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

July 21, 2016

Mr. Philip W. Noss
Licensing Manager
AREVA Federal Services LLC
505 S. 336th St., Suite 400
Federal Way, WA 98003

SUBJECT: REVISION NO. 5 OF CERTIFICATE OF COMPLIANCE NO. 9341 FOR THE MODEL NO. BRR TRANSPORTATION PACKAGE

Dear Mr. Noss:

As requested in your application dated June 26, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15188A084), as supplemented on January 26, 2016 (ADAMS Accession No. ML16054A510), May 26, 2016 (ADAMS Accession No. ML16168A266), and June 2, 2016 (ADAMS Package Accession No. ML16160A206), enclosed is Certificate of Compliance No. 9341, Revision No. 5, for the Model No. BRR transportation package. The staff's safety evaluation report is also enclosed.

The 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 Title 49 of the *Code of Federal Regulations* 173.471 "Requirements for U.S. Nuclear Regulatory Commission Approved Packages."

Upon removal of Enclosure 3, this document is uncontrolled.

If you have any questions regarding this certificate, please contact me or Norma Garcia Santos of my staff at (301) 415-6999.

Sincerely,

/RA/
John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9341
TAC No. L25031

Enclosures:
1. Certificate of Compliance No. 9341, Revision No. 5
2. Safety Evaluation Report
3. List of Registered Users

cc w/encls. 1 & 2: R. Boyle, U.S. Department of Transportation
J. Shuler, U.S. Department of Energy, c/o L. F. Gelder

If you have any questions regarding this certificate, please contact me or Norma Garcia Santos of my staff at (301) 415-6999.

Sincerely,

/RA/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9341
TAC No. L25031

Enclosures:
1. Certificate of Compliance No. 9341, Revision No. 5
2. Safety Evaluation Report
3. List of Registered Users

Upon removal of Enclosure 3, this document is uncontrolled.

cc w/encls. 1 & 2: R. Boyle, U.S. Department of Transportation
J. Shuler, U.S. Department of Energy, c/o L. F. Gelder

DISTRIBUTION: (Closes TAC No L24708)
SFST r/f NMSS r/f DMarcano

ADAMS Package Accession No.: ML16204A306
Cover Letter and SER ADAMS Accession No.: ML16204A337
CoC ADAMS Accession No.: ML16204A336
This closes CAC No. L25031.

OFC	DSFM	E	DSFM		DSFM		DSFM		DSFM		DSFM		DSFM	
NAME	NGarcia-Santos		SFigueroa by e-mail		QTE by e-mail		Z. Li		E. Goldfeiz by e-mail		Y.Kim by e-mail		J. Piotter by e-mail	
DATE	6/23/2016		7/20/2016		7/1/2016		7/21/2016		7/13/2016		7/14/2016		7/12/2016	
OFC	DSFM		DSFM		DSFM		DSFM		DSFM		DSFM		DSFM	
NAME	D. Tarantino by e-mail		T. Ahn by e-mail		ACsontos by e-mail		J. Piotter by e-mail		DChung for MRahimi		J. McKirgan			
DATE	7/14/2016		7/12/2016		7/21/2016		7/21/2016		7/21/2016		7/21/2016			

C = COVER E = COVER & ENCLOSURE N = NO COPY
OFFICIAL RECORD COPY

This page was intentionally left blank.

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

SAFETY EVALUATION REPORT

Docket No. 71-9341
Model No. BRR
Certificate of Compliance No. 9341
Revision No. 5

This page was intentionally left blank.

CONTENTS

SUMMARY ..... - 1 -
1.0 GENERAL INFORMATION ..... - 2 -
1.1 Packaging ..... - 2 -
1.2 Drawings ..... - 3 -
1.3 Contents ..... - 3 -
1.4 Criticality Safety Index ..... - 6 -
1.5. Evaluation Findings ..... - 6 -
2.0 STRUCTURAL EVALUATION ..... - 6 -
2.1 Description of Structural Design ..... - 6 -
2.2 Chemical and Galvanic Reactions ..... - 6 -
2.3 Radiation Effects on Materials ..... - 7 -
2.4 Normal Conditions of Transport ..... - 7 -
2.5 Hypothetical Accident Conditions ..... - 7 -
2.6 Evaluation Findings ..... - 8 -
3.0 THERMAL EVALUATION ..... - 8 -
3.1 Description of the Thermal Design ..... - 8 -
3.2 Material Properties and Component Specifications ..... - 9 -
3.3 Evaluation of Accessible Surface Temperatures ..... - 9 -
3.4 Normal Conditions of Transport ..... - 9 -
3.5 Hypothetical Accident Conditions ..... - 11 -
3.6 Evaluation Findings ..... - 12 -
4.0 CONTAINMENT EVALUATION ..... - 12 -
5.0 SHIELDING EVALUATION ..... - 12 -
5.1 Shielding Design Description ..... - 12 -
5.2 Summary Table of Maximum Radiation Levels ..... - 13 -
5.3 Source Specification ..... - 13 -
5.4 Shielding Model ..... - 18 -
5.5 Shielding Evaluation ..... - 19 -
5.6 Evaluation Findings ..... - 21 -
6.0 CRITICALITY SAFETY EVALUATION ..... - 21 -
6.1. Description of Criticality Safety Design ..... - 22 -
6.2 General Considerations ..... - 23 -

6.3 Evaluation Findings ..... - 28 -
7.0 PACKAGE OPERATIONS ..... - 28 -
8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW ..... - 28 -
REFERENCES ..... - 29 -
CONDITIONS ..... - 29 -
CONCLUSIONS ..... - 35 -

SAFETY EVALUATION REPORT

Docket No. 71-9341
Model No. BRR
Certificate of Compliance No. 9341
Revision No. 5

SUMMARY

As requested in your application dated June 26, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15188A084), as supplemented on January 26, 2016 (ADAMS Accession No. ML16054A510), May 26, 2016 (ADAMS Accession No. ML16168A266), and June 2, 2016 (ADAMS Package Accession No. ML16160A206), AREVA Federal Services LLC (thereafter, the applicant), requested that the U.S. Nuclear Regulatory Commission (NRC) approve a revision to the Model No. BRR package. This revision includes adding 21 new types of TRIGA fuel, six types of research reactor fuel, and a new basket design (including a loose plate box) to the certificate of compliance (CoC) for this package. This revision includes:

1) Adding several square or nearly square research reactor fuel elements or loose plates as payloads (including PULSTAR fuel).

2) Adding a new basket design and a loose plate box for transporting square or nearly square irradiated fuel. The loose plate box accommodates flat or slightly curved loose fuel plates disassembled from fuel elemenets of three research reactors, namely University of Massachusetts at Lowell [U-Mass (aluminide)], University of Florida (U-Florida), and Purdue University (Purdue) reactors.

3) Increasing the types of authorized TRIGA fuels from 5 to 26 (The new TRIGA fuels will use the previously approved basket.

The Model No. BRR (thereafter, BRR) is a Type B(U)F-96 package to ship irradiated fuel from research reactor facilities. The package's design allows transporting one package per conveyance, with its longitudinal axis vertical, by highway truck or by rail in exclusive use.

The NRC staff (thereafter, the staff) reviewed the application, including relevant information in the attachment to the application, using the guidance in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." The staff also considered supplemental information provided by the applicant, proprietary calculation packages, and conference calls. Based on the statements and representations in the application, as supplemented, and the "conditions" section of this safety evaluation report (SER), the staff concludes that the package meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, "Packaging and Transportation of Radioactive Material."

1.0 GENERAL INFORMATION

1.1 Packaging

The BRR package body is a right circular cylinder 77.1 in. long and 38 in. in diameter. It comprises inner and outer shells that is connected by a thick lower end casting. The shells and lower end casting are made of American Society for Testing and Materials (ASTM) Type 304 stainless steel with an encased lead shield. The cast-in-place lead shielding fills the annulus between the shells. Together with the removable 11.2-in. thick shield plug under the closure lid, the package body assembly constitutes the payload cavity, which has a diameter of 16 in. and a length of 54 in.

The principal components of the BRR are:

1) a lead-shielded package body
2) a separate, removable upper shield plug
3) a bolted closure lid
4) upper and lower impact limiters containing polyurethane foam
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 a welded structure using Type 304 austenitic stainless steel. Drawing No. 1910-01-01 of the application provides the details of the structural design of the package body assembly. In addition, a set of eight receptacles are attached to the outer shell at each end of the body to serve as impact limiter attachments.

The applicant added a basket and a loose plate box to the design of the packaging to accommodate the new payloads requested in this revision to the CoC. The following sections include the evaluation of the changes to the packaging to accommodate the proposed contents in this revision request.

1.1.2 Package Body

There are no changes in the design of the package body.

1.1.3 Baskets

The BRR packaging system can currently transport four different kinds of payload baskets to accommodate MURR, MITR-II, ATR, and TRIGA fuels. Each basket has a cavity that fits the size and shape of the fuel to minimize “free play” between the fuel and the basket and facilitate the insertion and removal of the elements. The baskets are open on the top and the top of the fuel is near the bottom of the shield plug. The design of the baskets allows water to drain freely during the movement of the package outside of the spent fuel pool.

The applicant added a basket design to accommodate several fuel types including flat or slightly curved loose plates (loose plates) and fuel with square or rectangular cross section. A square box is used for transporting loose plates in the square fuel basket. The square fuel basket, the loose plate box, and the fuel pedestals used with the square fuel basket are made of stainless steel. Drawing No. 1910-01-03, Revision 6, of the application provides structural details of the new basket design.

1.1.4 Impact Limiters

The applicant did not make any changes related to the design of the impact limiters for the BRR package.

1.2 Drawings

For the contents related to this revision of the CoC for the BRR package, the applicant proposed changes to drawing No. 1910-01-03, "BRR Package Fuel Baskets," Revision 6. The changes to drawing No. 1910-01-03 include adding a new basket design, with a square fuel loose plate box, to accommodate the proposed contents requested in this revision of the CoC. The staff reviewed the drawings and these included dimensions, package markings, materials of construction, and the codes and standards the applicant used to design the package.

1.3 Contents

The BRR package is currently used to transport fuel elements irradiated in various test and research reactors, including:

MURR	University of Missouri Research Reactor
MITR–II	Massachusetts Institute of Technology Nuclear Research Reactor
ATR	Advanced Test Reactor
TRIGA	Training, Research, Isotopes, General Atomics reactors

The applicant is requesting to add the following types of fuel as authorized contents of the BRR:

1) 21 configurations of TRIGA fuel

2) Square fuel and loose plates
   a. Rhode Island Nuclear Science Center (RINSC) fuel
   b. University of Massachusetts at Lowell (U-Mass) fuel [U-Mass (aluminide) loose plates]
   c. Ohio State University (Ohio State) fuel
   d. Missouri University of Science and Technology (Missouri S&T) fuel
   e. University of Florida (U-Florida) fuel (fuel assemblies and loose plates)
   f. Purdue University (Purdue) fuel (fuel assemblies and loose plates)
   g. PULSTAR fuel

3) A maximum quantity of plutonium in the BRR package is 6,500 Ci (at 4% ^{235}U enrichment and 20 GWd/MTU burnup of the PULSTAR fuel)

For irradiated MURR, MITR-II, ATR, and square fuel elements or loose plates in loose plate boxes, the BRR package may contain up to 8 of these fuel elements or fully loaded loose plate boxes.

Only one fuel element is allowed per basket location. Nevertheless, users of the BRR package may load loose plate boxes in the same basket containing same fuel elements. A loose plate box can accommodate a maximum of 31 fuel plates. Table 1 includes a summary of the fuel characteristics of the proposed content in this revision request of the CoC for the BRR. The following sections include a discussion of the fuel types pertaining to this revision.

1.3.1 TRIGA Fuel

Section 1.2 of the application includes a description of the TRIGA fuel. TRIGA fuel elements fall into five general categories of General Atomics series:

1) *Standard 100 series*.

2) *Instrumented 200 series*. The fuel region is as the same as 100 series but contain thermocouples used to measure temperature during reactor operation. Instrumented rods may be longer than 100 series.

3) *Fueled follower control rods (FFCR) (300 series)*. The rods contain boron carbide neutron absorber outside the active fuel region.

4) *Cluster rods (400 series)*. It is typically built with three or four cluster rods to make a cluster assembly.

5) *Instrumented cluster rods (500 series)*. Fuel is the same as cluster rod but thermocouples are used to measure temperature during reactor operation. Instrumented cluster rods may be longer.

The cluster rods are disassembled from the cluster assembly for transportation in the BRR package.

Tables 1.2-1 and 1.2-2 of the application include characteristics of the TRIGA fuel.

1.3.2 Square Fuel

Sections 1.2.1.3.5 and 1.2.2.5 of the application include a description of the square fuel. There are two types of fuel assemblies available for transport within the square fuel basket:

1) *Flat fuel element(s) (fuel plates)*—A uranium-oxide dispersion or uranium-silicide dispersion meat in an aluminum matrix, bonded with an aluminum alloy cladding.

2) *PULSTAR fuel assembly*—A 5×5 array of fuel rods enclosed within a rectangular can. Each fuel rod contains cylindrical uranium oxide fuel pellets. Plutonium can be present in solid form within the fuel matrix as a consequence of the irradiation of reactor fuel. The maximum quantity of plutonium in the BRR package is 6,500 Ci (at 4% ^{235}U enrichment and 20 GWd/MTU burnup of PULSTAR fuel). Table 1.2-4 the application includes the key characteristics of the PULSTAR fuel.

Table 1.2-3, “Square Plate Fuel Characteristics,” and Table 1.2-4, “PULSTAR Fuel Characteristics,” of the application include a summary of the characteristics of the square fuel.

Table 1. Overview of Fuel Characteristics for the Proposed Authorized Contents for Transport in the Model No. BRR

	Units	TRIGA [1, 2]	SQUARE FUEL						
			PULSTAR[3]	Square Plate Fuels[4]					
				U-Mass	RINSC	Ohio State	Missouri S&T	U-Florida	Purdue
Irradiated Fuel Elements Maximum Burnup	Megawatt-day/fuel element (except PULSTAR)	22-122	20,000/MTU	9.7	52.5	64	74	87	0.57
Minimum Cooling Time	days	90-600	548	1,000	120	120	365	120	120
Fuel Element Fissile Material	None	zirconium-hydride fuel matrix [5]	uranium oxide (UO_2) fuel matrix [6]	uranium silicide (U_3Si_2) and uranium aluminide (UAl_x)[4]	uranium silicide (U_3Si_2) fuel matrix mixed with aluminum[4]				
Cladding Material	None	aluminum (Al) or stainless steel (SS) [5]	zirconium alloy [6]	aluminum					
Active Fuel Length	inches	14–15	24.1	23.25					
Overall Length	inches	28.62-45.50	38.23 [7]	27.38–39.75					
Minimum Nominal Cladding Thickness	inches	0.02	0.0185 (minimum)	0.005[8]					
Nominal ^{235}U Enrichment	weight %	20 – 70	4.0/6.0	19.75					
Maximum Decay Heat[9]	watts	20.0 per fuel element[5]	30 per basket compartment						

Value ranges vary with more detailed fuel specifications.

---

1 TRIGA elements include five general categories of General Atomics: series of 100 (standard), 200 (instrumented), 300 (fuel follower control rod), 400 (cluster rods), and 500 (instrumented cluster rods)
2 As provided by the applicant in Table 1.2-1, "TRIGA Fresh Fuel Characteristics," of the application.
3 As provided by the applicant in Table 1.2-4, "PULSTAR Fuel Characteristics," of the application.
4 As provided by the applicant in Table 1.2-3, "Square Plate Fuel Characteristics," of the application.
5 As provided by the applicant in Section 1.2.2.4, "TRIGA," of the application.
6 As provided by the applicant in Section 1.2.2.5, "Square Fuel and Loose Plates," of the application.
7 As provided by the applicant in Table 1.2-4. Length includes 0.25-in. for irradiation growth.
8 As provided by the applicant in Section 6.2.6, "Square Plate Fuels," of the application.
9 As provided by the applicant in Section 3.1.2, "Content's Decay Heat," of the application.

The staff reviewed the materials selection for new fuels, and determined that they are acceptable, consistent with package drawing, and materials specifications with Codes and Standards needed for transportation.

1.4 Criticality Safety Index

The criticality safety index (CSI) for the BRR package remains zero.

1.5. Evaluation Findings

The staff has reviewed the description of the contents and concludes that it meets the requirements of 10 CFR Part 71.

2.0 STRUCTURAL EVALUATION

The purpose of this evaluation is to verify that the proposed changes to the BRR transportation package provide adequate protection against loss or dispersal of radioactive contents and to verify that the package design meets the requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). The following sections document the staff's evaluation of the proposed design changes to the BRR transportation package.

2.1 Description of Structural Design

The BRR package is a shipping container to transport irradiated research reactor fuels. The major structural components of the BRR package are the following:

1) package body
2) impact limiters
3) baskets for fuel

The applicant proposed adding new contents and a basket design for transporting the proposed new contents. The staff discusses the proposed basket design and contents in Sections 1.1.3 and 1.3 of this SER, respectively. The applicant did not propose changes to the designs of the package body and impact limiters.

2.2 Chemical and Galvanic Reactions

The applicant performed visual inspections to identify signs of damaged fuel such as excessive corrosion and erosion, mechanical and wear damage, and plate swelling or blistering. Based on the structural analyses for the square plate fuel elements, corrosion that may be present after many years in storage does not affect the outcome of already approved structural analyses. Section 2.7.1.8 of the application includes the conservative methods that the applicant used to evaluate fuel damage. The applicant provided the basis for these conservative methods in the structural analysis section of the application. Based on the information provided by the applicant in the application, the staff finds that the applicant appropriately identified possible fuel damage of the proposed contents for Revision 5 of the CoC.

2.3 Radiation Effects on Materials

The staff reviewed the information related to fuel irradiation growth in the application. The contents proposed in the application show insignificant irradiation growth (e.g., 0.2-in. growth in 45.5-in. length) compared to previous fuels. The staff concludes that other effects of radiation on materials are negligible (e.g., presence of polyurethane foam) because the payload of the BRR package is heavily shielded with previously approved fuels. The staff finds the information provided by the applicant acceptable and in compliance with 10 CFR 71.43(d).

2.4 Normal Conditions of Transport

The applicant performed structural analyses of the square fuel basket to demonstrate structural adequacy of the basket design for the temperatures specified in 10 CFR 71.71(c)(1). The applicant provided detailed thermal analyses and their results in Chapter 3 of the application. The applicant also provided a summary of the thermal analyses under the NCT in Chapter 2 of the application. Tables 2.1-3, Table 2.2-1, and Table 2.12.8-1 of the application include the basket and fuel weights, the packaging materials, and the numeric values of the allowable stress for the square fuel basket, respectively.

The applicant evaluated the structural performance of the square fuel basket under the NCT using a bounding temperature of 400 degrees Fahrenheit (ºF). As part of its evaluation, the applicant considered the thermal-dependent material properties of the basket and the fuel to calculate acceptable minimum gap clearance between the basket cavity and the proposed contents. The applicant calculated a minimum clearance of 0.08-in. for the square fuel basket after considering the following:

1) the nominal length of any square fuel including the pedestal inserted in the fuel basket to fill the space not occupied by the fuel,
2) the thermal expansion of the materials, and
3) the allowed tolerances between the fuel and the shield plug.

Table 2.6-2b of the application includes the results of these calculations. Based on the results, the applicant concluded that the thermal expansions of the basket and fuel were not a concern. It is noted that the staff previously reviewed and accepted the applicant's calculations of the minimum clearances for the MURR, MITR-II, ATR, and TRIGA baskets. The NRC staff reviewed the information provided by the applicant related to the proposed contents and finds it acceptable.

Based on the staff's review and verification on the applicant's evaluation and its results, the staff finds that the application meets the regulatory requirements of 10 CFR 71.71(c)(1).

2.5 Hypothetical Accident Conditions

The applicant evaluated the square fuel basket for HAC of free drops as required by 10 CFR 71.73, "Hypothetical Accident Conditions." The applicant calculated stresses in the basket under HAC and performed buckling analysis using ASME B&PV Code Case N-284-2. The applicant's approach for the evaluations is identical to the approach previously used for the evaluations of the MURR, MITR-II, ATR, and TRIGA baskets, which the staff previously reviewed and accepted. The applicant considered the several modes of failure (i.e., bending, weld shear, and buckling), which were applicable to the square fuel basket design, and

evaluated the performance of the basket with the end- and side-drop orientations. The applicant used a bounding HAC impact acceleration of 120g in the evaluations.

The applicant provided a summary of the evaluation results and the corresponding margins of safety in Table 2.7-4 of the application, where the smallest margin of safety of any of these evaluations was +0.27. Based on the evaluations, the applicant concluded that the square fuel basket is adequate to support the fuels under HAC free drops. The staff reviewed the evaluations, and verified that the stress results of the evaluations were all within the allowable stresses. The staff finds the results acceptable.

The applicant also calculated a maximum deformation of the fuel elements caused by drops using an energy balance method. Table 2.7-5 of the application includes a summary of the calculated fuel impact deformations. The calculated maximum deformation for the square fuel was approximately 0.053 in. The applicant concluded that this maximum deformation, which was just below one tenth of an inch, is negligible from a structural, shielding, or criticality perspective. The staff reviewed the information, and finds that the deformation is negligible from a structural perspective; therefore, the fuel deformation is acceptable under the HAC drops.

Based on the staff's reviews and verifications on the applicant's analyses and their results, the staff determines that the application meets the regulatory requirements of 10 CFR 71.73(c)(1).

2.6 Evaluation Findings

The staff reviewed documentation provided by the applicant to verify that statements presented by the applicant are acceptable within engineering practices. Based on the review of the statements, representations, and supplemental calculations in the application, the staff concludes that the structural design has been adequately described and evaluated, and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The purpose of this evaluation is to verify that the proposed changes to the BRR transport package provide adequate protection against the thermal tests specified in 10 CFR Part 71 and to verify that the package design meets the thermal performance requirements of 10 CFR Part 71 under NCT and HAC.

3.1 Description of the Thermal Design

Because this amendment is requesting a change in the approved contents only, the packaging design features, with the exception of the baskets discussed below, and codes and standards have not changed. The applicant's licensing strategy for the heat load specification for the new approved contents was to demonstrate that the new contents were bounded by the heat loads of previously approved contents.

The staff has reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71.

3.2 Material Properties and Component Specifications

The material thermal properties and the technical specifications of components are unchanged with the exception of those discussed below. The applicant did not propose changes to the thermal design limits of package materials and components.

The application mentions that the package temperature is maintained within -40 °F and 400 °F. These are conservatively bounding values. The application includes typical bounding maximum temperatures for the 100 °F ambient NCT conditions. The applicant also considers differential thermal expansion during the HAC for a fire event, potential loss of polyurethane foam from thermal decomposition, and applicable codes and standards. The applicant notes that the properties of the materials (including weld and fabrication) for the previous fuel set bounds the new proposed contents and those materials of construction are not impaired by the -40 °F condition, including brittle fracture. For example, the peak temperatures achieved by the packaging components for the transport of the ATR, square fuel, and TRIGA payloads are bounded by those predicted for either the MURR or the MITR-II payloads.

The staff finds the applicant's approach for demonstrating the applicability of previously approved materials properties to the proposed content acceptable considering the conservatism applied in its evaluation and the use of common engineering practices. The staff reviewed the material properties and component specifications used in the thermal evaluation and concludes that these are sufficient to provide a basis for evaluation of the package against the requirements of 10 CFR Part 71.

3.3 Evaluation of Accessible Surface Temperatures

The accessible surface temperatures of the package are bounded by the temperatures produced by the previously approved contents. The staff has reviewed the accessible surface temperatures of the package as it will be prepared for shipment and concludes that they satisfy 10 CFR 71.43(g).

3.4 Normal Conditions of Transport

An expected bounding decay heat of 240 watts (W) was considered by the applicant in the thermal evaluation for the square fuel payload and 380 W for the TRIGA fuel payload which were subsequently bounded by the 1264 W total heat load for the BRR MURR fuel and the 1,200 W for the MITR-II fuel contents. The applicant noted that the package ambient conditions are the same for the all fuel payloads; therefore, the maximum temperature of all package body components for the square fuel and the TRIGA fuel contents would be bounded by the maximum package component temperatures for the MURR and MITR-II fuel contents. It follows that the package inner shell temperature for the MURR fuel contents bounds the maximum package inner shell temperature for the square and TRIGA fuel core configurations, as asserted by the applicant.

3.4.1 Square Fuel Basket

The square fuel basket payloads were evaluated by the applicant qualitatively using similarity arguments as well as establishing that existing payloads bound the new square payloads. The maximum individual fuel compartment heat load was reported as 23 W, however the applicant used 30W to qualify the square type fuel. As illustrated by the applicant, the MURR total heat

load of 1,264 W and the MITR heat load of 1,200 W were significantly higher than the total allowable heat load for the square fuel of 240 W.

The staff reviewed Figure 3.3-9 of the application to confirm that the decay heat per inch of fuel for the MURR and MITR fuel are higher than the square fuel by approximately a factor of 5 over the active fuel region, and as such the square fuel thermal response for the package is bounded by the MURR and MITR-II fuels.

The applicant also provided an evaluation to illustrate that the square fuel would also be bounded by the TRIGA fuel with a maximum heat load of 380 W. The square fuel basket is a slightly different design than the MURR and the MITR-II baskets in that the fuel is arranged radially around the periphery of the basket and also in the center of the basket structure. This makes the heat profile more similar to that of the TRIGA basket, which shares a similar design feature of the fuel spatial distribution, as asserted by the applicant. The TRIGA fuel in this arrangement has 42 percent of the heat load arranged in the interior of the fuel basket whereas the square fuel has only 25 percent of its heat load arranged in the interior of the fuel basket. Based on the percentage of heat load in the center of the TRIGA basket versus the periphery, the temperature profile of the square fuel is expected by the applicant to be lower than the TRIGA fuel, and therefore bounded by the TRIGA fuel thermal response.

The NRC staff reviewed qualitative technical evaluation including the heating profiles and distributions of total heat load for the TRIGA and square fuels by the applicant and finds that the overall TRIGA fuel thermal influence on the package bounds that of the square fuel.

3.4.2 TRIGA Fuel Basket

The added TRIGA fuel types had dimensional differences from the previously approved fuel types, which required a reanalysis by the applicant of the conductive heat flow as the dimensional changes affected the flow path. The evaluation by the applicant demonstrated that the decrease in outer fuel diameter increased the maximum TRIGA element temperature by 7 °F, which reduced the temperature margin from 45 °F to 38 °F.

The NRC staff reviewed the calculation and assumptions and finds this increase in temperature acceptable because of the available remaining margin.

3.4.3 Fuel Temperature

The applicant used similarity and geometric arguments to demonstrate that the square plate fuel was bounded by the previously approved MITR-II fuel plate contents and that the PULSTAR fuel response was bounded by the TRIGA fuel. For the plate fuel, the applicant demonstrated that the materials of construction and geometry of the fuel plate structures are similar such that any minor variations would not significantly affect the performance outcome the fuels evaluated. Because of this similarity, the applicant asserted that the temperature of the MITR-II fuel plates with a higher individual heat load would bound the fuel temperatures of the lower heat load square fuel. In the case of the cylinder type fuels, the applicant demonstrated that the TRIGA fuel temperatures would bound those realized by the PULSTAR based on radial heat flow in a cylinder. In this case, for the same heat load, the applicant asserted that the PULSTAR fuel would have a lower temperature than the TRIGA fuel because the PULSTAR fuel has a smaller diameter. Factoring in the lower heat load per fuel compartment of the PULSTAR fuel the applicant further confirmed that the temperature is bounded by the TRIGA fuel.

The NRC staff reviewed qualitative technical evaluation including the heating profiles and distributions of total heat load for the MITR-II fuel and the square fuel and finds that the overall MITR-II fuel thermal influence on the package bounds that of the square fuel. In addition, the NRC staff reviewed the qualitative and quantitative technical evaluation including the heating profiles, distributions of total heat load, and geometry for the TRIGA fuel and the PULSTAR fuel and finds that the overall TRIGA fuel thermal influence on the package bounds that of the PULSTAR fuel.

3.4.4 Fission Gas Release

The PULSTAR fuel was evaluated by the applicant for a pressure rise due to fission gas release. Using the assumption of 3 percent of fission gas available from breached fuel, the applicant determined that the pressure rise for NCT would be negligible.

Inspection and evaluation of the calculation by the NRC staff for the increase in pressure during HAC confirms this conclusion for the PULSTAR fuel.

3.4.5 Conclusion

The staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not extend beyond the specified allowable limits during NCT consistent with the tests specified in 10 CFR 71.71, “Normal Conditions of Transport.”

3.5 Hypothetical Accident Conditions

As with NCT, the applicant noted that the package ambient conditions are the same for the all fuel payloads. Therefore, the technical argument by the applicant that the maximum temperature of all package body components for the square fuel and TRIGA fuel contents would be bounded by the maximum package component temperatures for the MURR fuel contents is consistent with NCT. It also follows for HAC conditions that the package inner shell temperature for the MURR fuel contents bounds the maximum package inner shell temperature for the square fuel or TRIGA fuel configuration, as asserted by the applicant.

3.5.1 Fission Gas Release

The PULSTAR fuel was evaluated by the applicant for a pressure rise due to fission gas release. Using the assumption of 30 percent of fission gas available from breached fuel, the applicant determined that the pressure rise for NCT would be approximately 2.9 psi. Combining this value with a mean normal operating pressure of 10 psig, which is conservatively higher than the maximum calculated pressure value of 5.2 psig for NCT, the applicant concluded that the pressure generation from fission gas release was significantly below the design pressure of 25 psig.

Inspection and evaluation of the calculation by the staff for the increase in pressure during HAC confirms this conclusion. It is evident to the staff that the pressure generation from fission gas release is significantly below the design pressure of 25 psig.

3.5.2 Conclusion

The staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not exceed the specified allowable short time limits during HAC consistent with the tests specified in 10 CFR 71.73, “Hypothetical Accident Conditions.”

3.6 Evaluation Findings

The NRC staff reviewed the calculations and results of the methods used by the applicant and determined that these are representative and conservative for this application. The NRC staff also determined that using of the higher heat loading of MURR and MITR-II fuel as being a surrogate for the lower heat loading of the TRIGA fuels and square fuels is appropriate and acceptable.

Based on its review of the statements and representations in the application, the staff concludes that the applicant adequately described and evaluated the thermal analysis of the square fuel and additional TRIGA fuel type and that the addition of these fuel types as approved contents meets the requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The applicant did not proposed any changes to the containment evaluation of the BRR package.

5.0 SHIELDING EVALUATION

The purpose of this evaluation is to verify that the BRR package design with the proposed changes continue to provide adequate protection against direct radiation from its contents and to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under NCT and HAC.

5.1 Shielding Design Description

The BRR packaging system consists of a payload basket and a lead–shielded package body. The package body includes a lead-filled side wall, lead filled bottom, an upper shield plug, a closure lid. The total thickness of the side wall of the package is eight inches. Licensing drawing No. 1910-01-01-SAR, sheets 2 and 3, show the general structural layout and dimensions of the cask body. Table 2 below includes an overview of the package components and materials relevant for the shielding evaluation.

The package design includes five custom made baskets for the fuel types authorized for transport. The baskets maintain their geometry under NCT and HAC, as demonstrated in Section 2.7.1.5, “Fuel Basket Stress Analysis,” thereby, maintaining geometry and position of the source as loaded.

Table 2. Package Components and Shielding Materials

Component	**Thickness**	**Material of Construction**
Inner Wall	1 in.	steel
Outer Wall	2 in.	stainless steel
Outer Wall (Shielding)	5 in.	lead
Package Bottom (Through The Centerline)	7.7 in.	lead
Package Bottom - Cover Plate	1 in.	stainless steel
Package Bottom - Inner Forging	1 in.	stainless steel
Shield Plug (Top)	9.5 in.	lead
Shield Plug (Bottom)	1 in.	stainless steel

5.2 Summary Table of Maximum Radiation Levels

Due to the heavy weight of the package, only one package will be transported. Therefore, the dose limits for exclusive use package are used. Table 5.1-1 of the application includes the maximum dose rates for NCT and HAC for MITR-II, square fuel basket, and TRIGA fuel that are transported in baskets custom designed for each fuel type. (Section 1.3 of this SER includes a discussion of the proposed contents.)

The applicant modeled the fuel elements homogenized over the active length of the fuel and distributed across the width of each basket compartment. The fuel element homogenization is a common practice used to simplify complex source geometry with adjustment of source terms along the axial direction based on burnup profile. However, because of the low burnup of the content, the impact of burnup variation on source terms along the axial direction of the fuel is not significant. Tables 5.3-4, 5.3-5, and 5.3-6 of the application contain a summary of the basic fuel dimensions used in the homogenization calculation. For the TRIGA fuel, the applicant assumed a fuel with a cylindrical shape and a source distribution over the fuel pellets. Table 5.2-8 of the application includes the fuel type associated with each dose rate.

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 for all dose rate locations. Therefore, the applicant evaluated the dose rates of the package for all fuel types. The applicant calculated a dose rate at 2 m from the vehicle side and the occupied location in the vehicle (i.e., the driver) at 25 ft. from the centerline of the package.

5.3 Source Specification

The source term for all neutron and gamma fuel elements are developed using TRITON sequence of SCALE 6 (Reference 3) depletion sequence, primarily NEWT and ORIGEN-S. The source terms for MURR, MITR-II, and ATR are developed using the TRITON sequence of SCALE 6 are explained in previous SER. Tables 5.2-1 through 5.2-6 of the application include the input to the TRITON's code and gamma source and neutron source. Table 5.2-3 of the application summarizes the gamma source terms for each fuel assembly design.

5.3.1 Gamma Source

5.3.1.1 TRIGA Reactors

The applicant generated the 26 TRIGA gamma source terms and ran them using the TRITON

sequence of SCALE 6 computer code. The parameters for the TRIGA source terms are summarized in Table 5.2-7 of the application. Table 5.2-7 of the application includes the data used to build TRITON models for the various TRIGA fuel elements.

For all fuel types, since it is desirable to transport the fuel with the shortest possible cooling time, the applicant limited the maximum decay heat to 20 watts per element. The applicant used this limit to determine a bounding source term for the dose rate calculation. The TRIGA fuel is burned to reach about 80 percent ^{235}U depletion, and the fuel cooled until the decay heat reaches maximum 20 W. The cooling time is more than 90 days for about 80 percent depletion. The 90-day cooling time is achieved by reducing the burnup incrementally for the same decay heat.

To simplify the analysis, the set of 26 TRIGA elements is reduced to 16 elements. The applicant stated that the 200 series instrumented elements bound the 100 series standard elements and the 500 series instrumented cluster elements bound the 400 series cluster elements. The residence time for TRIGA reactor fuels in the reactor is around 10 years and reactors do not operate continuously. The applicant used a 4-year (1,461 days) cooling time for all inputs.

The decay heat, radionuclide inventory, and dose rate calculations are all based on an assumed 4-year burnup period or time in the reactor core. However, not all fuel elements will remain in the reactor core for exactly four years. Therefore, the burn up is not constant and would have an effect on decay heat. In order to model this effect, the staff assumed a constant, steady-state power over the burnup period and ensure that the total integrated fuel burnups are equal for each of the variable burnup periods through appropriate adjustment of the fuel element power level.

Stainless steel in the cladding of TRIGA results in ^{60}Co source when irradiated because of ^{59}Co activation in stainless steel. The impurity of ^{59}Co in stainless steel is on average about 800 ppm. The ^{60}Co is also generated by radiation of ^{60}Ni one of the basic element of stainless steel through (n,p) reaction. In the TRITON model, the ^{59}Co and ^{60}Ni are added to account for ^{60}Co activation source.

The mass of stainless steel in the stainless steel clad standard rod for fuel element Type 103 with length of 29.15 in. is 800 g. The applicant assumed that all TRIGA fuel rods with a length less than 30 in. have 800 g stainless steel. The longest rods in TRIGA fuel have an overall length of 45.5 in. This TRIGA fuel is 16.35 in. longer compared to fuel element Type 103. The mass of stainless steel in 16.35 in. of cladding is 195 g. Therefore, the total stainless steel mass of the longest element is approximately 800 g + 195 g = 995 g. The applicant assumed that all rods with a length greater than 30 in. have 1,000 g stainless steel due to increase in rod length.

An effective mass of stainless steel is $M_{AF} + 0.2 \times M_O$ where M_{AF} is the steel mass in the active fuel region and M_O is the steel mass outside the active fuel region and 0.2 is scaling factor for mass outside the active region and is used to estimate the ^{59}Co and ^{60}Ni inputs.

The applicant used data for the IPR-R1 TRIGA reactor for modeling fuel element pitch. This pitch used on the LATTICECELL card for modeling each rod diameter D in the NEWT geometry model. Modeling assumptions were as follows:

1) the pitch is 4.404 cm and TRIGA outer diameter is 3.76 cm,
2) the pitch to diameter ratio is approximately constant,
3) square outer boundary modeled (since a hexagonal outer boundary is not allowed in the model),
4) a fuel temperature of 333K,
5) a cladding temperature of 323K, and
6) a water temperature of 293K.

The applicant obtained the TRIGA reactor design data from the manufacturer and used these data in to revise the shielding analysis.

Based on the above model, gamma and neutron sources are computed for each TRIGA element type for a variety of burnups and cooling times as shown in Table 5.2-8 of the application. Table 5.2-8 of the application provides minimal cooling time for qualification of TRIGA fuel for shipment and the corresponding dose rates at the surface of the package. The applicant modeled a Type 109 fuel element as the bounding conditions in Monte Carlo N-Particle (MCNP) modeling. To determine the bounding TRIGA source, the applicant developed a MCNP model to estimate which source maximizes the dose rate on the surface of the BRR package. In the MCNP model, fuel element Type 109 is modeled and the source is distributed uniformly throughout the fuel matrix. The applicant chose the fuel element Type 109 as model in MCNP because:

1) it has highest ^{235}U enrichment of all TRIGA reactor fuels [^{235}U enrichment (70 percent)] and
2) it is the most reactive among all TRIGA reactor fuels as demonstrated in the criticality analysis.

The fuel matrix is conservatively modeled as fresh fuel for adjustment of neutron source to account subcritical neutron multiplication. Therefore, subcritical neutron multiplication is maximized using the most reactive element.

The dose rate from TRIGA fuel is dominated by neutron radiation for the high-burnup elements because the neutron source increases substantially at high burnups and the BRR package does not have a hydrogenous neutron shield.

Table 5.2-8 of the application provides the dose rates estimated at the side of the package for each TRIGA source. The maximum surface dose rate of 32.2 mrem/hr is for Type 219 with a burnup of 119 MWD per rod and a cooling time of 530 days.

The dose rates are dominated by neutron radiation because of the high burnup and lack of hydrogenous neutron shield in the BRR package. The TRIGA fuel element Type 219 has the highest uranium content between TRIGA reactors and highest burnup. Table 5.2-7 of the application shows elements with higher ^{235}U loadings contain erbium to suppress the excessive reactivity of the fuel at low burnup. The bounding TRIGA Type 219 contains 0.9 wt.% erbium. The erbium is not modeled in developing the source term. This is acceptable based on the study published in NUREG/CR-6802 because burnable poison will harden the neutron spectrum and increases the production of curium, primary contributor to neutron source. The neutron source increases and the side dose rate increases to 33.9 mrem/hr as result of presence of erbium.

TRIGA fuel element Type 219 has the largest source term and is bounding for the shielding analysis (as well as criticality safety), since it is the most reactive fuel element type, as documented in the application. In the MCNP shielding models, a hybrid approach is employed, where the bounding Type 219 source (gamma/sand neutrons/s) is modeled but the pellet material is modeled with the Type 109 material composition since the Type 109 material composition has the most reactive and maximizes the neutron dose rate due to enhanced subcritical neutron multiplication. Table 5.2-8 of the application provides the package surface dose rate for each TRIGA fuel element. These dose rates are based on MCNP models that feature the actual source term for the fuel type but conservatively use Type 109 pellet material compositions. The maximum dose rate occurs for Type 219, with a package surface dose rate of 61.6 mrem/hr. Type 219 bounds all other TRIGA element types for dose rate by a large margin. The package surface dose rate of Type 109 is 12.3 mrem/hr. Type 219 is the bounding TRIGA fuel element source term for all enrichments, burnups, and cooling times.

5.3.1.2 Square Fuel Basket Fuels

The square fuel basket accommodates the fuel with square geometry, including plate fuel elements, individual fuel plate in the loose plate box, and PULSTAR fuel (see the contents described in Section 1.3 of this SER). The square fuel basket heat load is limited to 30 watts per compartment. The fuel for U-Mass (aluminide) is partially burned in the Worcester Polytechnic Institute (WPI) reactor before being transferred to U-Mass for further irradiation. The burnup is the sum of the burnups in both WPI and U-Mass reactors.

The source term for each of these fuels is computed using TRITON computer code. Because these reactors operate as needed, the residence time for the fuel is long and the power history is not uniform. To model power history, the applicant assumed that reactors operated continuously at maximum power until reaching the desired burnup. The exception to this assumption was the RINCS fuel for which the applicant assumed it operated 40 hours per week at its maximum power until it reached 52.5 MWD and a residence time of 1,575 days. Table 5.2-10 of the application shows the key TRITON input data for the seven flat-plate fuel elements.

Table 5.2-11 of the application summarizes the decay heat and depletion results for the seven flat plate elements. The results show that the U-Florida has the largest burnup (87 MWD), with a ^{235}U depletion of 60.1 percent with a decay heat of 5.1 W. The RINCS has the highest decay heat of 22.5 W. Table 5.2-12 of the application shows the gamma source terms for the flat plate elements.

U-Mass (aluminide), U-Florida, and Purdue reactors are the only loose plates authorized for using the loose plate box for shipment. The maximum number of plates per box is 31 as shown in Table 5.2-11. The decay heat for all 31 plates is 11.4 W. However, the maximum number of plates for the U-Florida plates is bounding with 14 because this fuel plate is thicker at the edge, which limits the number of plates that can be put into a loose plate basket. Since the loose plate box is limited to 31 plates, therefore, the source strength for the loose plate box is 31/14 = 2.2 times the strength of the U-Florida fuel element source.

The U-Mass flat plate fuel is representative for all the flat plates that have active length around 23 in. and similar burnup profiles as shown in the Table 5.2-13 of the application. The burnup profile is renormalized to 1.000 and this renormalized profile is the gamma axial source distribution because the gamma source is proportional to the burnup profile. The neutron axial source distribution is shown in Table 5.2-13 of the application.

Two PULSTSAR fuel elements with enrichment 4 and 6 percent are considered in the application. The fuel is similar in design to a typical PWR fuel rod. The fuel has UO_2 fuel rod with zirconium cladding with 5×5 lattice arrangement with X-pitch of 0.607 and Y-pitch of 0.525 in. The fuel length is 24 in. and maximum burnup of 20,000 MWD/MTU and minimum cooling of 1.5 years. Table 5.2-14 of application shows the key TRITON input data for PULSTAR fuel. Figure 5.2-6 provides the PULSTAR fuel diagram. Table 5.2-15 of the application shows the decay heat for two PULSTAR types. The 4 percent has higher decay heat since it has higher ^{235}U depletion since its initial ^{235}U is smaller than 6 percent fuel enrichment.

5.3.2 Neutron Source

The applicant demonstrated that element Type 219 is bounding design in terms of neutron source strength. Table 5.2-8 of the application includes data related to the qualification of the TRIGA fuel. The data include enrichment, maximum burnup, cooling time, decay heat and the package surface dose rates for package containing each TRIGA fuel type. The neutron sources presented are the combined sources from spontaneous fission and (α,n) reactions. The neutron source for TRIGA is from spontaneous fission because there are very limited (α,n) target nuclides present in the fuel. The neutron are mainly resulting from ^{242}Cm and ^{244}Cm, which come from a sequence of neutron capture reaction starting from ^{238}U and subsequently plutonium transmutations. However, there is a very small amount of ^{238}U in the fuel because the fuels are at very high enrichment.

The square plate fuel is 20 percent of ^{235}U enrichment, and enrichment for the PULSTAR fuel has minimum 4 percent ^{235}U enrichment. These fuels have higher neutron sources because these two have the lowest enrichments in their class. The axial source distribution of the neutron source is computed explicitly. Based on NUREG/CR-6700, the curium production in a fuel element is approximately proportional to the 4th power of the burnup and it is non-linear with burnup. Therefore, the neutron source strength is much higher in the middle section of the fuel because the burnup is higher at this section of the fuel. Table 5.2-13 of the application shows the neutron axial source distribution for flat plate fuels.

The neutron source generated by two dimensional TRITON code is the average source per MWD/MTU which does not provide information on the axial variation of the source terms. Due to non-linearity of neutron source with burnup, the neutron fuel for axial burnup is larger than the source computed by TRITON. The neutron sources presented in Table 5.2-13 of the application are multiplied by 1.586 peaking factor and presented in Table 5.2-18 of the application.

U-Florida reactor fuel has the largest neutron source of the flat plate fuels due to the high burnup value. Approximately 85 percent of the neutron source is from (α,n) reactions in the aluminum and silicon in the fuel matrix, and the remaining 15 percent is from spontaneous fission. Eighty-five percent of neutron source comes from ^{242}Cm.

Table 5.2-19 of the application shows the neutron source for the two PULSTAR fuel elements. Because the active length is approximately the same as the flat plate fuels in the square fuel basket, the same axial source distribution is used for PULSTAR fuel. The same neutron peaking factor of 1.586 is also applied to Table 5.2-19 of the application. The 4% enriched fuel element neutron source is more than twice as large as the 6% enriched fuel element for the same burnup, due to lower enrichment of ^{235}U. The PULSTAR neutron source includes both spontaneous fission and (α,n) reactions with osmium-17 (^{17}O) and ^{18}O.

The staff performed confirmatory analysis for the BRR spent fuel transportation package source term. The staff's confirmatory analysis confirmed the applicant's results in the radiation source term analysis and conclusion that the radiation source term are appropriate for the shielding analysis for the BRR package with the new contents.

5.4 Shielding Model

5.4.1 Configuration of Source and Shielding

Section 5.3.1 of the application 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 5.3-1 of the application. The key dimensions relevant to the MCNP model are summarized in Table 5.3-1 of the application and are obtained from Section 1.3.3 of the application. Additional details were not included in this table but were inferred from the drawings.

Table 5.3-2 of the application shows the key parameters for all five basket design and loose plate box. U-Mass (aluminide), U-Mass (silicide), Ohio State, Missouri S&T, U-Florida, Purdue, and PULSTAR are various square fuels and they are transported in the square fuel basket. Figures 5.3-2 through 5.3-7 of the application show the basket and payload for square fuel basket. Only the number of plates are different in modeling the U-Florida, RINSC, Ohio State, and Missouri S&T fuel elements.

The TRIGA and square fuels were modeled as fresh fuel. To minimize self-shielding, the active fuel source is distributed through the fuel and cladding. Table 5.3-3 of the application shows the geometric data used to model TRIGA fuel Type 109 modeled with Type 219 source to maximize subcritical neutron multiplication.

The source in the square plate and PULSTAR are modeled in the fuel because the cladding contains almost no source. For flat plate, the aluminum is conservatively not modeled. For PULSTAR, the outer zircaloy is modeled.

The bounding condition for the square fuel basket in either RINSC, Ohio State, Missouri S&T, U-Florida, PULSTAR or Loose plate, since U-Mass(aluminide), U-Mass (silicide), and Purdue fuel have low burnup and low decay heat. The dose rate on the side of BRR is computed for RINSC, Ohio State, Missouri S&T, U-Florida, and PULSTAR. The applicant considers the U-Florida as the bounding condition for loose plates, since it has the highest burnup and largest neutron source among all loose plate fuel types.

Various length pedestals are used in the package to support different fuel baskets that are designed to hold specific fuel geometries. Except for the TRIGA fuel contents, these various lengths of pedestals assure the top of the fuel always align with the bottom of the top shield plug with a small gap. Table 5.3-3 of the application shows both the active fuel length and total overall length of each fuel element

The applicant also included the insertion of aluminum dunnage into the loose plate box beside the fuel plates to limit the potential for motion of the loose plate during transport. The clearance between the loose plate box and top of the basket is relatively small, which limits the movement of the loose plate box in an axial direction. Therefore, there is no need for the loose plate box cover. Dunnage sheets will prevent the movement of the loose plate when the box is partially loaded with loose plates. Figure 7.1-1 of the application shows the spacer plates for loading the U-Florida fuel elements. The spacer plates is required for loading the for U-Florida fuel as explained in the Section 7.1.2.1 of the application.

The key geometrical parameters for the each basket design are summarized in Table 5.3-2 of the application.

5.4.2 Material Properties

In Section 5.3.2 of the application, 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 ^{235}U, ^{238}U, and aluminum is estimated for each fuel element.

Table 5.3-6 of the application provides fuel composition based on Type 109 for TRIGA reactor. The composition of stainless steel is provided in Table 5.3-9 of the application from SCALE library and most reactive TRIGA modeled. For RINSC, Ohio State, Missouri S&T, and U-Florida have an identical fuel meat composition provided in Table 5.3-7 of the application. The plates are clad in aluminum, and modeled as pure aluminum with a density of 2.7 g/cm^3. In the MCNP model, the square fuel is modeled as U_3Si_2 fuel mixed with aluminum. PULSTAR modeled as UO_2 with enrichment of 4 percent and 6 percent with the lowest possible density of 10.4g/cm^3 of pellet to minimize self-shielding, and data provided in Table 5.3-8 of the application.

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.

5.5 Shielding Evaluation

5.5.1 Methods

The methods used in the shielding analyses for this application are described in Section 5.4.1 of the application. The dose rates were computed using the MCNP5 v1.30 for MURR, MITR-II, and ATR Computer Program while for square fuel basket and TRIGA Version MCNP5 V1.51 used. 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.

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, and square because the fuel is a simple cylinder, the fuel is modeled explicitly, and the source is distributed over the fuel matrix.

The staff reviewed the selection of materials in the shielding analysis models. The staff finds the material compositions selected for shielding calculations are correct and acceptable.

5.5.2 Input and Output Data

A sample input file (gamma source, MITR-II fuel) was included in Section 5.5.3.2 of the application. The staff compared the input file against the gamma sources in Table 5.2-3 of the application and gamma axial distribution in Table 5.2-5 of the application. The staff verified that the information provided by the applicant had the proper model setup, model geometry, and material descriptions.

5.5.3 Flux-to-Dose Rate Conversion

ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors are provided in Table 5.4-1 of the application. This approach is consistent with the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

5.5.4 External Radiation Level

The external radiation levels are described in Section 5.4.4 of the application. The dose rates determined for the bounding payload in each of the five unique baskets: MURR, MITR-II, ATR, TRIGA, and square fuel basket. For MURR, MITR-II, and ATR, only one fuel type is allowed in each basket.

The bounding payload for TRIGA fuel is Type 219 as shown in Table 5.2-8 of the application. The dose rate for square fuel basket are determined at the side of package for square fuel basket candidate payload. The boundary payload is determined by modeling it in the MCNP. The square fuel basket with the following payloads: RINSC, Ohio State, Missouri S&T, U-Florida, PULSTAR (4 percent and 6 percent enriched), and the loose plate box with 31 U-Florida fuel plates modeled in MCNP which have higher burnup and enrichment. The U-Mass (aluminide and silicide) and Purdue fuel elements have low burnups and low decay heat, and are bounded by the other square fuel basket payloads.

The results shown in Table 5.4-2 of the application includes the square fuel basket payloads ranked in order from high to low dose rate. The bounding square fuel basket payload is the loose plate box with 31 U-Florida plates per box. The Florida loose plate dose rate is twice of that of the U-Florida fuel and it bounds the dose rates for the square fuel basket fuel.

The applicant studied the various scenarios of the fuel configurations under hypothetical accident conditions. Because of the wide range of lengths available for TRIGA fuel, the TRIGA models are run with the fuel shifted up or down. The dose rates at the top and side of the TRIGA fuel package are reported for the fuel shifted up, and the dose rates at the bottom of the TRIGA fuel package are reported for the fuel shifted down. The other fuels have standard lengths and are modeled consistent with their location in the package.

5.5.5 Confirmatory Analyses

The staff reviewed the applicant's models used in the shielding analyses. The staff examined the code input file in the calculation packages and confirmed that the applicant used the proper material properties and bounding conditions. The staff also reviewed the engineering drawings to verify that proper geometry dimensions were translated to the analysis model.

The staff reviewed the material properties presented in the application to verify that these were appropriately referenced and used.

The staff reviewed the applicant's source term and shielding analysis. The staff also performed confirmatory analyses of the source term. Confirmatory analysis performed by the staff on source term evaluations using the SCALE 6.1 computer code with the ORIGEN/ARP isotopic depletion and decay sequence with the 238-group ENDF/VII cross section library. Using irradiation parameter assumptions similar to the applicant's, the staff obtained bounding source terms that were similar to, or bounded by, those determined by the applicant and finds the applicant's result acceptable. Confirmatory analysis also performed for the shielding analysis and staff concluded that the applicant's result was also acceptable. Based on its review of the application and its confirmatory analyses, the staff finds that the applicant has correctly calculated the sources and the dose rates. The methods are appropriate for these types of calculations and results are acceptable.

5.5.6 Conclusions

The staff concludes that the design of the shielding system for the BRR complies with 10 CFR Part 71. The staff has reasonable assurance that the shielding evaluation of the shielding system provided by the applicant will allow for the safe transport of the new contents in the BRR packaging system. This finding is based on a review of the specifications in the application, the applicable regulations, appropriate regulatory guides, staff confirmatory (including calculations and modeling) analyses, and accepted engineering practices. The staff reviewed the external radiation levels under NCT and HAC and found reasonable assurance that these satisfy 10 CFR 71.43(f) and 71.51(a)(1).

5.6 Evaluation Findings

The staff evaluated the adequacy of the description, methods, and analyses of the package design bases related to the shielding evaluation of the BRR package and found them acceptable. The staff reviewed the maximum dose rates for NCT and HAC and determined that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51.

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

6.0 CRITICALITY SAFETY EVALUATION

The objective of this review is to verify that the BRR spent fuel transportation package design containing the new contents as specified in Chapter 1 of the application meets the criticality safety requirements of 10 CFR Part 71. The staff reviewed the description of the package design and criticality safety analyses presented in Chapters 1 and 6 of the Safety Analysis Report for the BRR transportation package and the following sections of this report document the staff's criticality safety evaluation for this package design.

6.1. Description of Criticality Safety Design

The BRR packaging system is a cylindrical lead lined package designed for transportation of Type B quantity radioactive materials and/or solid fissile materials. The packaging consists of a payload basket, a lead-shielded package 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 leak-tight containment of the radioactive contents under NCT.

Five fuel basket types are used to properly position the fuel within the package cavity. One of these baskets is the square fuel basket that is designed to house all of the square fuel element and loose plates as listed in Section 1.2.2.5 of the application. A new loose plate box is added to the design. The loose plate basket is a square stainless steel box and is used to house loose fuel plates of U-Mass (aluminide), U-Florida, and Purdue reactor fuel elements. Loose fuel plates are the unassembled square fuel elements from the fuel elements of the afermentioned reactors. The loose plate box is loaded into a square fuel basket.

The fuel basket geometry and loose plate box are used to provide control of the spacing between the fuel elements. The basket geometry also may limit the number of fuel elements or loose plates that may be shipped per package. Spacers and dunnage are used to control the space between the loose plates and prohibit relative movement of the fuel plates in the basket. The separation provided by the packaging is to maintain the geometric form of the fissile materials for criticality safety. No neutron poison is utilized in the package.

The initial design of the loose plate basket\box used aluminum as the material of construction. The design also included significant free space inside the loose plate basket for the loose plates to move inside the basket because the maximum allowable content (31 loose plates) would not fill up the space in the loose plate basket. The staff was concerned with a potential geometry change of the content under normal conditions of transport, which is not allowed per the requirement of 10 CFR 71.55(d)(2). In its responses to the staff's RAIs, the applicant changed the material of construction of the loose plate basket from aluminum to stainless steel and added requirements for using dunnage and spacers to restrict the movement of the loose plates inside the loose plate basket.

The package is designed for wet loading and unloading. The package is under-moderated when fully loaded.

6.1.1 Spent Nuclear Fuel Contents

The applicant requests to revise the CoC to add the following new fuel: 1) 21 new TRIGA fuel types, 2) PULSTAR fuel elements, 3) flat plate fuel elements, and 4) loose fuel plates from the various fuel elements in square loose fuel plate baskets as new authorized contents. Specifically, the requested new contents include TRIGA fuel types 105, 107, 119, 201, 205, 207, 217, 219 303, 305, 317, 319, 403, 405, 417, 419, 503, 505, 517, and 519. The flat or slightly curved plate types include RINSC, U-Mass, Ohio State University, Missouri University of Science and Technology, State University of Florida, and Purdue University experimental reactors. However, only the loose plates from U-Mass (aluminide), U-Florida, and Purdue fuel elements are allowed in the loose plate box. Up to 31 loose fuel plates are allowed per box in a loose plate box and a maximum of eight loose fuel plate boxes can be loaded into a BRR package. For U-Florida reactor fuel, only 17 fuel plates can be loaded into a loose plate basket because of this fuel plate is much thicker. Detailed characteristics of these fuel types are

defined in Table 1.2-1 and 1.2-3 of the application for TRIGA fuel elements and square fuel plates respectively. The difference between fuel element and fuel plate is that fuel element is an assembly of multiple fuel plates.

6.1.2 Summary Table of Criticality Evaluations

The applicant performed criticality safety analyses for the BRR spent fuel transportation package with the requested contents. Table 6.1-1 of the application provides a summary of the results of the applicant's criticality analyses. Through these analyses, the applicant demonstrated that the BRR package with the requested contents meets the requirements of 10 CFR 71.55.

The applicant performed criticality safety analyses for packages under HAC. For the aluminum plate fuels, the most reactive credible configuration is utilized by maximizing the gap between the fuel plates. Maximizing this gap maximizes the moderation and hence the reactivity because the system is under-moderated. In all single package models, a 12 in. of water reflection is assumed. The maximum reactivity of the criticality calculations for each of the fuel element types are summarized in Table 6.1-1 of the application. The maximum calculated k_s is 0.827, which occurs for the HAC array case for MURR payload. The maximum reactivity is less than the upper subcriticality limit (USL) of 0.9209. The most reactive MITR-II, ATR, TRIGA, PULSTAR, and square plate fuel cases are well below the USL. Note that PULSTAR, the LPB, and the square plate fuels are transported in the SFB.

Based on its analyses, the applicant determined that the BRR package containing MURR fuel elements has the highest k_{eff} among all of the packages. The maximum k_{eff} is 0.827 which is under the USL of 0.9209 as established by its code benchmarking calculation. The criticality safety index (CSI) of this package is 0.

6.2 General Considerations

The applicant analyzed the criticality safety of the BRR package with the requested contents. The applicant analyzed the criticality safety of a single package with the various contents to demonstrate that the package meets the regulatory requirements of 10 CFR 71.55 under NCT as well as under HAC. The applicant also analyzed the criticality safety of array of packages under HAC to demonstrate that the package meets the regulatory requirements of 10 CFR 71.59.

On page 6.4-8, the applicant states the following:

> "Results are provided in Table 6.4-10. As expected, the reactivity decreases as plates are removed from the LPB. Therefore, the criticality analysis bounds the loading and unloading operations of 10 CFR 71.55(b)."

Based on the information provided by the applicant, the staff finds that the HAC analyses bound the requirements of 71.55(b) and the approach acceptable.

6.2.1 Model Configuration

The applicant used a simplified configuration of the package in its criticality safety analyses. Impact limiters are ignored in all cases and 12 in. of water reflector is used for the single package analysis. For an array of packages, removing the impact limiters conservatively

minimizes the separation between the packages and increases the reactivity. The package body itself is simply modeled as cylinders of steel-lead-steel without modeling the minor package details, as these minor details have a negligible effect on the system reactivity.

The applicant provided model geometry in Figure 6.3-1 of the application. The key model dimensions are determined based on the drawings provided in Section 1.3.3, "Packaging General Arrangement Drawings," of the application.

The applicant modeled the fuel baskets as undamaged in all NCT and HAC models. The baskets have been shown to be elastic in all accident scenarios and maintain their geometry (see Section 2.7.1.5 of the application, "Fuel Basket Stress Analysis"). The fuel is modeled in the most reactive axial location and the spacer pedestals are ignored. Plate fuel is also modeled as undamaged in all models (with end structures conservatively removed), as has also been demonstrated by the structural analyses that the plate fuel maintains its structural integrity during accident conditions (see Section 2.7.1.6 of the application, "Fuel Impact Deformation").

PULSTAR fuel is similar in design to commercial PWR fuel. Therefore, it is conservatively assumed under HAC that the pitch of PULSTAR fuel rods expands until the rods at the outer rows reach the zircaloy fuel box, which bounds potential fuel damage.

The applicant took no credit for the LPB to constrain the loose plates under HAC. To assure the calculation conservatively bound any possible damage to the LPB under HAC, the entire LPB is removed from the MCNP models, and the fuel plates are arranged within the SFB cavity in the most reactive orientation, which is an unlikely condition, even for package under HAC. In those cases, the plates are modeled in their nominal geometry (as flat plates).

The staff reviewed the applicant's criticality safety analysis model sample input files. Based on the information provided, the staff determined that the applicant correctly constructed the models with consideration of potential manufactural tolerances of the packaging (overpack and fuel baskets) and the fuels, material uncertainties, and computational bias and uncertainties. Based on its review, the staff determined that the applicant's model configurations are conservative and acceptable.

6.2.2 Material Properties

The applicant provides the fuel meat compositions in Tables 6.2-2, 6.2-3, 6.2-6, and 6.2-8 of the application for MURR, MITR-II, ATR, and TRIGA fuel, respectively. Fuel meat compositions for PULSTAR and square plate fuels are provided in application Table 6.2-10 and Table 6.2-12, respectively. For all fuels, aluminum structural material is modeled as pure aluminum with a density of 2.7 g/cm^3. Similarly, all zirconium alloy is modeled as pure zirconium with a density of 6.5 g/cm^3.

For the TRIGA fuels, the applicant used material compositions in the standard material compositions provided in the SCALE material library (Standard Composition Library, ORNL/TM-2005/39, Version 5, Vol. III, Section M8, April 2005) and is provided in Table 6.3-3 of the application.

For the TRIGA fuels that contain a zirconium rod in the center of the fuel element, the zirconium rod is modeled as pure zirconium with a density of 6.5 g/cm^3. The graphite reflectors in the TRIGA fuel elements are modeled as pure graphite with a density of 1.6 g/cm^3. The density is

obtained from the TRIGA benchmark experiments (LEU-COMP-THERM-003) listed in the International Handbook of Evaluated Criticality Benchmark Experiments.

For packaging materials, the inner and outer shells of the package are constructed from stainless steel Type 304. The stainless steel composition and density utilized in the MCNP models are provided in Table 6.3-3 of the application. Lead is modeled as pure with a density of 11.35 g/cm^3. Water is modeled with a density ranging up to 1.0 g/cm^3 with the chemical compound H_2O.

The staff reviewed the selection of materials in the criticality safety analysis models. The staff finds the material compositions selected for nuclear calculations are correct and acceptable because the material properties used in the models are taken directly from well established sources.

6.2.3 Computer Codes and Cross-Section Libraries

The applicant used MCNP5 Version 1.30 in its criticality safety analyses for the BRR package for the additional 21 TRIGA fuel types and seven types of loose plates in the loose plate basket. A combination of different cross-section libraries was used to meet the need for the criticality safety calculations. The applicant includes S(α, β) modifications for neutron upper scattering in light water moderator and zirconium hydride in the TRIGA fuel packages.

The applicant initially used 2,500 particle histories per cycle, a total of 250 cycles per calculations with 50 cycles discarded in its MCNP criticality safety analysis models. The staff was concerned with whether calculations had correctly converged or not because it typically takes much longer (more cycles and number of particles per cycles) for the models of loosely coupled systems like the BRR fuel packages to converge in search of eigainvalue and source distribution. To help determine appropriate convergence of the MCNP calculations, the developer of the code implemented a parameter named Shannon Entropy in the MCNP code version 5.1 and later. However, the applicant stated that the code version it used does not have this feature. As a remedy, the applicant revised the models to run 1,050 cycles, 5,000 particles per cycle, skipping the first 50 cycles. The staff ran an example case and finds the revised models assured correct convergence of the eigenvalue calculations.

The staff reviewed the applicant's selection of computer code and cross sections. The staff finds that the MCNP computer code is a widely used and rigorously validated computer code for criticality safety analyses and is adequate for this application. The cross sections selected encompassed the energy and temperature ranges of the system. On this basis, the staff determined that the computer code and cross sections selected are acceptable and adequate for criticality safety analyses of this application.

6.2.4 Demonstration of Maximum Reactivity

The applicant searched for the maximum reactivity by varying the quantity of the contents in the package under HAC. The applicant demonstrated compliance with 71.55(b), by performing a few sensitivity studies on reactivity changes versus number of plates for package under HAC. The staff determined that the applicant's criticality safety analyses for a package under HAC provide a bounding condition for the package under NCT (i.e., the applicant analyzed a flooded package under HAC and demonstrated that the package meets criticality safety requirements and also there is a sufficient safety margin in the k_{eff} value compared with the USL). Therefore, the staff finds the applicant's approach acceptable.

6.2.5 Confirmatory analyses

The staff performed confirmatory analysis for the most reactive BRR spent fuel transportation package, which is the loose plate package with the Ohio State fuel plates and maximum of 31 plates per loose plate basket. The staff's confirmatory analysis confirmed the applicant's criticality analysis results and conclusion that the package remains subcritical with sufficient safety margin under HAC.

6.2.6 Single package evaluation

The applicant evaluated the criticality safety of a single package as loaded, under NCT, and under HAC for each of the contents the applicant requested for approval. The applicant modeled the package with some conservative simplifications. Specifically, the impact limiters are not included and the package is reflected with 12 in. of water in the models. The package body itself is simply modeled as cylinders of steel-lead-steel without modeling the minor package details, as these minor details have a negligible effect on the system reactivity. In the cases for array of packages, removing the impact limiters conservatively minimizes the separation between the packages and increases the reactivity.

For square plate fuels, the fuel element geometry is consistent with the most reactive fuel element models, including tolerances, as determined in Section 6.9.2 of the application, "Parametric Evaluations to Determine the Most Reactive Fuel Geometries."

The TRIGA fuel package models are developed for a bounding subset of the allowable rod types. The bounding rods (catalog numbers 109 and 119) contain the highest enrichment fuel (136 g ^{235}U at 70 wt.%) and the highest ^{235}U loading (163 g ^{235}U at 20 wt.%), respectively. Other permissible rods are all enriched to 20 wt.% ^{235}U. Given that there is no other significant differences in the design of other permissible rods, Type 109 and 119 are identified as bounding. Molybdenum discs, a mild neutron absorber, are conservatively ignored.

The applicant modeled fuel element at the nominal pitch with a maximum ^{235}U enrichment of 6 percent for the PULSTAR fuel package under NCT. No credit is taken for the fuel burnup.

The square plate fuels are modeled with minimum cladding thickness and maximum possible channel spacing, as the analysis for MURR, MITR-II, and ATR fuels, which have been approved previously, indicated this is the most reactive configuration for plate-type fuel.

The fuel baskets are modeled as undamaged in all NCT and HAC models. The fuel is modeled in the most reactive axial location and the spacer pedestals are ignored. Plate fuel is also modeled as undamaged in all models (with end structures conservatively removed). These modeling assumptions provide some additional safety margin in the calculated k_{eff} values.

The applicant demonstrated compliance with 71.55(b) by performing adequate sensitivity studies on reactivity changes versus number of plates for the package under HAC. The staff finds this approach acceptable.

The staff reviewed the applicant's criticality safety analyses for single package as designed under NCT. The k_{eff} values of the package with various contents under NCT are very low because no water is assumed in the package.

The applicant performed criticality safety analyses for package under HAC with a fully flooded package cavity. The results of the criticality safety analyses show single package meets the regulatory requirements of 10 CFR 71.55(e). The infinite array of the packages loaded with MURR fuel is the most reactive for package under HAC; the package has a k_{eff} of 0.827. This case bounds all packages under NCT. Table 6.1-1 of the application provides a summary for the results of the criticality safety evaluations for the packages under NCT and HAC.

6.2.7 Evaluation of Package Arrays under NCT

The applicant calculated the reactivity of an infinite array of package with assumption that the cavity of the package remains dry (not flooded). The k_{eff} values of the package with various contents under NCT are very low because there is no water assumed in the package.

6.2.8 Evaluation of Package Arrays under Hypothetical Accident Conditions

The applicant calculated the k_{eff} of an infinite array of the MURR packages, which is the most reactive package under HAC. The result show the k_{eff} is 0.827 for an infinite array of packages under HAC and the k_{eff} is below the USL. Therefore, 2N of the package is subcritical.

The staff reviewed the applicant's criticality safety analyses for array of packages under HAC and finds that the applicant's analyses correct and acceptable.

Based on the criticality safety analyses of the package under NCT and HAC, the staff finds that the applicant has correctly calculated the criticality safety index. The applicant's choice of CSI equal to zero is correct because an infinite array of packages under both NCT and HAC is subcritical.

6.2.9 Benchmark Evaluations

The applicant benchmarked the MCNP v1.5 code with the selected cross sections for the BRR fuel transportation packages loaded with requested contents. The fuel types analyzed fall into four distinct categories (1) high-enriched aluminum plate fuel, which includes MURR, MITR-II, and ATR, (2) low-enriched plate fuel with an aluminide or silicide fuel matrix, which includes all of the Square plate fuels, (3) UO_2 rods of the PULSTAR fuel, and (4) zirconium hydride (TRIGA) fuel. The applicant performed separate benchmark analyses for categories (1) and (4) and applied to all four fuel categories. The critical experiment benchmarks are selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments based upon their similarity to the packaging and contents.

The applicant supplies the results for code benchmarking analyses and USL. The selected critical experiments include systems with various enrichment of uranium homogeneously mixed with water and heterogeneous array of rods with various rod geometry and pitches. The USL for the content is 0.9209.

The staff reviewed the applicant's code benchmarking analyses and finds that the applicant has adequately benchmarked the computer code together with cross section library for this specific application. The selected critical experiments cover the range of the parameters of the contents. In particular, the applicant selected the critical experiments for the TRIGA fuels that have very unique characteristics in the material compositions, i.e., zirconium hydride mixed with uranium.

6.3 Evaluation Findings

The staff reviewed the application and the applicant's responses to the staff's requests for additional information (RAIs) under the regulations of 10 CFR Part 71. The staff followed the guidance provided in NUREG-1617 in its review. Based on the information presented in the safety analysis report, the staff's independent calculations, the applicant's responses to the RAIs, clarification teleconferences, and the commitment by the applicant to include the modified calculations for the requested contents, the staff found with reasonable assurance that the packages with additional 21 types of TRIGA fuel rods and loose plate fuel in loose plate baskets as allowable contents in the BRR package meets the regulatory requirements of 10 CFR 71.55 and 71.59.

Based on its review of the information and representations made by the applicant in the application, the staff finds reasonable assurance that the package design with the proposed contents meets the criticality requirements identified in 10 CFR Part 71.55 and 71.59.

7.0 PACKAGE OPERATIONS

The purpose of this evaluation is to verify that the proposed changes to the operating controls and procedures of the BRR transport package meet the requirements of 10 CFR Part 71. The applicant will confirm that no residual water is present after vacuum drying of new fuels. The criteria for dryness is that the pressure within the package must not exceed 3 Torr after 30 minutes of isolation from the vacuum pump. Subsequently, the package cavity is backfilled with helium. Gas generation from radiolysis, or (galvanic) corrosion of the fuel, will not occur under these conditions. The staff finds that the drying process is in accordance with ASTM Standard Guide for Drying of Spent Nuclear fuel (ASTM C1553).

The applicant requires dunnage sheets to prevent the movement of the loose plate when the box is partially loaded with loose plates. Figure 7.1-1 of the application shows the spacer plates (dunnage) for loading the U-Florida fuel elements. The spacer plates is required for loading the for U-Florida fuel as explained in the Section 7.1.2.1 of the application. The applicant will also use stainless steel pedestals to accommodate fuel shorter in length than the compartment of the fuel basket appropriate for a particular payload.

The staff added conditions to the certificate of compliance requirements for using dunnage (and stainless steel spacer pedestals), when transporting a particular payload. These condidions are included in Condition No. 6 of the CoC (see "Conditions"section of this SER).

Based on review of the statements and representations in the application and conditions imposed in the CoC for the BRR package, the staff concludes that operating controls and procedures for the BRR package meet the requirements of 10 CFR Part 71, and that these controls and procedures are adequate to ensure the safe use of the package.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

The staff reviewed the change to Chapter 8 of the application. Based on review of the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71, and that the maintenance program is adequate to ensure packaging performance during its service life.

REFERENCES

1. Title 10, *U.S. Code of Federal Regulations*, Part 71 (10 CFR Part 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. J.W. 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.

CONDITIONS

The CoC includes the following condition(s) of approval:

Revised Condition No. 2.b., "Revision Number," from "4" to "5."
Revised Condition No. 3.b., "Title and Identification of Report or Application," to change the application date to May 30, 2016, which corresponds to the consolidated application for the BRR.

Revised Condition No. 5.(a)(2), "Description," as follows:

> Moved the last sentence of the second paragraph including a description of the impact limiters to the third paragraph under Condition No. 5.(a)(2), since the sentence in question related to description of the baskets. Also, reflected the addition of the squre fuel basket and the square box. The revised text is as follows:
>
> > "The purpose of the Model No. BRR package is to transport irradiated fuel elements or loose plates of a square fuel element from various test and research reactors. The package is comprised of a lead–shielded package body, payload basket, square loose plate box, an upper shield plug…
> >
> > *Impact Limiters.* Impact limiters are attached to each end of the package body. [Editorial change for clarity]
> >
> > *Fuel Baskets*. There are five baskets used with the package, one for each type of fuel transported. The baskets are made from welded construction using ASTM Type 304 stainless steel in plate, bar, pipe, and tubular forms. Each basket has a diameter of 15.63 inches and a length of 53.45 inches, and features a number of cavities that fit the size and shape of the fuel. The basket for square fuel accommodates two types of fuel assembly: (1) flat-type fuels and (2) a 5x5 array of fuel rods enclosed within a rectangular can.

Spacer Pedestals. For fuel elements or assemblies shorter than the length of a basket cavity, spacer pedestals are used in each cavity, as required, to support the fuel elements at the top of the basket. All spacer pedestals are made of stainless steel

Square Box or Loose Plate Box. A square box accommodates square fuel loose plates. A loose plate box is used to transport up to 31 loose plates per box. The square fuel basket and loose plate box are made of stainless steel."

Revised Condition No. 5.(a)(3), "Drawings," as follows:

"1910–01–03–SAR, "BRR Package Fuel Baskets SAR Drawing," Sheets 1-4, Rev. 6"

Revised Condition No. 5.(b)(1)(iv), "Type and form of material," as follows:

"*Irradiated TRIGA fuel elements*. Table 1.4 includes the dimensions of pre-irradiated Training, Research, Isotopes, General Atomics (TRIGA) fuel elements. The TRIGA fuel matrix is uranium mixed with zirconium hydride. The BRR package is limited to the transportation of the following types of TRIGA fuel:

1. Standard 100 series.
2. Instrumented 200 series. The fuel region is as the same as 100 series but contain thermocouples used to measure temperature during reactor operation. Instrumented rods may be longer than 100 series.
3. Fueled Follower Control Rods (FFCR) (300 series). The rods contain boron carbide neutron absorber outside the active fuel region.
4. Cluster Rods (400 series). It is typically built with three or four cluster rods to make a cluster assembly.
5. Instrumented Cluster Rods (500 series). Fuel is the same as cluster rod but thermocouples used to measure temperature during reactor operation. Instrumented cluster rods may be longer.

Cluster rods (i.e., TRIGA fuel series 400 and 500) must be disassembled from the cluster assembly for transport in the BRR package.

Table 1.4. Characteristics of Pre-Irradiated TRIGA Fuel

Type	ID[1]	Cladding	Fuel Length (in.)	U (wt. % Fuel)	^{235}U (wt. %)	U (g)	^{235}U (g)	Fuel OD[2] (in.)	Rod OD (in.)	Cladding Thickness (in.)	H/Zr	Overall Length[3] (in.)	Erbium (wt. %)
Standard 100 series	**101**	Aluminum	14	8.0	20	166	32	1.41	1.48	0.03	1.0	28.62	0
	101		15	8.5	20	189	37	1.41	1.48	0.03	1.6	28.62	0
	103	Stainless Steel	15	8.5	20	197	39	1.44	1.48	0.02	1.6	29.15	0
	105		15	12	20	285	56	1.44	1.48	0.02	1.6	29.15	0
	107		15	12	20	271	53	1.4	1.48	0.02	1.6	30.14	0
	109		15	8.5	70	194	136	1.44	1.48	0.02	1.6	29.15	1.2
	117		15	20	20	503	99	1.44	1.48	0.02	1.6	29.93	0.5
	119		15	30	20	825	163	1.44	1.48	0.02	1.6	29.93	0.9
Instrumented 200 series	**201**	Aluminum	15	8.5	20	189	37	1.44	1.48	0.03	1.6	28.78	0
	203	Stainless Steel	15	8.5	20	197	39	1.44	1.48	0.02	1.6	45.5	0
	205		15	12	20	285	56	1.44	1.48	0.02	1.6	45.5	0
	207		15	12	20	271	53	1.4	1.48	0.02	1.6	45.5	0
	217		15	20	20	503	99	1.44	1.48	0.02	1.6	40.35	0.5
	219		15	30	20	825	163	1.44	1.48	0.02	1.6	40.35	0.9
Fueled Follower Control Rods (FFCR) (300 series)	**303**	Stainless Steel	15	8.5	20	163	32	1.31	1.35	0.02	1.6	44	0
	305		15	12	20	237	47	1.31	1.35	0.02	1.6	44	0
	317		15	20	20	418	82	1.31	1.35	0.02	1.6	44	0.5
	319		15	30	20	685	135	1.31	1.35	0.02	1.6	44	0.9
Cluster rods (400 series)	**403**	Stainless Steel	15	8.5	20	166	33	1.37	1.41	0.02	1.6	30.38	0
	405		15	12	20	243	48	1.37	1.41	0.02	1.6	30.38	0
	417		15	20	20	427	85	1.37	1.41	0.02	1.6	30.38	0.5
	419		15	30	20	710	141	1.37	1.41	0.02	1.6	30.38	0.9
Instrumented cluster rods (500 series)	**503**	Stainless Steel	15	8.5	20	166	33	1.34	1.41	0.02	1.6	45.5	0
	505		15	12	20	243	48	1.34	1.41	0.02	1.6	45.5	0
	517		15	20	20	427	85	1.34	1.41	0.02	1.6	45.5	0.5
	519		15	30	20	710	141	1.34	1.41	0.02	1.6	45.5	0.9

CoC footnotes:
1. General Atomics catalog numbers are not necessarily unique. TRIGA elements with the same ID could have different fuel parameters. Table 1.4 includes two variants of the Type 101 element
2. Outer Diameter.
3. Overall length includes 0.25 inches for irradiation growth.

"The maximum length of a TRIGA fuel element, including irradiation growth, is 45.50 inches. For all fuel elements, stainless steel spacers are utilized within the TRIGA baskets… Table 1.5 includes parameters for irradiated TRIGA fuel."

Table 1.5 Maximum Burnup and Minimum Cooling Time for TRIGA Fuel Elements[4]

TRIGA Fuel Type (Enrichment)	Maximum Burnup (MWD)	Minimum Cooling Time (days)
101 (8.0%)	23	90
201/101 (8.5%)	26	90
109	88	350
	70	250
	52	170
	34	90
203/103	27	90
205/105	39	120
	33	90
207/107	38	120
	33	90
217/117	71	280
	52	180
	34	90
219/119	122	600
	91	370
	63	220
	34	90
303	22	90
305	32	90
317	58	210
	46	150
	34	90
319	97	420
	76	290
	55	180
	34	90
503/403	23	90
505/405	33	90
517/417	60	220
	47	150
	34	90
519/419	101	430
	79	290
	56	180
	34	90

CoC footnote No. 4: Based on an in-core residence time of 4 years resulting on a decay heat less than or equal to 20 W. Not applicable to fuel with an in-core residence time less than 4 years with a decay heat greater than 20 W."

Added Condition No. 5.(b)(1)(v), "Type and form of material," as follows:

"(v) *PULSTAR Fuel*. Table 1.6 includes the characteristics of the PULSTAR fuel. A 5×5 array of fuel rods enclosed within a rectangular can. Each fuel rod contains cylindrical uranium oxide fuel pellets. The weight of a PULSTAR element is

48 lb, including a spacer pedestal. The maximum heat load of the square fuel basket per compartment is 30 W.

Table 1.6. Characteristics of PULSTAR Fuel

Parameter	Value
Nominal ^{235}U Enrichment (%)	4.0/6.0
Fuel matrix	UO_2
Maximum burnup (MWD/MTU)	20,000
Decay time (years)	1.5
Maximum fuel pellet diameter (in.)	0.423
Minimum cladding thickness (in.)	0.0185
Cladding material	Zirconium alloy
Maximum cladding OD (in.)	0.474
Maximum active fuel length (in.)	24.1
Fuel rod pitch X (in.)	0.607
Fuel rod pitch Y (in.)	0.525
Box outer dimensions (in.)	3.15 x 2.74
Box thickness (in.)	0.06
Box material	Zirconium alloy
Maximum overall length (in.) (1)	38.23

Notes:
1. Maximum length includes 0.25 in. for irradiation growth.
2. The references section contains the original application and the supplements provided as part of the review process."

Added Condition No. 5.(b)(1)(vi), "Type and form of material," as follows:

"(vi) *Square Fuel and Loose Plates (excluding PULSTAR)*. Table 1.7 includes the main characteristics of square fuel and square-loose-plate fuel. These types of fuel have a square, or nearly square-rectangular cross section. The flat-type fuels consist of either a uranium-oxide dispersion or uranium-silicide dispersion meat in an aluminum matrix, bonded with an aluminum alloy cladding. The maximum heat load of the square fuel basket per compartment is 30 W.

Table 1.7. Square Plate Fuel Characteristics

Parameter	RINSC	Ohio State	Miss. S&T	U-Florida	Purdue	U-Mass (Al)	U-Mass (Si)
^{235}U loading (g)	275±7.7	200±5.6	225±6.3	175±4.9	129.92±2.52	167±3.3	200±5.6
Nominal ^{235}U enrichment (%)	19.75	19.75	19.75	19.75	19.75	19.75	19.75
Fuel matrix	U_3Si_2+Al	U_3Si_2+Al	U_3Si_2+Al	U_3Si_2+Al	U_3Si_2+Al	UAl_x	U_3Si_2+Al
Maximum burnup per fuel element (MWD)	52.5	64.0	74.0	87.0	0.57	9.7	9.7
Minimum decay time (D)	120	120	365	120	120	1,000	1,000
Nominal fuel meat width (in.)	2.395	2.395	2.395	2.395	2.395	2.320	2.395
Nominal fuel meat thickness (in.)	0.02	0.02	0.02	0.02	0.02	0.03	0.02
Nominal fuel plate thickness (in.)	0.05	0.05	0.05	0.05	0.05	0.06	0.05
Nominal active fuel length (in.)	23.25	23.25	23.25	23.25	23.25	23.25	23.25
Number of fuel plates	22	16	18	14	14	18	16
Maximum channel spacing (in.)	0.099	0.127	0.139	0.117	0.175	0.119	0.122
Weight (lb)	14	12	14	10	10	12	12
Maximum overall length (in.) ④	39.75	35.25	34.50	27.38	32.49	39.75	39.75
Maximum cross section (in.)	3.097× 3.097	3.05× 3.05	3.036× 3.212	2.9×2.424	3.011× 3.011	3.097× 3.097	3.097× 3.097
Loose plate ④⑤	no	no	no	yes②	yes③	yes①	no

Notes:
1. U-Mass (Al) loose plates have a ^{235}U loading of 9.28 ± 0.18g and dimensions of 2.78 inches wide by 24.88 inches long.
2. U-Florida loose plates have a ^{235}U loading of 12.5 ± 0.35g and dimensions of 2.85 inches wide by 25.88 inches long.
3. Purdue loose plates have a ^{235}U loading of 9.28 ± 0.18g and dimensions of 2.85 inches wide by 25.88 inches long.
4. Maximum length includes 0.25 inches for irradiation growth.
5. Loose plates shall be extracted from fuel elements that meet the per-element burnup limits provided in this table."

Revised Condition No. 5.(b)(2)(iv), "Maximum quantity of material per package," as follows:

"26 types of TRIGA fuel."

Added Condition Nos. 5.(b)(2)(v) to (vii) as follows:

(v) For the contents described in 5(b)(1)(v)

8 irradiated PULSTAR fuel elements. Only one fuel element is allowed per basket location.

(vi) For the contents described in 5(b)(1)(vi)

8 irradiated square fuel elements or loose plate boxes. Only one fuel element or loose plate box is allowed per basket location (i.e., compartment). Up to 31 loose plates may be placed in each loose plate box.

(vii) *Plutonium Quantity*. The maximum quantity of plutonium in the BRR package is 6,500 Ci (at 4% ^{235}U enrichment of PULSTAR fuel).

Revised Condition No. 6 as follows:

"(a) Each package shall be operated and prepared for shipment in accordance with Chapter 7 of the application, as supplemented

(i) For TRIGA fuel, spacer pedestals shall be used as described in Table 7.1-2 of the application.

(ii) For PULSTAR fuel, spacer pedestals shall be used as described in Table 7.1-1 of the application.

(iii) For square fuel and loose plates, spacer pedestals shall be used as described in Table 7.1-1 of the application.

(iv) When shipping loose plates, use aluminum dunnage sheets to reduce the free space between the flat face of the loose plates and the box opening to a value of ¼ inches or less. The dimensions of the dunnage sheets shall be as shown in Figure 7.1-1 of the application."

Revised the "References" section of the CoC to read as follows:

"AREVA Federal Services LLC application dated May 30, 2016. (Safety Analysis Report, Revision 10)"

The staff also made editorial changes to the CoC to improve its readability.

CONCLUSIONS

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

Issued with CoC No. BRR, Revision No. 5
on July 21, 2016.