform to ASME Code, Section II, Part D, and are presented in Table 2.2-1 through Table 2.2-6 of the application. Section No. 2.2.1 of the SAR provides density and Poisson's ratio values used for SS and lead. The temperature range covered is from -40ºF to 800ºF. The staff reviewed the materials selected and determined that they are acceptable.

Material specification ASTM A276 specifies Type 304 bar to be furnished in several conditions, such as Condition A-annealed, Condition B-relatively severe cold work and Condition S-strain hardened by relatively light cold work. The applicant states that any of these conditions may be used and the staff concluded that these conditions will not affect the bar stocks function as a lifting point for the shield plug.

Lid closure bolts of Grade L43 (Nickel-Chromium-Molybdenum) material are plated with nickel in accordance with specification MIL-DTL-26074, Revision F, Class 1, Grade B. The staff notes this is a canceled specification. The applicant stated that to ensure an active specification is also approved for use the drawing list of materials (note 18) has been revised to add the option to plate the bolts in accordance with a currently supported specification which is equivalent to the military specification given. The staff finds that revising the BRR package assembly drawing list of materials as stated above is acceptable.

2.3.3 BRR Non-Structural Containment Material

Two O-ring seals are located and retained in dovetail grooves machined in the closure lid with the inner O-ring serving as the containment seal and the outer as the test seal. The O-ring seals (including sealing washers) are made from butyl rubber (3/8-inch diameter) to Compound No. R0405-70 meeting classification ASTM D2000 M4AA 710 A13 B13 F17 F48 Z Trace Element. The staff reviewed the material selected and determined that it is acceptable.

The closure lid vent port and the lower body drain port are designed and tested to ensure leak-tight sealing integrity. Both utilize brass (ASTM B16) bolts to compress butyl washers to seal their respective ports and are protected by identical brass dust covers. This brass (free-cutting) is a copper-zinc alloy with an HO2 or half-hard temper produced by cold work. Lead has been added for improved machining. The staff reviewed the material selected and determined that it is acceptable.

2.3.4 BRR Impact Limiter Material

Impact limiters are constructed of Type 304 SS sheets welded together forming a void which is filled (poured in place) with rigid, closed-cell (non-water absorbent) polyurethane foam (9 lb/ft^3). The foam is free of halogens and chlorides. The applicant states in SAR Section No. 3.5.4 that the foam is rigid polyurethane foam and the FR-3700 formulation is specially designed (proprietary). Section No. 8.1.5 of the SAR states the foam supplier shall certify that the polyurethane foam constituents have been properly stored prior to use and that the polyurethane foam constituents have been used within their shelf life. Section No. 8.1.5.1 of the SAR presents the details of acceptance tests for this material. The nominal, room-temperature crush properties of the polyurethane foam are given in Table No. 2.2-6 of the SAR. Properties for both "parallel to rise" and "perpendicular to rise" are given. The "rise" direction is parallel to the force of gravity during solidification, and is oriented to be parallel to the cylindrical axis of the impact limiters. The staff reviewed the properties of the foam provided by the manufacturer in terms of shock, thermal protection and resistance to water adsorption and determined it was acceptable.

2.3.5 BRR Shielding Material

ASTM B29 lead shielding is of the refined chemical-copper grade intended for applications requiring corrosion protection and formability. An angle iron is utilized to prevent lead, poured in place, from leaking through the final closure weld. General note 27 of SAR drawing 1910-01-01-SAR states that to prevent cross contamination a region of lead (1/2-inch max.) may be removed adjacent to the weld joint and may be filled with ceramic rope or a weld backing bar may be used.

The applicant states that the removed area of lead is minimal relative to the cask size and thickness of the bottom shield. In addition, the removed area is on the corner of the rectangular cross section of the shield, where the pathway of gamma rays through the lead is naturally the longest. The applicant also stated that the integrity of the weld is ensured, as is the case for all single-sided complete joint penetration welds, by means of the specified nondestructive examination (NDE) methods and the required weld procedure qualification used to make the weld. In addition, the general note provides for an optional weld joint location and design; other weld joint locations may be used, such as placing the weld groove in the cask body rather than in the bottom plate.

The staff concludes that methods used to ensure external radiation standards, weld, and shielding integrity are acceptable.

2.3.6 BRR Payload Material

The MURR, MITR-II, and ATR fuel element fissile material is fabricated of uranium-aluminide (UAlx). Cladding and fuel element structural material is fabricated from Al alloys 6061-T6, 6061-T651, or 6061-0 to ASTM B209.

The TRIGA fuel is limited to five specific fuel types (General Atomics catalog number): 101, 103, 109, 117, and 203. TRIGA fuel elements utilize a uranium-zirconium hydride (UZrH) fuel matrix. The staff notes that Zr alloys absorb hydrogen which may lead to embrittlement when used as cladding material. Also, the formation of ZrH accompanied by a volume increase can degrade the mechanical properties of Zr alloy cladding and lead to cracking. Cladding materials for the various TRIGA fuel types are Al and SS. The staff reviewed the materials selected and determined that they are acceptable.

2.3.7 Chemical, Galvanic, or other reactions

Section No. 2.2.2 of the SAR discusses chemical, galvanic, or other reactions expected for the materials of construction of the BRR package. Galvanic or dissimilar metal corrosion requires dissimilar metals in contact and both metals immersed in the same electrically conducting solution simultaneously. Aluminum will typically act as an anode when in contact with other metals and may begin to corrode. However, when in contact with SS, aluminum alloys are known to show slight electrolytic attack in atmosphere or some stagnant solutions due to polarization. Therefore, the staff finds that the payload materials are acceptable and that the possibility of galvanic corrosion due to dissimilar metals in contact is low.

Section Nos. 2.1 and 2.2 of the SAR discusses brittle fracture failure and the effects of radiation on BRR package materials, respectively. The staff notes that when SS is exposed to a fluence of 10^{21} neutrons/cm^2, its ultimate (tensile) and yield strength increases however its ductility (elongation) decreases by about one third. Also, the temperature at which metal changes from ductile to brittle fracture (i.e., the Nil-Ductility Transition (NDT) temperature) is often increased. Therefore, the chance of brittle fracture increases. The neutron fluence expected from the BRR package is many orders of magnitude lower than the stated known fluence expected to cause changes of material mechanical properties. Also, austenitic SS does not exhibit an impact ductile/brittle transition, but a progressive reduction in charpy impact values as the temperature is lowered and is suitable in situations where sub-zero (ambient) temperatures (typically down to -40°C) are expected as a result of its face centered cubic atomic structure. The staff concluded no degradation of the austenitic SS mechanical properties are expected as a result of BRR package neutron fluence.

The BRR package is fabricated with butyl O-rings and polyurethane foam. Butyl O-rings exposed to 1 x 10^6 rads show slight effects and can endure many loading cycles. Polyurethane foam can resist radiation exposure up to 2 x 10^8 rads with no effect on density or crush strength. The low-halogen, Ni-based, nuclear grade lubricant used for threaded fasteners and vacuum grease used for O-ring seals are both known to be suitable for use in a radiation environment. Also, the required replacement frequency of the O-rings is much shorter than the time required for radiation exposure to degrade its material properties. The staff finds that these materials are acceptable for use.

The staff determined that the requirements of 10 CFR 71.43(d) have been met.

Section No. 8.0 of the SAR discusses the BRR package acceptance tests and maintenance program. Each package will be subjected to visual inspection and measurements, weld examinations, structural and pressure tests, fabrication leakage rate tests, component and materials tests, shielding integrity tests, and thermal tests. The staff finds that the required inspections, repairs and tests of fasteners, containment sealing surfaces, seals and impact limiters performed as part of the component and material tests to be acceptable.

2.4 General Standard for All Packages (10 CFR 71.43)

2.4.1 Minimum Package Size

The minimum dimension of the BRR package is approximately 38.5 inches. It is greater than the minimum overall dimension of 4 inches, which meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamper-Indicating Features

A temper-indicating seal is made by passing a lock wire through a hole in one of the upper impact limiter blade-to-receptacle attachment assemblages. The package cannot be opened by an unauthorized person without damaging the seal, thus providing evidence of possible tampering. This satisfies the tamper-proof requirements of 10 CFR 71.43(b).

2.4.3 Positive Closure

The BRR package cannot be opened unintentionally. The impact limiters, which are each secured with eight, 1-inch diameter ball lock pins, fully conceal all cask openings. This satisfies the requirements of 10 CFR 71.43(e) on positive closure.

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

2.5.1 Lifting Devices

The BRR package is lifted, using two cables, from four lift points located in the upper end structure, where lifting devices are installed in threaded holes. Section 2.5.1 of the SAR notes the structural failure mode by shear tear-out of the threads. The applicant considers a bounding weight of 30,000 lb, which is greater than 27,000 lb, the weight of the loaded cask with the bottom impact limiter, to calculate the governing shear stress in the inner threads. With the minimum thread length of 1.5 inches for the 1-8 UNC-2B bolts and three times the bounding weight of the package, the calculated stress margin of safety is 1.2, which is positive and acceptable. This demonstrates that the lifting devices are designed with a minimum safety factor of three against the material yield strength. Additionally, the application notes that, in the case of lifting overload, the devices will strip out of the parent material without damage to the cask. Thus, the lifting devices satisfy the load capacity and excessive load cask protection requirements of 10 CFR 71.45(a).

2.5.2 Tie-Down Devices

During transport, the BRR package rests on a steel pallet and is held down to the pallet by means of a steel frame, which rests on top of the upper impact limiter. The steel “tie-down” frame is, in turn, attached by wire ropes or equivalent to the conveyance. In this configuration, the package lifting holes are covered by the upper impact limiter and the steel tie-down frame and are rendered inoperable for tying down the package during transport. This satisfies the requirements of 10 CFR 71.45(b)(2). As noted further in the application, the package contacts only the pallet on the bottom and the steel from on the top. It has no tie-down devices as integral structural part of the package. Therefore, the staff agrees with the applicant's conclusion that no evaluation of tie-down devices is required for meeting the requirements of 10 CFR 71.45(b)(1) and (3).

2.6 Normal Conditions of Transport (10 CFR 71.71)

2.6.1 Heat

Table No. 3.1-1 of the SAR lists the maximum temperatures calculated for various locations of the package components. They are based on the maximum decay heat, an ambient temperature of 100^{o} F, and no insolation under the NCT. The corresponding maximum cask cavity pressure is calculated to be 5.2 psig. Section No. 2.6.1.1 of the SAR notes that, for structural evaluation purposes, the bounding temperature of 400^{o} F is considered for the fuel baskets and 250^{o} F for other cask body components, including the end structures, shells, and closure lids and bolts. A design pressure of 25 psig, which is significantly higher than the conservatively defined maximum operating pressure (MNOP) of 10 psig, is also used to evaluate the package structural performance.

Section No. 2.6.1.2 of the SAR evaluates differential thermal expansion (DTE) of package components for possible interference among package components. Section No. 2.6.1.2.1 of the SAR calculates reduction in radial and longitudinal gap sizes between the cask body cavity and the baskets and concludes that the thermal expansion of the baskets is not a concern.

Table No. 2.6-2 of the SAR lists the resulting minimum axial clearances ranging from 0.19 inches to 0.21 inches between the basket cavity and the four individual fuel types.

Section No. 2.6.1.2.3 of the SAR considers the lead solidification temperature at 620^{o} F to result in an estimate of the maximum sustainable interface pressure of 337 psi between the lead and cask inner shell. As presented in Section No. 2.12.4.4.2 of the SAR, a conservative, upper bound, lead shrinkage pressure of 350 psi is applied to the inner shell, together with other appropriate loadings, in the finite element analysis of the cask body. The resulting stress margin of safety is 7.65, which is positive, and is acceptable.

Section No. 2.6.1.3 of the SAR presents the stress evaluation, using finite element models, for the cask body subject to the pressure and thermal loadings. These include the maximum design pressure of 25 psi on the inner shell, cask body temperature distribution, and lead shrinkage pressure of 350 psi on the inner shell. The evaluation demonstrates large, acceptable stress margins of safety.

As stated in Section No. 2.6.1.5 of the SAR, the closure lid is sized such that support against lateral inertia loads, in the plane of the lid, is obtained from the fit

between the lid and the cask opening. This configuration prevents any shear loading from being imposed on the closure bolts. Considering the bounding cask body temperature of 250° F, the applicant computes an average axial bolt stress of 8,747psi, based on a temperature change of 180° F from the ambient temperature of 70° F. This, together with the bolt preload stress of 19,200 psi, results in a tensile stress margin of safety of 0.41 for DTE consideration. When the residual torsional stress of 9,514 psi is combined with the bolt tensile stresses, the corresponding stress intensity margin of safety is 0.75.
These stress margins of safety are positive and are, therefore, acceptable.

The staff reviewed structural performance of the package under the heat condition and concludes that the DTE and stress effects have properly been evaluated. Thus, the requirements of 10 CFR 71.71(c)(1) are satisfied.

2.6.2 Cold

Section No. 2.6.2 of the SAR evaluates effects of cold environment on the package performance by considering an ambient temperature of -40° F combined with zero insolation and zero decay heat. Considering the maximum lead strain of 0.671%, the applicant determines the maximum sustainable interface pressure of 787 psi between the lead shield and cask body inner shell. The calculated inner shell membrane stress of 6,690 psi corresponds to a stress margin of safety of 1.99, which is positive and acceptable.

A reduction of closure bolt preload will occur at the NCT cold condition due to the DTE between the bolt and the lid materials. At the ambient temperature of -40° F, the preload reduction is calculated to be 4,890 lb. This amounts to only a 31% reduction of the bolt preload, which ensures a large positive preload force to remain at the NCT cold temperature of -40° F.

On the basis of the above, the staff has reasonable assurance to conclude that the package structural performance is adequate for meeting the cold condition requirements of 10 CFR 71.71(c)(2).

2.6.3 Reduced External Pressure

Section No. 2.6.3 of the SAR notes that the effect of reduced external pressure of 3.5 psia is considered negligible compared to other design loadings. This is in recognition that the 25 psig internal design pressure, which bounds the conditions and tests of reduced external pressure, has been used in structural analysis to demonstrate structural integrity of the cask inner shell containment boundary. Thus, the requirements of 10 CFR 71.71(c)(3) for reduced external pressure are satisfied.

2.6.4 Increased External Pressure

For an increased external pressure of 20 psia, Section No. 2.6.4 of the SAR computes a net external differential gas pressure of 7.8 psi for the cask inner shell. This differential pressure is based on an ambient temperature of -20° F, no insolation, no decay heat, and minimum internal pressure of 14.7 psia. By also considering the fabrication stress of 787 psi caused by lead shrinkage, the applicant uses an upper bound pressure of 800 psi, which is greater than the combined pressure of 794.8 psi (787 +7.8 = 794.8), to evaluate structural integrity of the inner shell. The resulting stress intensity of 6,632 psi corresponds to a margin of safety of 2.02. Table Nos. 2.6-3 and 2.6-4 of the SAR summarize the analysis parameters and results for the buckling strength evaluation, respectively, based on the ASME Code Case N-284-2 provisions. All interaction equation checks are shown to be satisfied, including the maximum value of 0.5972, which is less than unity, and is acceptable. Thus, the increased external pressure will not affect package structural performance adversely and the requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5 Vibration

Section No. 2.6.5 of the SAR uses the ANSI N14.23 truck transport vibration standards for the light packages tie-downed to a trailer bed to select peak acceleration inputs for evaluating the BRR package. Considering the peak vertical acceleration of 2 g for the package transported in the upright configuration, the applicant calculates the maximum stress of 242.6 psi in the closure lid. This stress is much below the endurance limit of 13,119 psi, thereby demonstrating that the vibration load is not expected to cause structural damages to the package. Thus the requirements of 10 CFR 71.71(c)(5) are satisfied.

2.6.6 Water Spray

Section No. 2.6.7 of the SAR notes and the staff agrees that the materials of construction used in the BRR package are not affected by the water spray test identified in 10 CFR 71.71(c)(6).

2.6.7 2-Ft Free Drop

Section No. 2.6.7 of the SAR considers drop orientations for which the maximum damage is expected of the package. This results in selecting a bounding deceleration of 40 g, in both the end- and side-drop orientations, for evaluating structural performance of the package subject to a NTC 2-ft free drop.

End Drop: For the NTC end free drops, the applicant evaluates stress margins of safety for the cask body, using the finite element analysis results for four loading conditions. hey are: (1) the bottom-down end drop, (2) the bottom-down end drop with thermal, (3) the top-down end drop, and (4) the top-down end drop with thermal. For the primary membrane stress category, the margins are determined to be 0.32 and 0.51 for the bottom-down and top-down end drops, respectively. With the thermal effect also combined for drop conditions, at higher stresses, the corresponding margins of safety for the primary-plus-secondary stress intensity category are 3.11 and 3.53.

The SAR notes that the bolt load in the NTC free drop event is governed by the preload-plus-thermal load. By referring to the closure bolt stress evaluation for the accident condition of 120 g, which demonstrates that bolt preload remains to be effective, the staff agrees with the applicant's conclusion that the bolt stress margins of safety are also acceptable for the NTC free drop event.

An equivalent load of 80,000 lb, which amounts to a 40 g inertia load for the combined weight of the lid and package internals, is considered together with the 25 psig internal pressure for evaluating the closure lid structural integrity. The margin of safety is determined to be 2.22 for the membrane-plus-bending stress intensity category. For weld stress in the lower closure plate, the margin of safety is 0.83 for the same stress category, using the prorated stress results associated with an inertia load of 120 g but at a lower stress allowable than that for the HAC. These stress margins of safety are positive and are acceptable.

Table Nos. 2.6-5 and 2.6-6 of the SAR summarize analysis parameters and results, respectively, for the buckling strength evaluation based on the ASME Code Case N-284-2 provisions. Considering a bounding compressive stress of 2,372 psi in the inner shell and a factor of safety of 2.00 for the NTC condition, all interaction equation checks are shown satisfied, including the maximum value of 0.2, which is less than unity, and is acceptable.

Side Drop: The applicant evaluates stress margins of safety for the cask body for two NCT loading conditions. They are, as presented in Appendix 2.12.5 of the SAR: (1) Case No. 9, NCT side drop and (2) Case No. 10, NCT side drop with thermal. For the primary membrane stress intensity category of Case No. 9, the margin of safety is determined to be 0.06. With the thermal effect also combined for the side drop, the margin of safety for the primary-plus-secondary stress intensity category is 1.64 at a higher stress allowable. These margins are positive, and are, therefore, acceptable.

On the basis of the above, the staff concludes that the package is capable of maintaining its structural integrity to meet the requirements of 10 CFR 71.71(c)(7).

2.6.8 Corner Drop

The BRR package is not a fiberboard, wood, or fissile material package and weighs more than 220 lbs. Therefore, the corner drop test does not apply, in accordance with 10 CFR 71.71(c)(8).

2.6.9 Compression

Because the BRR package weighs more than 11,000 lb, the staff agrees with the applicant’s assessment that the BRR package needs not be evaluated for meeting the 10 CFR 71.71(c)(9) requirements on compression test.

2.6.10 Penetration

Section No. 2.6.10 of the SAR notes that the impact of a 13-lb steel bar, which is hemispherically ended and dropped from a height of 40 inches, has no significant effects on the BRR package. Only slight denting of the thermal shield on the outside of the cask can occur. Thus, the staff concludes that the package needs not be evaluated explicitly for satisfying the requirements of the 10 CFR 71.71(c)(10) penetration test.

2.7 Hypothetical Accident Conditions (10 CFR 71.73)

2.7.1 30-FT Free Drop

Section No. 2.7.1 of the SAR evaluates structural adequacy of the package, by both tests and analyses, for the HAC 30-ft free drops. The ½-scale packaging drop tests are used to demonstrate performance of the impact limiters. They are also considered together with analytical predictions to establish bounding deceleration g-loads for the package structural components evaluation by analysis. Both the hand calculation and the finite element analysis method are used to determine structural margins of safety for the cask body components and basket internals.

In the following, after reviewing test performance of the impact limiters, the staff evaluates structural design margins for the cask body and baskets subject to a bounding deceleration of 120 g.

Impact Limiter Performance: The applicant used the 1/2-scale, essentially prototypical impact limiters and a dummy cask body to perform HAC drop tests. Table No. 2.12.3-2 of the SAR summarizes the test parameters, including drop orientation, puncture bar target, and impact limiter arrangement for the five 30-ft free drops and six puncture drops. The 30-foot drops were for the end, slapdown, and center of gravity over corner orientations, which involved one repeated and one confirmatory slapdown drops in finalizing design changes to the impact limiter attachments.

Section No.2.12.3.5 of the SAR presents test results, including time-history deceleration cask body response traces and photograph records of deformations and damaged attachments of the impact limiters, for the initial series of four, 30-ft free drop tests and five puncture drop tests. Section No. 2.12.3.4.2 of the SAR notes that the test articles were chilled generally between -10^{o} F and -20^{o} F. For the end drop, the peak average deceleration in the cask axial direction reads 116 g, instrumental. For the center of gravity over corner drop, the peak deceleration, in the direction perpendicular to the ground, is 117 g. In the case of the 15-degree slapdown drop, the peak deceleration at the accelerometer station is 115 g perpendicular to the ground for the primary impact and 114 g transverse to the cask axis for the secondary impact.

On overall capabilities of the impact limiters, the applicant note that, although the impact limiters were damaged with some exposure of the polyurethane foam, they remained attached to the cask body and met the performance criteria. Since damaged configurations are also considered in the thermal model for the HAC fire event analysis, the staff agrees with the applicant's conclusion that the impact limiters have been demonstrated structurally acceptable for sequential application of the HAC conditions and tests.

Bounding Cask Body Decelerations: Section No. 2.12.5 of the SAR considers at-temperature crush strengths of the polyurethane foam and uses the proprietary code CASKDROP to compute load-deflection curves for the impact limiter subject to a full range of impact incident angles from 0 to 90 degrees. With the force-deflection curves defined for different impact angles, the applicant applies the proprietary code SLAPDOWN to select a slapdown angle of 15 degree for evaluating the maximum damages expected of the package undergoing the HAC free drops. Table No.2.12.5-24 of the SAR compares the tested with the calculated maximum cask body decelerations and impact limiter crush distances for the end, slapdown, and center of gravity over corner drops. For the governing, slapdown drop, the calculated primary and secondary impacts of 71.0 g and 86.6 g are seen higher than the corresponding test results of 69.0 g and 71.8 g. Since the predicted results are consistently higher than those of the tested, there is reasonable assurance to conclude that the analytically predicted cask deceleration and impact limiter deformations are conservative.

Section No. 2.7.1.1 of the SAR provides that a bounding HAC impact of 120 g is utilized to evaluate cask body structural performance for all drop orientations. This is in recognition that, as shown in Table No. 2.12.5-24 of the SAR, all predicted and measured peak decelerations for key drop orientations are much below the bounding 120 g. As is also delineated in Table No. 2.7-3 of the SAR, by invoking a relatively high, bounding, side drop deceleration of 120 g for the cask body structural evaluation, there is no need for analyzing the cask body for the slapdown drop.

In the following, the staff reviews structural performance of the cask body and baskets under the HAC end- and side-drop decelerations of 120 g. The fuel end impact deformation is also evaluated

Cask Body - End Drop: The applicant evaluates stress margins of safety for the cask body, using the finite element analysis results, for loading conditions such as the bottom- and top-down end drops. Hand calculations are also used to estimate upper bound stress results and corresponding margins of safety for other structural components of the cask body, including the closure bolts and lid and basket internals.

As reported in Section No. 2.12.4.4.5 of the SAR for the Case No. 5, Bottom-Down End Drop, the maximum primary membrane and the membrane-plus-bending stress intensities are 22,260 psi and 43,080 psi, respectively. These linearized stresses, which are associated with the lower massive end structure, correspond to the respective stress margins of safety of 0.98 and 0.49. Similarly, at the inner shell, the minimum membrane and the membrane-plus-bending stress intensity margins of safety for the Section No. 2.12.4.4.8 of the SAR, Case No. 8, Top-Down End Drop, are 0.97 and 0.92. These margins are all positive and are, therefore, acceptable.

Section No. 2.7.1.2 of the SAR evaluates closure bolt performance. For a top-down inertia load of 120 g, the calculated tensile force is 28,140 lb in each of the twelve closure bolts. This value exceeds the bolt preload of 19,200 lb. However, the corresponding bolt stress, considering also the inner shell design pressure of 25 psi, is determined to be 47,779 psi, which is less than the at-temperature yield strength of 97,350 psi for the ASTM A320 L43 bolt. This suggests the elastic bolt behavior for meeting the NUREG-1617 performance provisions for the closure bolt assembly. Additionally, per the ASME code stress limit of 87,500 psi, at

250^{o} F, for the loading associated with the HAC, the stress margin of safety is 0.83, which is positive and is acceptable.

An equivalent load of 240,000 lb, which corresponds to a 120 g inertia load for the combined weights of the lid and package internals, is considered for evaluating the closure lid structural integrity. The margin of safety is determined to be 1.65 for the membrane-plus-bending stress category, per the NUREG/CR-6007, at-temperature stress allowable of 68,600 psi. As noted in the application, the calculated lid bending stress of 25,865 psi exceeds the yield strength of 23,700 psi by about 5%. However, recognizing that the calculated maximum stress occurs at the lid center and is also based on an upper bound inertia load, the staff agrees with the applicant's assertion that effects of this mild overstress on the containment O-ring compression capability are limited. Thus, the staff concludes that the closure lid is acceptable for meeting the intent of the NUREG-1617 guidance on structural integrity.

Tables Nos. 2.7-1 and 2.7-2 of the SAR summarize analysis parameters and results, respectively, for the buckling strength evaluation based on the ASME Code Case N-284-2 provisions. Considering a bounding compressive stress of 7,117 psi in the inner shell and a factor of safety of 1.34 for the HAC condition, all

interaction equation checks are shown satisfied, including the maximum value of 0.4024, which is less than unity, and is acceptable.

The structural performance evaluation for the cask body includes also the prevailing lower closure plate weld stress, the shield plug shell stress, and the gap between the top of the lead cavity and the top of the lead material resulting from lead slump. The stress margins of safety are shown to be positive and are, therefore, acceptable. The effect of the lead slump is evaluated in Section No. 5 of this Safety Evaluation Report (SER).

Cask Body – Side Drop: Load Case 11 in Section No. 2.12.4.4.11 of the SAR evaluates stress margins for a side-drop deceleration of 120 g. The maximum primary membrane and primary membrane-plus-bending stress intensities are 16,330 psi and 51,990 psi, respectively. The corresponding stress margins of safety are 1.75 and 0.23, which are positive and are, therefore, acceptable.

Fuel Baskets – End Drop and Side Drop: Section No. 2.12.8 of the SAR provides details of the structural evaluation of fuel baskets for shipping the MURR, MITR-II, ATR, and TRIGA fuel. For the 120 g end and side drops, each basket is analyzed for modes of failure applicable to its design, such as bending, weld shear, and buckling for the bounding weights. Table No. 2.12.8-2 of the SAR summarizes the buckling strength evaluations for the MURR, MITR-II, and TRIGA fuel baskets. All Code Case N-284-2 interaction checks are less than unity. Table No. 2.12.8-3 of the SAR lists calculated stress margins of safety, which are all positive. These results demonstrate that the BRR package fuel baskets are adequate to support the fuel in all HAC free drops.

Fuel Impact Deformation – End Drop: Section No. 2.7.1.6 of the SAR evaluates potential fuel deformations with respect to the cask or basket structures during the HAC end drop. Using an energy balancing method for the governing ATR fuel, the applicant calculates a maximum axial fuel deformation of 0.096 inches. The staff reviewed the analysis assumptions and agrees that this deformation is negligibly small. This is acceptable from a structural, shielding, or criticality perspective.

On the basis of the above, the staff concludes that the package is structurally capable of meeting the requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

The application notes and the staff agrees that the weight of the BRR package exceeds 1,100 lb and the crush test specified in 10 CFR 71.73(c)(2) does not apply.

2.7.3 Puncture

Section No. 2.7.3.1 of the SAR summarizes the five puncture drops performed on the half-scale certification test unit to demonstrate structural adequacy of the package. The tests showed that the puncture bar would neither penetrate beyond the impact limiter shell located on the flat bottom nor create a significant exposure of foam adjacent to the cask to the containment seal. The impact limiters would remain attached the cask ends. Also shown in one of the tests was that the puncture bar would not enter the impact limiter through a side impact on the limiter shell and rip open a large area. For the package thermal evaluation, the tests provided bounding impact limiter damaged configurations for fire event modeling consideration.

Section No. 2.7.3.2 of the SAR evaluates the puncture resistance of the outer surface of the cask body using Nelms' equation. The required thickness of the outer shell to resist puncture is determined to be 0.61 inches, which is far less than the outer shell thickness of 2 inches.

On the basis of the above, the staff agrees with the applicant's conclusion that the BRR package is structurally capable of meeting the 10 CFR 71.73 (c)(3) requirements on puncture tests.

2.7.4 Thermal

See Section No. 3.0 of this SER on thermal performance of the package.

2.7.5 Immersion - Fissile Material

The criticality evaluation presented in Chapter 6 of the application assumes optimum hydrogenous moderation of the contents. Thus, the staff agrees with the applicant's conclusion that the effects and consequences of water in-leakage are conservatively addressed to meet the requirements of 10 CFR 71.73(c)(5) immersion test.

2.7.6 Immersion - All Packages

As reviewed in the next section of this safety evaluation, the BRR package is structurally adequate for the deep water immersion test pressure of 290 psig. This pressure is much higher than the equivalent hydrostatic pressure of 21.7 psig associated with the 50-ft water immersion test. Therefore, the staff agrees with the applicant's conclusion that the immersion test does not need to be evaluated to satisfy the requirements of 10 CFR 71.73(c)(6).

2.7.7 Deep Water Immersion Test

Section No. 2.7.7 of the SAR adds conservatively the "cold" lead shrinkage pressure of 787 psi to the hydrostatic pressure of 290 psi for concurrent application on the cask inner shell for analyzing its buckling strength. Table Nos. 2.7-7 and 2.7-8 of the SAR summarize the analysis parameters and results, respectively, for the buckling strength evaluation based on the ASME Code Case N-284-2 provisions. For the factor of safety of 1.34 applicable to the accident condition, all interaction equation checks are shown to be satisfied, including the maximum value of 0.4286, which is less than unity and is acceptable. This demonstrates that the package can withstand the deep water immersion test without collapse, buckling, or in-leakage of water to meet the requirements of 10 CFR 71.61.

2.7.8 Summary of Damage

Section No. 2.7.8 of the SAR summaries analysis results for the cask body and basket components. These include acceptable stress margins of safety for the structural components and adequate buckling strength interaction equation checks for the containment boundary inner shell. As noted in the application, the impact limiter performance is demonstrated by tests of the 1/2-scale, prototypical units and a dummy cask body. The tests demonstrated that the impact limiters were capable of limiting the cask body deceleration to the design basis of 120 g applicable to all package drop orientations. The application notes that, although the impact limiters were damaged with some exposure of the foam, they remained attached to the cask body. Also noted is that the damaged configuration is included in the thermal model for the HAC fire event analysis. Hence, the BRR package is shown to perform adequately under the sequential application of the HAC free drop, puncture, and thermal tests.

2.8 Conclusion

The staff has reviewed the BRR Transport Package designed for shipment of Department of Energy owned reactor fuel proposed in Revision 0 to Certificate of Compliance No. 9341. Based on the statements and representations contained in the application, response to the staff request for additional information and the conditions given in the Certificate of Approval, the staff concludes that the package has adequately been described and evaluated to demonstrate its structural and materials capabilities to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The objective of the review of the BRR package design is to verify that the design satisfies the thermal requirements of 10 CFR Part 71 and that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC. This submittal was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section No. 3 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," as well as associated Interim Staff Guidance (ISG) documents.

3.1 Description of the Thermal Design

3.1.1 Package Design Features

The principal components of the BRR are: 1) a lead-shielded cask body, 2) a separate, removable upper shield plug, 3) a bolted closure lid, 4) upper and lower impact limiters containing polyurethane foam, and 5) various payload baskets specifically designed for each type of fuel being transported. Except for the closure bolts, the lead shielding, and the impact limiter attachment pins, the package is primarily of welded construction, using Type 304 austenitic stainless steel.

The primary heat transfer mechanisms within the BRR packaging are conduction and radiation. Heat transfer from the exterior of the packaging is via primary convection and radiation to the ambient environment. The impact limiters provide the principal thermal protection to the ends of the packaging, while a thermal shield is used to protect the portion of the packaging between the limiters from the high heat flux generated during the transient HAC fire event. The BRR design basis decay heat loadings are as follows:

- MURR fuel: 158 W maximum per element, 1264 W per basket
- MITR-II fuel: 30 W maximum per element, 330 W per basket
- ATR fuel: 30 W maximum per element, 240 W per basket
- TRIGA fuel: 20 W maximum per element, 380 W per basket

Table No. 3.1-1 of the SAR provides a summary of the package component temperatures under normal and accident conditions. The temperatures for normal conditions are based on an analytical model of the BRR package for steady-state operation with an ambient temperature of 100°F and the 10 CFR 71.71(c)(1) prescribed insolation averaged over 24 hours. The temperatures for accident conditions are based on a transient simulation using an analytical model of a damaged BRR package. Table No. 3.1-2 of the SAR presents a summary of the maximum pressures predicted under NCT and HAC conditions. The BRR package has a design maximum pressure of 25 psig.

The staff reviewed the design description of BRR thermal design and finds it acceptable. The staff reviewed the temperature and pressure design limits and calculated temperatures and pressures for the package and found them to be acceptable and consistent in the SAR.

3.1.2 Material Properties and Component Specifications

The BRR packaging is fabricated primarily of a variety of Type 304 stainless steel product forms, lead, and polyurethane foam. The payload materials include 6061-T6 and/or 6061-0 aluminum, uranium-aluminide (UAlx), and uranium-zirconium hydride (UZrH). The package application provided material thermal properties such as thermal conductivities, densities, and specific heats for all modeled components of the cask. The staff found these properties acceptable.

The materials used in the BRR packaging that are considered temperature sensitive are the lead used for the radiation shielding, the polyurethane foam used in the impact limiters, the epoxy coating used on the impact limiter exterior surfaces, the butyl rubber compound used for the containment boundary seals, and the aluminum cladding and UAlx fuel matrix used for the enclosed fuel assemblies. The other materials either have temperature limits above the maximum expected temperatures or are not considered essential to the function of the package. The staff reviewed the thermal properties used for the package analyses and determined that they were appropriate for the materials specified.

3.2 General Considerations for Thermal Evaluations

The BRR package thermal performance is analyzed by the applicant using a three-dimensional model developed in the Thermal Desktop and SINDA/FLUINT computer programs. SINDA/FLUINT is a finite difference general purpose code that solves steady-state and transient thermal problems. SINDA/FLUINT and Thermal Desktop computer programs have been validated for safety basis evaluations for nuclear related projects and the staff has found these codes acceptable for applications regarding transportation of spent fuel and other radioactive nuclear materials.

The applicant’s thermal model provides a full height, half symmetry representation of the packaging and payload components. The modeling approach permits simulation of the varying insolation loads along the length of the package, captures the various degrees of symmetry within the fuel baskets, and allows the non-symmetry conditions of the HAC free drop damage to be simulated. A separate thermal model is used to evaluate the thermal performance during NCT for each of the four potential fuel payloads. A thermal model is developed for each separate basket and geometry to consider the different heat transfer characteristics of each of the four packages. The details of the NCT thermal modeling are provided in Appendix No. 3.5.3 of the SAR.

The analytical model for HAC is a modified version of the half symmetry NCT model described in Appendix No. 3.5.3.1, Description of BRR Packaging Thermal Model for NCT Conditions, with the MURR fuel element payload. The MURR payload is selected as a basis for the HAC evaluation since its decay heat loading is more than 3 times greater than any of the other potential payloads. The principal model modifications made to the NCT thermal model to convert it to the HAC model consist of:

- modifying the impact limiter attachment thermal model to reflect the design modifications following the drop testing,
- simulating the expected package damage resulting from the HAC defined drop events,
- capturing the thermal decomposition of the polyurethane foam under HAC conditions,

- changing the package surface emissivities to reflect the assumed presence of soot and/or surface oxidization,
- assuming contact between the thermal shield and the outer shell and zero lead gap to maximize the heat flow into the package, and
- changing the package orientation from upright to horizontal to reflect its probable orientation following the HAC drop event.

The staff confirmed that the methods used for the thermal analyses were sufficiently described and provided an adequate representation of the thermal performance of the package for NCT and HAC.

3.3 Evaluation of Accessible Surface Temperatures

The applicant performed an evaluation of the package for an ambient air temperature of 100°F without insolation loads, and demonstrated that the temperatures of all exterior surfaces of the packaging are below the maximum temperature of 185 °F permitted by 10 CFR 71.43(g) for accessible surface temperature in an exclusive use shipment.

3.4 Thermal Evaluation under Normal Conditions of Transport

Table No. 3.3-1 of the SAR presents the predicted BRR package temperatures under NCT of a fully loaded MURR fuel basket dissipating 1264 W of decay heat. The analysis assumes a helium gas backfill in order to limit the peak temperature of the MURR fuel plates to 400°F or less, based on structural considerations.

The minimum temperature achieved within each of the fuel baskets would be achieved with a zero decay heat load and an ambient air temperature of -40 °F per 10 CFR 71.71(c)(2). The evaluation of this thermal condition requires no thermal calculation. Instead, all package components will eventually achieve the -40 °F temperature under steady-state conditions. As stated in Section No. 3.2.2, Technical Specifications of Components," of the SAR, the -40 °F temperature is within the allowable operating temperature range for all package components.
Assuming the backfill gas has an initial temperature of 70 °F at the time of filling and that a fill pressure of one atmosphere is used, the applicant predicted a maximum operating pressure within the cask cavity for the transport of the MURR payload of 5.2 psig. Based on this calculated pressure during NCT, the applicant set up a maximum normal operating pressure (MNOP) of 10 psig for the BRR package. The maximum thermal stresses under NCT and HAC are addressed in Section Nos. 2.6.3.1 and 2.7.4.3 of the SAR. These stresses are found below acceptable levels.

The staff reviewed selected calculations and results for NCT and found them acceptable.

3.5 Thermal Evaluation During Drying Operations

The applicant performed transient calculations of vacuum drying operations using the modified NCT thermal model described in Appendix No. 3.5.3, “Analytical Thermal Model”. The modifications made for this evaluation consisted of assuming air as the backfill gas. While the impact limiters will not be installed during vacuum drying operations, and the cask lid will not be installed until just before vacuum drying begins, leaving these components in the thermal model greatly simplified the model modifications required and is seen as having no significant impact on the transient temperatures. The effect of being submerged in the reactor pool is addressed by assuming all cask components are at equilibrium with a maximum temperature of 80°F. The applicant assumed that the thermal conductivity of the gas filling the voids of the packaging and the payload remain unchanged from its base value at atmospheric pressure conditions for vacuum pressures of 1 torr or greater. Based on the transient calculation results, the applicant determined that a minimum of 8 hours exists before the peak fuel plate temperature reaches the NCT limit of 400 ºF. Based on the described model and the thermal evaluation results, the staff finds the BRR thermal evaluation during vacuum drying operations acceptable.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

The initial conditions for the HAC are assumed to be 100°F ambient with no insolation. At the request of the staff, the applicant revised the HAC fire analysis with a fire temperature of 1475°F and a conservative package surface-emissivity of 1.0 to meet the 10 CFR 71.73(c)(4), The applicant had initially used a lower temperature in their analytical model, due to the limitation of not being able to set a flame emissivity of lower than 1.0. This provides an adequate representation of the fully engulfing fire environment consisting of a 1475°F ambient with an effective flame emissivity of 0.9.

The applicant's HAC evaluation uses an ambient temperature of 1475°F for all convection based heat transfer calculations and all radiation based heat transfer calculations. The convection heat transfer coefficients between the package and the ambient during the 30-minute fire event are based on an average gas velocity of 10 m/sec. Following the 30-minute fire event the convection coefficients are based on still air. The ambient condition of 100°F with insolation is assumed following the 30-minute fire event. The applicant continued the post-fire analysis to capture the peak temperatures and continue until a pseudo steady-state is achieved. A solar absorptivity of 0.9 is assumed for the exterior surfaces to account for potential soot accumulation on the package surfaces.

Table No. 3.4-1 of the SAR presents the predicted peak temperature for the BRR package with the MURR fuel payload under HAC conditions. Given that the MURR payload dissipates a significantly higher decay heat than the other potential payloads, the presented temperatures are considered by the applicant to be bounding for all payloads. As seen from the table, significant thermal margins exist for all components. The temperature of the lead shielding in the package remains below its melting point. The closure and vent/drain port seals remain below their maximum allowable temperature due to a combination of their location, the amount of impact limiter foam remaining, even after the conservative damage assumptions, and the surrounding thermal mass of the upper and lower end structures.
The staff reviewed selected calculations and the SAR results for HAC and found them acceptable. The HAC model of the MURR payload and assumed damage configurations adequately represents the various basket configurations and heat loads that are requested.

3.7 Vacuum Drying Procedures

The vacuum drying operations start after the cavity water is drained back into the fuel pool. At this point the cask cavity is filled with air. Per package operation procedures, a vacuum pump is connected to the vent port and the cask cavity is evacuated to a pressure of 1-2 torr. The cavity pressure is monitored for 30 minutes. If after 8 hours of vacuum drying the pressure exceeds 3 torr, the vacuum pump is disconnected and the cask cavity is filled with helium. The vacuum drying operation is repeated until the pressure acceptance criteria are met (evacuating the cask cavity for up to 8 hours and maintaining a cavity pressure of 3 torr for 30 minutes).

3.8 Thermal Tests

A thermal test of the BRR package is not required because there are significantly large thermal margins for all components as demonstrated in the thermal evaluation, and the package design does not appear to contain unique design features that could be sensitive to fabrication errors.

3.9 Conclusion

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

Material Properties and Component Specifications: The staff reviewed the material properties and component specifications used in the thermal evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

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

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

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

Evaluation under Hypothetical Accident Conditions: The staff reviewed the package design, construction, and preparations for shipment and found reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-time limits during hypothetical accident conditions, consistent with the tests specified in 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

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

The BRR package is designated as a Type B(U)F-96 Category I container and is used for shipment of spent fuel elements that have been irradiated in various test and research reactors, including MURR, MITR-II, ATR, and TRIGA reactors. The BRR package may contain the plutonium up to 872.5 Ci, with plutonium in excess of 20 Ci in solid form to meet the requirements of 10 CFR 71.63.

The package is designed to provide a leaktight containment, defined as a leakage rate of less than $1x10^{-7}$ ref-cm^3/s, air, per ANSI N14.5, for fabrication and maintenance/periodic helium leakage rate tests. Therefore, the design criteria must meet the safety requirements of ANSI N14.5 and 10 CFR 71.51: there is no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour for NCT, and there is no escape of radioactive material exceeding an A_2 in one week for HAC.

4.1 Description of Containment System

4.1.1 Containment Design

The BRR package consists of a payload basket, a lead-shielded package body, an upper shield plug, a closure lid, twelve closure bolts, and upper and low impact limiters. The containment boundary of the package consists of inner cylindrical shell, closure lid, upper end structure, lower end structure, containment O-ring seal, vent port, and drain port. Except for the closure bolts and impact limiter attachments, the BRR package is of primarily welded construction using Type 304 austenitic steel.

The BRR package features a vent port in the closure lid and a drain port in the lower end structure, as part of the containment boundary. There are no other valves or receptacles used in the BRR package.

The staff reviewed the containment design features presented in the General Information and Containment chapters of the SAR and concluded that the containment system are consistently defined in General Information and Containment chapters, and ensured that all components of the containment system are adequately displayed in the drawings of 1910-01-01-SAR, sheets 1-4. The staff confirmed that the package design is described and evaluated to demonstrate that it meets the containment requirements of 10 CFR 71.31(a)(1), 71.31(a)(2), 71.33, and 71.35(a).

O-Ring Seals: The seals that comprise the containment boundary are a 3/8-inch diameter, butyl rubber O-ring face seal located in the inner groove in the closure lid, and seal washer sealing elements (an O-ring integrated with a stainless steel washer) for the vent and drain ports. The applicant calculated compression acting on the O-ring containment seals by a minimum compression of 22% and a maximum compression of 26%, based on groove depth and O-ring cross-sectional diameter. The staff referenced the O-ring handbook for recommended minimum compression of 16% and maximum compression of 31.2% when the O-ring cross-section, adjusted for maximum temperature, fills the cross sectional area of the dovetail groove. Therefore, the staff concluded that the compression range of 22% and 26% is 6% above the recommended minimum value and 5.2% below the recommended maximum value, and therefore the compression range is acceptable under NCT.

The applicant performed both NCT and HAC analyses to evaluate the temperatures of O-ring seals at closure lid, vent port, and drain port. The applicant predicted the maximum O-ring seal temperature of 216°F which is below the required limit of 250°F for NCT and the maximum O-ring seal temperature of 373°F which is below the required limit of 400°F for HAC. The staff confirmed that the temperature of containment boundary seals at closure lid, vent port, and drain port will remain within their specified allowable limits under both NCT and HAC and the thermal performance of containment seals under heat loads will satisfy 10 CFR 71.51(a)(1) and 71.71 for NCT and 10 CFR 71.51(a)(2) and 71.73 for HAC.

The applicant declared that the evaluation of the thermal performance under extreme cold conditions (ambient air temperature of -40°F) requires no thermal calculation because the -40°F temperature is within the allowable operating temperature for all package components. The staff confirmed that the butyl rubber has a long term temperature range of -75°F to 250°F and therefore ensured that the butyl rubber O-ring seal remain intact under cold conditions and meets the requirement of 10 CFR 71.71.

The applicant provided the containment boundary closure torque requirements of BRR package in Chapter 2. Twelve closure bolts are used to attach the closure lid to the cask opening and the closure bolts are tighten to 220±20 ft-lb of torque, or a maximum of 240 ft-lb. The staff reviewed the maximum tensile force due to pre-load, pressure loads, and thermally-induced loads in the Structural Evaluation of SAR, and agreed that the torque of the closure bolts are adequate. Therefore, the staff confirmed that the containment system, securely closed by a positive bolt torque, meets the requirements of 10 CFR 71.43(c).

4.1.2 Codes and Standards

The applicant specified the package as a primarily welded construction using Type 304 austenitic stainless steel. All welds used in the containment boundary are full penetration and volumetrically inspected to ensure structural and containment integrity. All welds are subject to visual examination. The welds between the inner containment shell and either end structure, the welds between the outer shell and either end structure, and the longitudinal welds in the outer shell are examined by ultrasonic inspection in accordance with ASME Code, Subsection NB, Article NB-5000. All welds on the BRR package, except seal welds, are liquid penetrant inspected on the final pass in accordance with the ASME Code, Subsection NB, Article NB-5000. The staff reviewed the drawings of 1910-01-01-SAR (sheets 1–4) and the Chapter 8 for the containment related welds for Category I shipping BRR packages, and ensured that all containment boundary welds are to be examined and inspected appropriately in accordance with ASME Code, Section III, Division 1, Subsection NB, Article NB-5000, and satisfy 10 CFR 71.35(a) and 71.37, in accordance with the NUREG/CR-3019 (Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials).

4.1.3 Special Considerations for Damaged Spent Nuclear Fuel

Damaged spent fuel assemblies are not a requested content for this approval.

4.2. Containment under Normal Conditions of Transport

The applicant designed the containment system to have a leakage rate of $1x10^{-7}$ ref-cm3 /sec or less for NCT, which is leaktight in accordance with ANSI N14.5. The user of the package must also perform fabrication, design, periodic, and assembly leak tests to verify containment capability in accordance with ANSI N14.5. Therefore, a NCT containment analyses and allowable leak rate calculation does not need to be performed.

The applicant clarified in Chapter 3 of the application (Section 3.1.4) that the release of fission gases generated from uranium-aluminide and uranium-zirconium hydride based fuels is diffusion-limited as opposed to the direct release mechanism for commercial spent nuclear fuel, and therefore, the pressurization of the cask cavity due to gaseous release from breached fuel elements is insignificant. Then, the applicant set up the MNOP at a bounding level of 10 psig, based on the maximum pressure rise of 5.2 psig under NCT for an assumed fill gas temperature of 70^{o}F.

The staff recognized that the pressurization of the package cavity due to gaseous release from breached fuel elements is insignificant and determined that the MNOP for the BRR package is acceptable and within the safety margin of the design maximum pressure of 25 psig (or 39.7 psia) to meet the requirements of 10 CFR 71.71. The staff also confirmed that the leaktight containment demonstrates the hydrogen generated in the package will not exceed 5% by volume.

The results of the applicant's structural and thermal analyses show that the containment system retains the capability to maintain a seal of $1x10^{-7}$ ref-cm3/sec or less under the conditions specified in 10 CFR 71.71. Therefore, the staff concludes that the loss or dispersal of radioactive material from the package will be less than 10^{-6} A_2 per hour under normal conditions of transport, as required in 10 CFR 71.51(a)(1).

4.3 Containment under Hypothetical Accident Conditions

The applicant designed the containment system to have a leakage rate of $1x10^{-7}$ ref-cm^3/sec or less after HAC, which is leaktight in accordance with ANSI N14.5. Therefore, a HAC containment analyses and allowable leak rate calculation does not need to be performed.

The review procedures for containment under HAC are similar to those under NCT. The maximum pressure of 8.8 psig under HAC is still bounded by both MNOP of 10 psig and design maximum pressure of 25 psig. The staff also confirmed that the evaluation of the maximum pressure in the package design is based on the MNOP as it is affected by fire-caused increases in package component temperatures and is therefore in compliance with 10 CFR 71.73.

The results of the applicant's structural and thermal analyses show that the containment system retains the capability to maintain a seal of $1x10^{-7}$ ref-cm^3/sec or less under the conditions specified in 10 CFR 71.73. Therefore, the staff concludes that the loss or dispersal of radioactive material from the cask will be less than one A_2 per week under normal conditions of transport, as required in 10 CFR 71.51(a)(2).

4.4 Leakage Rate Tests for Type B Package

The BRR package is designed to provide leaktight containment of the radioactive contents under all NCT and HAC. As further evaluated in Section 7 and 8 of this SER, a fabrication leakage rate test is specified to verify the containment integrity of the packaging to a leakage rate not to exceed 1 x 10^{-7} ref-cm^3/s, air; a maintenance and periodic leakage rate test is specified to verify the sealing integrity of the containment seals to a leakage rate not to exceed 1 x 10^{-7} ref-cm^3/s, air; and a pre-shipment leakage rate test is specified to verify the sealing integrity of the containment seals to a leakage rate sensitivity of 1 x 10^{-3} ref-cm^3/s, air. All the leakage rate tests are specified to be consistent with the guidelines of ANSI N14.5.

4.5 Conclusion

Based on the statements and the containment evaluation in the application of BRR package, the staff concluded that the containment design of the BRR package has been adequately described, and evaluated and that the package design satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC.

5.0 SHIELDING EVALUATION

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

The staff's shielding review evaluated the capability of the BRR shielding features to provide adequate protection against direct radiation from its contents. This review included calculation of the dose rates from both gamma and neutron radiation at locations near the cask and at specific distances away from the cask during transportation, verification of computer code benchmarks, and the computer modeling of the cask system for shielding analyses.

5.1 Shielding Design Description

Design Feature: The principal design features of the BRR package are a lead-filled shield plug, lead-filled side wall, and lead-filled bottom. The lid is constructed of stainless steel. The side wall of the cask is constructed of lead. The inner steel shell and the outer are made of stainless steel. The cask bottom consists of lead through the centerline, with stainless steel bottom cover plate, and a stainless steel inner forging. The fuel is positioned within one of four custom-designed baskets. The baskets maintain their geometry under NCT and HAC, as demonstrated in Section No. 2.7.1.5 of the SAR, thereby maintaining the location of the source.

Summary Table of Maximum Radiation Levels: Maximum NCT and HAC dose rates are reported in Table No. 5.1-1 of the SAR. The fuel type associated with each dose rate is provided in the same table.

The applicant homogenized the fuel elements over the active length of the fuel and distributed across the width of each basket compartment. The staff found reasonable that fuel element homogenization is a standard practice utilized to simplify complex source geometry and has little effect on the final results. Basic fuel dimensions used in the homogenization calculation are summarized in Table Nos. 5.3-4, 5.3-5, and 5.3-6 of the SAR. For the TRIGA fuel, the fuel is a simple cylindrical design. Therefore, the TRIGA fuel elements were modeled explicitly, and the source was distributed over the fuel pellets. The fuel type associated with each dose rate is provided in the table. Because the geometry of the source, basket design, and source strength vary widely between the fuel types, no one fuel type may be considered bounding at all dose rate locations. The 2 m dose rate is computed 2 m from the vehicle side, while the occupied location (i.e., the driver) is computed 25 feet from the centerline of the cask.

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

5.2 Source Specification

A neutron and gamma source term is developed for MURR, MITR-II, ATR, and TRIGA fuel elements. The source terms for MURR, MITR-II, and ATR are developed using the TRITON sequence of SCALE6 [3]. The TRIGA source is derived from nuclide activities obtained from INEL-96/0482 [4].

The gamma source terms for each fuel assembly designs were summarized in SAR Table No. 5.2-3. The applicant examined each fuel assembly to determine which assembly gives the maximum source terms but, because the geometry of the source, basket design, and source strength vary widely between the fuel types, no one fuel type may be considered bounding at all dose rate locations.

5.2.1 Gamma Source

In the SAR, the applicant states that the MURR gamma source term is generated by the TRITON sequence of SCALE6. The gamma source term for both the maximum and minimum fuel loadings are nearly identical (within 0.02%), although the source term computed for the minimum fuel loading is slightly higher. The MURR gamma source computed with the minimum fuel loading is summarized in Table No. 5.2-3 of the SAR. According to the applicant, the MURR basket may transport up to eight fuel elements. A representative axial burnup distribution is provided in Table No. 5.2-4 of the SAR. This distribution is the ratio of the burnup in each segment to the average burnup. Key output data are summarized in Table No. 5.2-2 of the SAR. The fuel depletion may be computed based on the initial and final U-235 mass. The initial U-235 mass is 767.2 g, and the final U-235 mass is 530.4 g, or a depletion of 30.9%. The decay heat at a decay time of 180 days is 147.6 W.

The MITR-II gamma source term is generated by the TRITON sequence in the same manner as MURR fuel. Data used to develop the TRITON model is summarized in Table No. 5.2-1, and the NEWT model for MITR-II is shown in Figure No. 5.2-2 of the SAR. The MITR-II fuel element has a loading of 510.0 +3/-10 g U-235. Two TRITON models are developed, one for the minimum fuel loading (500.0 g U-235), and a second for the maximum fuel loading (513.0 g U-235). The U-235 enrichment is 93%. The balance of uranium is modeled as U-238. The fuel is burned in one 900 day cycle. The MITR-II gamma source is summarized in Table No. 5.2-3 of the SAR. Consistent with the MURR gamma source, the source is slightly larger using the minimum fuel loading. Note that the MITR-II basket may transport up to 11 fuel elements. Key output data are summarized in Table No. 5.2-2. The fuel depletion may be computed based on the initial and final U-235 mass. The initial U-235 mass is 500.0 g, and the final U-235 mass is 203.3 g, or a depletion of 59.3%. The decay heat at a decay time of 930 days is 23.7 W.

The applicant states that the ATR gamma source term is generated by the TRITON sequence in the same manner as MURR and MITR-II fuel. Data used to develop the TRITON model is summarized in Table 5.2-1 of the SAR, and the NEWT model for ATR is shown in Figure No. 5.2-3 of the SAR. An ATR fuel element is similar in geometry to a MURR fuel element, although an ATR fuel element has 19 fuel plates instead of 24. According to the applicant, the burnup parameters are selected to bound the highest burned ATR fuel element ever generated. This element had a starting U-235 loading of 1075 g, and a final U-235 loading of 457 g, or a depletion of 57.5%. The fuel is burned in one continuous cycle for 48 days to achieve approximately the same level of depletion of the highest burned ATR element. A bounding element power of 10 MW is utilized, for a total burnup of 480 MWD[2]. The source is allowed to cool 1670 days after reactor shutdown.

The TRIGA fuel gamma source term is derived from information in INEL-96/0482 [4]. This report provides detailed activity values for 145 key isotopes as a function of burnup and decay time for four different TRIGA fuel types. These four fuel types are included in Section No. 1.2.2 of the SAR:

- Type 101 = aluminum-clad standard
- Type 103 = stainless steel-clad standard
- Type 109 = High-enrichment Fuel Life Improvement Program (FLIP)
- Type 117 = Low-enrichment Fuel Life Improvement Program (FLIP-LEU-I)

Key parameters for the four fuel types are summarized in Table No. 5.2-7 of the SAR. Decay times range from discharge to 20 years. Note that the minimum decay time reported in this table has been selected to be the minimum for transportation purposes. The models used to generate the source are described in [4]. The TRIGA fuel is modeled with an irradiation time of 4 years. TRIGA reactors tend to run only sporadically rather than continuously, and TRIGA fuel elements often have residence times exceeding 10 years. According to the applicant, the source is conservative.

5.2.2 Neutron Source

The neutron sources are extracted from the same output files that define the gamma sources, as described in Section No. 5.2.1 of the SAR. The neutron source for MURR, MITR-II, ATR, and TRIGA are presented in Table No. 5.2-8 of the SAR. According to the applicant, the neutron sources presented are the combined spontaneous fission and (α,n) components. Aluminum in the fuel matrix is used as the target nucleus to generate the (α,n) source for the MURR, MITR-II, and ATR fuels. For the TRIGA fuels, no (α,n) target nuclides are present in the fuel matrix. By default, ORIGEN-S utilizes oxygen as a target nucleus if no other target nuclides are present. To be conservative, the (α,n) source with an oxygen target is included in the total for TRIGA, although the actual (α,n) source would be effectively zero because there is no applicable target nuclide in the fuel matrix. This assumption results in an additional conservatism of 9% in the neutron source for the TRIGA fuels.

The staff concludes that the radiation sources are appropriated for the shielding system for the BEA Research Reactor Package.

5.3 Shielding Model

5.3.1 Configuration of Source and Shielding

Section No. 5.3.1 of the SAR provides the model specifications for the shielding evaluation. All relevant design features of the BRR Package were modeled in three-dimensions in MCNP, as shown in Figure No. 5.3-1 of the SAR. The key dimensions relevant to the MCNP model are summarized in Table 5.3-1 of the SAR and are obtained from Section No. 1.3.3 of the SAR.
Minor details were not included in this table but may be inferred from the drawings.

Table No. 5.3-1 of the SAR shows some differences existing between the as-modeled and packaging general arrangement drawing dimensions. Most differences were small and might be neglected. The only notable differences were the outer diameter of the impact limiters, and the diameter of lead at the bottom of the cask. The outer diameter of the impact limiters was modeled to be conservative because the dose rate tally location is brought closer to the source. Also, the lead diameter in the cask bottom was modeled at 9.75-in rather than 10.3-in, which is conservative for shielding. Each fuel element type will be transported in its own unique basket. An axial lead slump of 1.18-in (see Section No. 2.7.1.2 of the SAR, *End Drop*) is modeled at the top of the cask. This slump represents the maximum expected slump due to lead shrinkage and a drop event. Also, an additional 0.0625-in radial lead shrinkage is assumed. Key geometrical parameters for the four basket designs were summarized in Table No. 5.3-2 of the SAR.

5.3.2 Material Properties

In Section No. 5.3.2 of the SAR, the material properties of the BRR System are described. Homogenized fuel number densities were utilized in the MURR, MITR-II, and ATR fuel models. For nominal fuel meat and cladding thicknesses, the total mass of U-235, U-238, and aluminum is estimated for each fuel element. The TRIGA fuel composition was provided in Table No. 5.3-6 of the SAR and was based on 196 g uranium, 2,060 g zirconium, H/Zr ratio of 1.6, and U-235 enrichment of 70%. The baskets were manufactured out of stainless steel, and the cask is constructed of stainless steel and lead. The stainless steel composition and density utilized in the MCNP models are provided in Table No. 5.3-7 of the SAR. Lead is modeled as pure with a density of 11.35 g/cm^3. Void spaces are filled with dry air. The composition is obtained from SCALE material library [5] and is provided in Table No. 5.3-8 of the SAR.

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

5.4 Shielding Evaluation

5.4.1 Methods

The methods for this application are described in Section No. 5.4.1 of the SAR. The dose rates were computed using the MCNP5 v1.30 Computer Program [2]. All relevant package features were modeled in three-dimensions. For simplicity, the impact limiters were modeled simply as air, although the outer surfaces of the impact limiters are treated as the outer surfaces of the package when computing surface dose rates at the ends of the package. It is assumed that under HAC the impact limiters remain attached and suffer 12-in axial crush on each end, and the same MCNP model is used to compute both NCT and HAC dose rates.

The applicant states that the approach was reasonable, because no shielding credit is taken for the impact limiters, other than distance.

Separate models are developed for neutron and gamma radiation. For MURR, MITR-II, and ATR fuel, the fuel plates are homogenized and fill the basket cavities. Homogenization is performed to simplify the source description. For the TRIGA fuel, because the fuel is a simple cylinder, the fuel is modeled explicitly, and the source is distributed over the fuel matrix.

5.4.2 Input and Output Data

A sample input file (gamma source, MITR-II fuel) was included in Section No. 5.5.3.2 of the SAR. The input file was compared against the gamma sources in Table No. 5.2-3 and gamma axial distribution in Table No. 5.2-5 and was verified for the proper model setup. Model geometry and material descriptions was verified by inspection of the supplied input file.

5.4.3 Flux-to-Dose Rate Conversion

ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors are provided in Table No. 5.4.1 of the SAR.

5.4.4 External Radiation Level

The external radiation levels are described in Section No. 5.4.4 of the SAR.

A total of eight input files were developed to compute the NCT and HAC dose rates. A gamma and neutron models were developed for each of the four sources. The files were itemized as follows, where N refers to neutron modeling and G refers to gamma modeling:

- MURR fuel: MURR_N2, MURR_G2
- MITR-II fuel: MIT_N2, MIT_G2
- ATR fuel: ATR_N2, ATR_G2
- TRIGA fuel: TRIGA_NG2, TRIGA_G2

5.4.5 Confirmatory Analyses

The staff reviewed the applicant's shielding models used in the analyses. The staff checked the code input in the calculation packages and confirmed that the proper material properties and boundary conditions were used. The engineering drawings were also consulted to verify that proper geometry dimensions were translated to the analysis model. The material properties presented in the SAR, were reviewed to verify that they were appropriately referenced and used.

The staff performed shielding and source term calculations using SCALE6 [11] to compare photon and neutron sources.

The staff concludes that the design of the shielding system for the BEA Research Reactor Package is in compliance with 10 CFR Part 71 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the BRR cask will provide safe transporting of spent fuel. This finding is based on a review that considered the specifications in the SAR Revision 2, the regulations, appropriate regulatory guides, staff confirmatory (including calculations and modeling) analysis, and accepted engineering practices. The staff reviewed the external radiation levels under normal conditions of transport and hypothetical accident conditions and found reasonable assurance that they satisfy 10 CFR 71.43(f) and 71.51(a)(1).

5.5 References

1. Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), Packaging and Transportation of Radioactive Material, 1–1–09 Edition.
2. MCNP5, "*MCNP – A General Monte Carlo N-Particle Transport Code, Version 5; Volume II: User's Guide*," LA-CP-03-0245, Los Alamos National Laboratory, April 2003. MCNP5 is distributed by the Radiation Safety Information Computational Center (www-rsicc.ornl.gov), Release C00710MNYCP02 (Windows PC).
3. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 6, Vols. I-III, January 2009.
4. JW Sterbentz, *Radionuclide Mass Inventory, Activity, Decay Heat, and Dose Rate Parametric Data for TRIGA Spent Nuclear Fuels*, INEL-96/0482, Idaho National Engineering Laboratory, March 1997.
5. *Standard Composition Library*, ORNL/TM-2005/39, Version 6, Vol. III, Section M8, January 2009.
6. *Nuclides and Isotopes, Chart of the Nuclides, Fifteenth Edition*, General Electric Co. and KAPL, Inc., 1996.

6.0 Criticality Evaluation

The objective of the criticality review is to verify that the package design satisfies the criticality safety requirements of 10 CFR Part 71 under NCT and HAC. The objective of the review is to ensure that the contents will remain subcritical under all credible normal, off-normal, and accident conditions encountered during handling, packaging, transfer, and storage. These objectives include a review of the criticality design criteria; features and fuel specifications; verification and review of the configuration and material properties for the BRR package; and a review of the methodology and results found in the criticality evaluation.

The staff reviewed the BRR criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the BRR package with contents identified in the SAR meet the following regulatory requirements: 10 CFR 71.31, 71.33, 71.35, and 71.59. The staff's review

also involved a determination on whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1617, "Standard Review Plan for Transportation Packages.

6.1 Description of Criticality Design

The BRR is a cask system developed to transport irradiated research reactor fuel. The cask consists of a basket structure used to hold highly-enriched uranium-aluminum fuel. The cask has a cylindrical stainless steel body with an upper shield plug and impact limiters. The BRR is designed to remain leaktight during all normal conditions of transport and hypothetical accident conditions. The cask has a drain port to ensure water is not present during transport. The basket structure consists of a fuel-specific payload basket used for one of the four fuel types authorized to be transported in the BRR. These basket structures control spacing between fuel elements, and limit the number of fuel elements that may be included in a shipment. The baskets also have an open top-end, and are designed to freely drain water out when the cask is lifted out of the spent fuel pool. The fuel is located at the top end, near the shield plug.

6.2 Spent Nuclear Fuel Contents

The BRR has been approved for transporting baskets containing one of the four authorized fuel types. The application authorizes the BRR cask to carry the following contents:

MURR: The MURR basket structure can accommodate up to 8 MURR fuel elements. Each MURR element contains up to 782.8 g of U-235 at enrichments up to 93 wt.%. Each MURR fuel element contains 24 curved fuel plates. The fuel matrix is a U-Al mixture with an aluminum alloy cladding. The active fuel length ranges from 23.25 in to 24.75 in.

MITR-II: The MITR-II basket structure can accommodate up to 11 MITR-II fuel elements. Each MITR-II element contains up to 513 g of U-235 at enrichments up to 93 wt.%. Each MITR-II fuel element contains 15 flat fuel plates. The fuel matrix is a U-Al mixture with an aluminum alloy cladding. The nominal active fuel length is 22.375 in.

ATR: The ATR basket structure can accommodate up to 8 ATR fuel elements. Each ATR element contains up to 1085 g of U-235 at enrichments up to 93 wt.%. Each ATR fuel element contains 19 curved fuel plates. The fuel matrix is a U-Al mixture with an aluminum alloy cladding. The nominal active fuel length is 22.375 in. The active fuel length ranges from 47.245 in to 48.775 in.

TRIGA: The TRIGA basket structure can accommodate up to 19 TRIGA fuel elements. There are 5 different types of TRIGA fuel elements authorized to be transported in the BRR cask:

- 8.0 wt.% uranium, 20 wt.% U-235, aluminum cladding
- 8.5 wt.% uranium, 20 wt.% U-235, stainless steel cladding
- 8.5 wt.% uranium, 70 wt.% U-235, stainless steel cladding
- 20.0 wt.% uranium, 20 wt.% U-235, stainless steel cladding
- 20.0 wt.% uranium, 20 wt.% U-235, stainless steel cladding

TRIGA fuel elements consist of a central active fuel region with graphite axial reflectors above and below the active fuel. The active fuel length ranges from 14 to 15 in.

6.3 General Considerations for Criticality Evaluations

The applicant performed analyses for a single package under conditions of 10 CFR 71.55(b), (d), and (e), and for undamaged and damaged arrays of packages under their respective conditions specified in 10 CFR 71.59(a)(1) and (2). The results of these analyses, presented in Section 6.4 through 6.6 of the SAR, showed the calculated k-effectives and their standard deviations. The upper subcritical limit (USL) for the BRR (package or package array) is determined by the applicant to be 0.9209. The package is considered to be subcritical if k_{safe} (k_s) for the cases is less than or equal to the USL. The computed k_{safe} is equated as $k_s = k_{eff} + 2\sigma < USL$. Staff reviewed the results and found that the most reactive cases are clearly indicated, and were demonstrated to be less than the USL. The reactivity for each fuel type and the USL are found in Table 6.1.1 of the SAR.

No credit is taken for fuel burnup in any of the criticality models generated for the BRR package. In addition, the gaps between individual fuel plates are maximized for the aluminum plate fuel-types (MURR, MITR-II, and ATR). Maximizing the gaps between elements would act to increase the reactivity by increasing the amount of potential moderation during the flooding scenarios used in the analyses. Infinite reflection is also used in the NCT and HAC models. Each fuel type has its own individual basket design as described in Section 1.3.3 of the SAR. The fuel elements and corresponding fuel baskets are modeled as being undamaged in all NCT and HAC models. The fuel and fuel baskets were demonstrated to remain undamaged per analyses found in Section 2.7.1.6 and Section 2.7.1.5, respectively.

MURR, MITR-II, and ATR fuel elements each contain uranium enriched up to 93 wt.% U-235. In the criticality models, these were modeled as being enriched to 94 wt.% U-235. For TRIGA fuel, the mass densities are computed from the uranium and U-235 contents in Table 6.2-7 of the SAR. The samarium trioxide and molybdenum discs located between the active fuel and the reflectors were ignored in the model for simplicity. Samarium trioxide was conservatively ignored because it acts as a poison. Parametric studies for each of the fuel types were performed in Section 6.9.2 of the SAR. It is shown that the inclusion of the molybdenum discs only resulted in a slight decrease in reactivity. Therefore, the molybdenum discs were also disregarded in the models.

Computer Codes and Cross Section Libraries: The applicant used the three-dimensional continuous energy code Monte Carlo N-Particle (MCNP5) for the criticality analysis. The uranium isotopes utilize preliminary ENDF/B-VII cross section data. Chromium, nickel, iron, and lead were taken from ENDF/B-V. The staff agrees that the codes and cross-section sets used in the analysis are appropriate for this application and fuel system.

Material Properties: The fuel meat materials used in the explicit MCNP5 analyses were provided in Section 6.2 of the SAR. Some other standard materials were taken from the SCALE material library, and are listed in Table 6.3-3 of the SAR.

6.4 Single Package Evaluation

The BRR cask is modeled as a steel-lead-steel cylinder. The cask cavity is modeled as a void (dry condition). Although the fuel baskets are designed to drain freely, it has been shown that the ATR fuel element retains a residual amount of water when it is raised from a spent fuel pool and allowed to drip dry. For the NCT cases, this small amount of water is neglected in the model. Maximum reactivity for the single package NCT case is 0.4167, which is less than the established k_{safe} of 0.9209

For the HAC case, water is modeled in the cask cavity for partial and fully flooded conditions to determine the moderation resulting in maximum reactivity. In addition, the orientation of the fuel elements within the cask is varied to determine the effects of center-to-center spacing. For MURR, MITR-II, and ATR fuels, maximum reactivity occurs when the cavity is fully flooded because the system is undermoderated. For TRIGA fuel the optimum moderation occurs at 0.6 or 0.7 g/cm^3. All four fuel types show that when the fuel elements are shifted up axially, reactivity increases. In addition, reactivity increases when the fuel elements are shifted toward the center of the cask; allowing more interaction. It is shown that the reactivity increases as the fuel is shifted up, due to the increased reflection from the shield plug.
Maximum reactivity for the single package HAC case is 0.8750, which is less than the established k_{safe} of 0.9209.

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

For the array cases, the packages are modeled with the cask cavities being dry (void). The water density was varied from 0 to 1.0 g/cm^3 between the packages to determine the most reactive configuration. The packages are modeled in a close-packed hexagonal array. To simplify the model, the impact limiters were neglected. Excluding the impact limiters in the models conservatively minimizes the separation between the packages, which acts to increase reactivity by increasing interaction between packages. The steel shield lid is also excluded from the package model. This is judged to be conservative since excluding the shield lid would allow more interaction in the axial direction for array models. It is shown in Section 6.3.4 of the SAR that the maximum reactivity occurs with no water between the packages. Maximum reactivity for the NCT package array cases is 0.5394, which is less than the established k_{safe} of 0.9209.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

For the array cases, the HAC array was modeled as being in a close-packed hexagonal array with varying water between the cask packages. The most reactive cases involved full-density water internal moderation with no external interstitial moderation. Any increased amounts of external water would act to isolate the fuel assemblies from each other within the array. Maximum reactivity for the single package and array cases HAC scenario is 0.890, as shown in Table 6.1.1 of the SAR.

This occurred for the MITR-II fuel configuration, which is less than the established k_{safe} of 0.9209.

6.7 Criticality Safety Index

An infinite number of BRR packages are evaluated in a close-packed hexagonal array for the NCT and HAC, making “N” infinite. In accordance with 10 CFR 71.59, the criticality safety index (CSI) is 50/N = 0.

6.8 Conclusion

Based on the review of the information and representations made by the applicant in the SAR, the staff finds reasonable assurance that the package design with the proposed contents meets the criticality requirements identified in 10 CFR Part 71.

7.0 PACKAGE OPERATIONS EVALUATION

The objective of this review is to verify that the operating controls and procedures meet the requirements of 10 CFR Part 71 and that the operating procedures are adequate to assure the package will be operated in a manner consistent with it’s evaluation for approval. The staff reviewed the BRR transport package operating procedures to ensure that the transportation package will be operated in accordance with 10 CFR Part 71, and to ensure the package loading, preparation for package shipment, and the package unloading procedures, are done in accordance with regulatory requirements.

7.1 Package Loading

The applicant delineated the procedures of the helium leakage rate tests on the containment O-ring seals located at closure lid, vent port and drain port during and after assembly for shipment in SAR 7.0. The package will be tested with the pre-shipment leakage rate of less than 1.0×10^{-3} ref-cm^3/s, air, in accordance with ANSI N14.5. The applicant also listed the procedures for loading a payload to the BRR package in SAR Chapter 7.1.2, “Loading of Content”. The staff reviewed the loading steps, and requested a modification of step 30 for a potential risk due to the leaking valve. The applicant corrected the step 30 from “Connect a vacuum pump to the vent port tool and evacuate the cavity until the internal pressure is 1–2 torr. Isolate the vacuum pump from the cask body cavity” to “Connect a vacuum pump and a shutoff valve to the vent port tool, and evacuate the cavity until the internal pressure is 1 – 2 torr. Isolate the vacuum pump from the cask body cavity by closing the shutoff valve and shutting off the vacuum pump, closing the shutoff valve and venting the suction line to atmosphere, or other appropriate means that does not maintain a vacuum on the outlet of the shutoff valve” per request.

The staff accepted the modification and concluded the loading operations satisfy the requirements of 10 CFR 71.31(c) and 71.35.

7.2 Preparation for Package Shipment

Procedures are provided to ensure the level of loose contamination on the external surface of the BRR package is below required limits. Procedures are provided which adequately control radiation below required limits. Procedures are provided to ensure rigging is appropriately removed and impact limiters are appropriately attached to support shipment.

Staff reviewed the proposed preparation for package shipment procedures and concludes the procedures satisfy 10 CFR 71.35.

7.3 Package Unloading

The unloading procedures are the reverse of the loading procedures. Procedures are provided to ensure safe removal of any gases, coolant, and contamination. Procedures are provided which adequately control radiation and contamination below required limits.

Staff reviewed the proposed special controls and precautions for unloading and handling and concludes the controls satisfy 10 CFR 71.35(c).

7.4 Evaluation Findings

Staff requested the applicant clarify impairment of the containment boundary in the loading procedure. Clarification was provided.

Based on review of the statements and modifications in the BRR package application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and

that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTERNANCE PROGRAM EVALUATION

The objective of this review is to verify that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure the packaging performance during its service life. The staff reviewed the BRR transport package operating procedures to ensure that the transportation package will be tested and maintained in accordance with 10 CFR Part 71, and to ensure that the welding examinations, the fabrication leakage rate tests and the maintenance/periodic leakage rate tests are done in accordance with regulatory requirements.

8.1 Acceptance Tests

Section 8.1 of the application specifies review, inspection, and testing of the package. The acceptance tests and inspections considered critical to the safe operation of the BRR package are requirements in the CoC.

8.1.1 Visual Inspections and Measurements

Visual inspection requirements are provided which verify the packaging has been fabricated and assembled in accordance with drawings and other requirements specified in the SAR. Dimensions and tolerances specified on the drawings are confirmed by measurement.

8.1.2 Weld Inspections

Weld inspection requirements are provided to verify fabrication in accordance with drawings, codes, and standards specified in the SAR to control weld quality. Location, type, and size of the welds are confirmed by measurement. Welds are examined by ultrasonic, radiographic, and liquid penetrant inspections in accordance with appropriate codes.

8.1.3 Structural and Pressure Tests

Applicable structural and pressure tests are provided. The BRR package does not contain any lifting devices that require load testing. The BRR package containment boundary will be pressure tested to 125% of the design pressure in accordance with appropriate codes. The test pressure is greater and more conservative than other required test pressures and meets requirements in 10 CFR Part 71.85. Visual and liquid penetrant testing of containment welds after pressure testing is required.

8.1.4 Leakage Tests

The applicant delineated the procedures of the helium leakage rate tests on the containment O-ring seals located at closure lid, vent port and drain port during and after fabrication in SAR Section 8.0 and declared the containment integrity of the BRR packaging to a leakage rate less than 1.0×10^{-7} ref-cm^3/s, air, for fabrication leakage rate test of acceptance tests. The applicant delineated the procedures of helium leakage rate test on the main containment O-ring seals in the closure lid through evacuating the containment system to a 90% vacuum or better (≤10% ambient atmosphere pressure) and backfilling the helium gas (99% purity or better) to ambient atmospheric pressure (+1 psi, -0 psi), to meet the requirements of fabrication leakage rate test criteria and maintenance/periodic leakage rate test criteria.

The staff verified that the fabrication leakage rate test provided in SAR Sections 4.4.1 and 8.1.4 is consistent with Sections 7.3 and 8.4 of ANSI N14.5; and the preshipment leakage rate test described in SAR Sections 4.4.3 and 7.4 is in accordance with Section 7.6 of ANSI N14.5. The staff also confirmed that the acceptance helium leakage rate testing procedures of the containment O-ring seals at closure lid, drain port, and vent port are well described and acceptable with respect to the guidelines of ANSI N14.5.

8.1.5 Component and Materials Tests

There are no valves or couplings in the BRR package. The lid is sealed using double elastomeric seals and will be leak tested per the guidelines in ANSI N 14.5.

Requirements and acceptance criteria for installation, inspection, and testing of the rigid, closed–cell, polyurethane foam utilized within the BRR packaging impact limiters are adequately provided and appropriately conform to ASTM testing standards.

8.1.6 Shielding Tests

In–situ or poured lead shielding integrity will be confirmed via gamma scanning. Two gamma scan techniques are utilized. The tests and acceptance criteria are sufficient to assure no voids or streaming paths exist in the shielding.

8.1.7 Neutron Absorber Tests

No neutron absorber tests are required.

8.1.8 Thermal Tests

Based on the information provided by the applicant, the staff has determined that a thermal test of the BRR package is not required because there are significantly large thermal margins for all components as demonstrated in the thermal evaluation, and the package design does not appear to contain unique design features that could be sensitive to fabrication errors.

8.2 Maintenance Tests

The maintenance and periodic leakage rate test expressed in SAR 4.4.2 and 8.2.2 is consistent with Sections 7.4 and 7.5 of ANSI N14.5. The staff also confirmed that the maintenance helium leakage rate testing procedures of the containment O-ring seals at closure lid, drain port, and vent port are well described and acceptable with respect to the guidelines of ANSI N14.5.

8.2.1 Structural and Pressure Tests

No structural or pressure tests are necessary to ensure continued performance of the packaging. The BRR package does not contain any lifting devices that require load testing.

8.2.2 Leakage Tests

Procedures for periodic leakage rate testing of the containment boundary penetrations during routine maintenance, or at the time of seal replacement or sealing area repair are provided. Maintenance leakage rate testing will follow the guidelines of Section 7.4, Maintenance Leakage Rate Test, and Section 7.5, Periodic Leakage Rate Test, of ANSI N14.5. Maintenance/periodic leakage rate testing will be performed on the main O–ring seal, the vent port sealing washer, and the drain port sealing washer for the containment boundary and conform with ANSI N14.5.

8.2.3 Component Tests

Periodic tests and replacement schedules for components are provided. Components include threaded fasteners such as, closure lid bolts, the vent port plug, and the drain port plug, impact limiters and seals. Prior to each use and at the time of seal replacement, containment sealing surfaces will be visually inspected for damage that could impair the sealing capabilities of the packaging. Leak testing procedures are provided upon completion of surface finish repairs. Before each use, the impact limiters will be inspected and repaired as appropriate for tears or perforations. The containment boundary O–ring seal, the vent port sealing washer, and the drain port sealing washer will be replaced within a one year period prior to shipment or when damaged, whichever is sooner.

8.2.4 Neutron Absorber Tests

No neutron absorber tests are required.

8.2.5 Thermal Tests

Based on the information provided by the applicant, the staff has determined that a thermal test of the BRR package is not required because there are significantly large thermal margins for all components as demonstrated in the thermal evaluation, and

the package design does not appear to contain unique design features that could be sensitive to fabrication errors.

8.3 Evaluation Findings

Based on review of the statements and modifications in the BRR package application, the staff concludes that the acceptance tests and the maintenance program for the packaging is adequate to assure packaging performance during its service life and the tests and program meet the requirements of 10 CFR Part 71.31, 71.37, and 71.85.

CONDITIONS

The following are conditions in CoC No. 9341, Revision No. 0:

- The package shall be prepared for shipment, operated, tested, and maintained in accordance with Chapter 7 and Chapter 8 of the SAR.

- The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

- Transport by air of fissile material is not authorized.

- The certificate expires on January 7, 2015.

CONCLUSION

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. BRR package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9341, Revision No. 0,
on January 21, 2010.