CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9341	8	71-9341	USA/9341/B(U)F–96	1	OF	12

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Orano Federal Services LLC
505 S. 336th Street, Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA Federal Services LLC application dated June 13, 2019.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.
(a) Packaging

(1) Model No.: BEA Research Reactor (BRR) Package

(2) Description

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, a closure lid, upper and lower impact limiters, and utilizes American Society for Testing and Materials (ASTM) Type 304 stainless steel as its primary structural material. The package is a right circular cylinder with a dimension of 77.1 inches in length and 38 inches in diameter, not including the impact limiter attachments and the thermal shield. Lead shielding is located between two circular shells, in the lower end structure, and in the shield plug. The payload cavity has a diameter of 16 inches and a length of 54 inches.

Impact Limiters. Impact limiters are attached to each end of the package body. Each impact limiter is 78 inches in diameter and 34.6 inches in length, with a 15-inches long conical section towards the outer end. The impact limiter design consists of ASTM Type 304 stainless steel shells and polyurethane foam with an approximate density of 9 pounds per cubic foot (lb/ft^3).

Fuel Baskets. There are six baskets used with the package, one for each type of fuel transported and one for isotope production targets. 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

5.(a) Packaging – Description (continued)

accommodates two types of fuel assembly: (1) flat type fuels and (2) a 5x5 array of fuel rods enclosed within a rectangular can.

Personnel Barrier. When transporting isotope production targets, a personnel barrier is used to limit access to the package body such that personnel are prevented from touching the cask surface where the surface temperature may exceed the allowable limit for exclusive use shipments. The barrier does not have a radiological purpose.

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.

The package is designed to be transported as one package per conveyance, with its longitudinal axis vertical, by highway truck or by rail in exclusive use. When loaded and prepared for transport, the package is 119.5 inches in length, 78 inches in diameter (over the impact limiters), and weighs 32,000 pounds (lb).

(3) Drawings

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

- 1910–01–01–SAR, “BRR Package Assembly SAR Drawing,” Sheets 1-5, Rev. 8
- 1910–01–02–SAR, “BRR Package Impact Limiter SAR Drawing,” Sheets 1-2, Rev. 1
- 1910–01–03–SAR, “BRR Package Fuel Baskets SAR Drawing,” Sheets 1-4, Rev. 6
- 1910–01–04–SAR, “BRR Package Isotope Target Basket SAR Drawing,” Sheets 1-2, Rev. 1

(b) Contents

(1) Type and form of material

Irradiated MURR Fuel Element. Irradiated University of Missouri Research Reactor (MURR) fuel element to a maximum burnup of 180 megawatt-day (MWD) or a depletion of 30.9% of Uranium-235 (^{235}U). The minimum cooling time is 180 days after reactor shutdown. Each MURR element contains 24 fuel plates. Each fresh MURR fuel element contains 775.0 ± 7.8 g ^{235}U. The enrichment range is 93 ±1 wt.% ^{235}U. The MURR element overall length, including irradiation growth, is 32.75 inches. The maximum decay heat per fuel element is 158 watts (W). The maximum number of fuel elements per basket is 8. The bounding weight of one element is 15 lb. Table 1.1 includes characteristics of a pre-irradiated MURR fuel element.

5.(b)(1) (i) Type and form of material - Irradiated MURR Fuel Element (continued)

Table 1.1. MURR - Key Fuel Element Parameters

Maximum active fuel length (inches)	24.8
Overall length (inches)	32.75
Minimum cladding thickness (inch)	0.008
Nominal fuel matrix thickness (inch)	0.02
Fuel matrix	UAl_x
Cladding material	Aluminum
Maximum ^{235}U per element (g)	782.8
Maximum enrichment (wt.%)	94.0
Maximum ^{235}U per fuel plate (g)	46.0

(ii) *Irradiated MITR–II Fuel Element*. Irradiated Massachusetts Institute of Technology Research Reactor (MITR–II) fuel element to a maximum burnup of 165 MWD or a ^{235}Udepletion of 43.9%. The minimum cooling time is 120 days after reactor shutdown. Each MITR-II element contains 15 fuel plates. Each fresh MITR–II element contains 510.0 +3.0/-10.0 g ^{235}U, which is 500 - 513 g ^{235}U. The enrichment range is 93 ±1 wt.% ^{235}U. The MITR–II element overall length, including irradiation growth, is 26.52 inches. The maximum decay heat per element is 150 W. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 10 lb. Table 1.2 includes the key parameters for a pre-irradiated MITR–II fuel element.

Table 1.2. MITR-II - Key Fuel Element Parameters

Maximum active fuel length (inches)	22.76
Overall length (inches)	26.52
Minimum cladding thickness (inch)	0.008
Nominal fuel matrix thickness (inch)	0.03
Maximum fuel matrix width (inches)	2.171
Fuel matrix	UAl_x
Cladding material	Aluminum
Maximum ^{235}U per element (g)	513
Maximum enrichment (wt.%)	94.0
Maximum ^{235}U per fuel plate (g)	34.3

(iii) *Irradiated ATR Fuel Element.* Irradiated Advanced Test Reactor (ATR) fuel element to a maximum burnup of 480 MWD or a ^{235}U depletion of 58.6%. The minimum cooling time is 1,670 days (4.6 years) after reactor shutdown. Each ATR fuel element contains 19 plates. The YA fuel element has 19 plates, but only 18 contain fuel.

5.(b)(1) (iii) Type and form of material - *Irradiated ATR Fuel Element* (continued)

There are two general classes of ATR fuel element, XA and YA. The enrichment range is 93 ± 1 wt.% ^{235}U. The XA fuel element has a fresh fuel loading of 1,075 ± 10 g ^{235}U. The YA fuel element has a fresh fuel loading of 1,022.4 ± 10 g ^{235}U. A second YA fuel element design (YA–M) has the side plate width reduced by 15 mils. The ATR element overall maximum length, after removal of the end box structures, 51.0 inches. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 25 lb. The maximum decay heat per element is 30 W. Table 1.3 includes characteristics of a pre-irradiated ATR fuel element.

Table 1.3. ATR - Key Fuel Element Parameters

Maximum active fuel length (inches)	48.77
Overall length (inches)	51
Minimum cladding thickness for Plate 1 (inch)	0.018
Minimum cladding thickness for Plates 2-18 (inch)	0.008
Minimum cladding thickness for Plate 19 (inch)	0.018
Nominal fuel matrix thickness (inch)	0.02
Fuel matrix	UAl_x
Cladding material	Aluminum
Maximum ^{235}U per element (g)	1,085
Maximum enrichment (wt.%)	94.0
Maximum ^{235}U per fuel plate (g)	85.2

(iv) *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.

5.(b)(1) (iv) Type and form of material - *Irradiated TRIGA fuel elements* (continued)

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	1.41	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

[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.

5.(b)(1) (iv) Type and form of material - *Irradiated TRIGA fuel elements* (continued)

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. The bounding weight of any TRIGA fuel element is 10 lb. The maximum decay heat per element is 20 W. The number of TRIGA rods per element is 1. 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

[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.

5.(b)(1) Type and form of material (continued)

(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

Note: Maximum length includes 0.25 in. for irradiation growth.

5.(b)(1) Type and form of material (continued)

(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.

5.(b)(1) Type and form of material (continued)

(vii) *Isotope Production Targets*. Targets are irradiated in nuclear reactors to produce Co-60 and may be made of aluminum and contain a large quantity of small pellets, or they may consist of a cylindrical rod of cobalt material inside a stainless steel tube. All targets must be placed into target holders prior to loading into the basket. There are two different payload types:

1. *Payload Type 1*. Type 1 consists primarily of higher-activity targets of a newer design, which may also include lower-activity targets as described under Payload Type 2. The pellets are arranged in several stacks in an annular configuration within the target body. Payload Type 1 consists of up to 10 targets, which must be loaded in the inner row of basket holes, and be arranged using a loading plan into five zones of two holes each. The maximum activity in any zone is 22,000 Ci. A loading collar must be installed to block access to the outer row of holes before loading payload Type 1 targets. Table 1.8 includes the characteristics of payload type 1 of the isotope production targets.

Table 1.8. Characteristics of Isotope Production Targets, Payload Type 1

Parameter	Value
Target Diameter	1/2 inches
Target Length	16 inches
Cladding Material	6061-T6 aluminum alloy
Target Contents	6,000 pellets (approximately)
Pellet Size	1mm diameter × 1mm thick
Maximum Activity	up to 14,100 Ci, Co-60
Payload Quantity	10 targets
Total Activity	up to 82,000 Ci

2. *Payload Type 2*: Type 2 consists of lower-activity targets of an older design, which include:

   A. Design in which an aluminum core rod holds pellets placed in dimples on the outer surface and which are retained by a close-fitting outer sleeve, welded to the core rod on each end and

   B. Design using a solid rod of cobalt inside a stainless steel tube with welded ends.

   Table 1.9 includes the characteristics of payload type 2 of the isotope production targets.

5.(b)(1) Type and form of material (continued)

Table 1.9. Characteristics of Isotope Production Targets, Payload Type 2

Parameter	Value
Target Diameter	5/8 inches (pellet design) 5/16 inches (solid rod design)
Target Length	Up to 16.5 inches
Cladding Material	Aluminum alloy 6061-T6 (pellet design) Stainless steel (solid rod design)
Target Contents	Approximately 5,500 pellets or one solid or segmented rod of cobalt metal
Pellet Size	1 mm diameter x 1 mm thick
Maximum Activity	Up to 4,000 Ci, Co-60
Payload Quantity	20 targets
Total Activity	Up to 80,000 Ci

5.(b)(2) Maximum quantity of material per package

(i) For the contents described in 5(b)(1)(i):

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

(ii) For the contents described in 5(b)(1)(ii):

8 irradiated MITR-II fuel elements. Only one fuel element is allowed per basket location.

(iii) For the contents described in 5(b)(1)(iii):

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

(iv) For the contents described in 5(b)(1)(iv):

19 irradiated TRIGA fuel elements. Only one fuel element is allowed per basket location.

26 types of TRIGA fuel.

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

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

5.(b)(2) Maximum quantity of material per package (continued)

(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).

(viii) For the contents described in 5(b)(1)(vii)(1)

10 target holders. For payload type 1, up to 10 target holders may be placed into the inner row of holes in the isotope basket.

(ix) For the contents described in 5(b)(1)(vii)(2)

20 target holders. For payload Type 2, up to 20 target holders may be placed into any of the 20 holes in the isotope basket.

(c) Criticality Safety Index (CSI): 0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(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.

(v) For isotope production targets, a personnel barrier shall be used as described in Section 7.1.4 of the application.

(b) Each package shall be acceptance tested and maintained in accordance with Chapter 8 of the application.

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

8. Transport by air of fissile material is not authorized.

9. Revision No. 6 of the certificate may be used until January 31, 2020.

10. Expiration date: January 31, 2020.

REFERENCES

AREVA Federal Services LLC application dated June 13, 2019. (Model No. BRR Safety Analysis Report, Revision 15)

Orano Federal Services LLC supplements dated: July 23 and July 30, 2019 (Safety Analysis Report, Revision 16).

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John McKirgan

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

Date: 8/15/19

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. 8

Summary

By application dated June 13, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19176A171), as supplemented on July 23, 2019 (ADAMS Accession No. ML19220A336) and July 30, 2019 (ADAMS Accession No. ML19219A162), Orano Federal Services, LLC (OFS or the applicant), requested that the U.S. Nuclear Regulatory Commission (NRC) revise Certificate of Compliance (CoC) No. 9341 for the Model No. BEA Research Reactor (BRR) package. The applicant requested the amendment due to a small change in the thickness of the inner shell resulting from a machining error and to consolidate its associated safety analysis report (SAR). The application includes one revised drawing, and structural and shielding analyses associated with the drawing change.

The Model No. 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 truck or by rail in exclusive use.

The NRC staff (the staff) reviewed the application, as supplemented, 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." Based on the statements and representations in the application, as supplemented, and the "conditions" section of this safety evaluation report, 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."

EVALUATION

1.0 GENERAL INFORMATION

1.1 Packaging Description

The BRR package consists of a payload basket, a lead–shielded package body, a separate, removable upper shield plug, a closure lid, 12 closure bolts, upper and lower impact limiters containing polyurethane foam, and a personnel barrier used only with the isotope payload.

The BRR package body is a right circular cylinder 77.1 inches long and 38 inches in diameter. It comprises inner and outer shells connected by a thick lower end casting. The shells and lower end casting are made of American Society for Testing and Materials 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-inch-thick shield plug under the closure lid, the package body assembly constitutes the payload cavity, which has a diameter of 16 inches and a length of 54 inches.

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, and
6) a personnel barrier for isotope production targets to limit access to the package body.

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, Rev. 8 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.

1.2 Drawings

OFS revised one drawing associated with the proposed changes.

1910-01-01-SAR, Sheets 1-5, Rev. 8	BRR Package Fuel Assembly

1.3 Evaluation Findings

The staff has reviewed the proposed changes and concludes that they meet the requirements of 10 CFR Part 71.

2.0 STRUCTURAL EVALUATION

The objective of the structural evaluation of the BRR package design is to verify that the design satisfies the requirements of 10 CFR Part 71 and that the structural performance of the package has been adequately evaluated for the conditions specified for normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

2.1 Background

The applicant proposed changing the thickness of the inner shell from a dimension of 1.0 inch minimum, as currently authorized, to a dimension of 1.00 ± 0.06 inches. The reason for the proposed change is due to an error in machining the inner diameter of one of the two packages currently being fabricated. The inner diameter has been partially machined such that part of the inner shell has a thickness less than 1.0 inch. This proposed change (1.00 ± 0.06 inches) allows a minimum thickness of 0.94 inch so that the package with the machining error can meet the dimension specified in the drawing. The applicant performed structural evaluation of the consequences of this change on the free drop analysis.

2.2 Evaluations

2.2.1 Center of Gravity

SAR Table 2.1-2, Rev. 16 shows the new center of gravity (CG) of the package. The new CG is located 38.7 inches from the bottom outside surface of the cask body. It is identical to the CG location previously reviewed and accepted. Since the CG location is not changed, the NRC staff found that the inner shell thickness change of ± 0.06 inch acceptable.

2.2.2 Total Weight

SAR Table 2.1-2, Rev. 16 shows the total weight of the package. The authorized maximum allowable total weight of the package is 32,000 pounds, which was previously reviewed and accepted. The NRC staff found that the weight increase due to the thickness change of ± 0.06 inch is very small and a new actual total weight of the package is still bounded by 32,000 pounds. Therefore, the total weight of the package is acceptable.

2.2.3 Buckling Assessment

The applicant previously performed a buckling analysis using the methodology of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-284-2. The applicant noted that there is no need for additional buckling analysis due to the thickness change because the strength of the inner shell for the buckling assessment was conservatively ignored in the previous buckling analysis. The NRC staff evaluated the applicant's assessment and concludes with reasonable assurance that the buckling performance of the inner shell continues to be acceptable.

2.2.4 Cask Body Stresses

The applicant calculated cask body stresses with a thickness of 0.94 inch for a free drop under NCT and HAC. The applicant used a linear elastic approach to calculate cask body stresses where the stresses were proportionally calculated based on the previously calculated cask body stresses with the inner shell thickness of 1.0 inch.

The calculations in SAR Section 2.6.7.3, Rev. 16 show that the minimum stress margins of safety of the inner shell for the top-down drop under NCT and HAC are +0.33 and +0.69, respectively. The calculations also show that the minimum margins of safety of the package for the side drop under NCT and HAC are +0.06 and +0.23, respectively, indicating that the effect of the inner shell thickness change of 0.06 inch is minor and the package with the inner shell thickness of 0.94 inch is safe. The NRC staff reviewed the calculations and found them acceptable.

2.3 Findings

Based on the statements and representations contained in the application, as supplemented, and the conditions given in the certificate of compliance, the NRC staff concludes that the package has adequately been described and evaluated to demonstrate that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

There were no changes that affected the package's thermal evaluation.

4.0 CONTAINMENT EVALUATION

There were no changes that affected the package's containment evaluation.

5.0 SHIELDING EVALUATION

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

5.1 Background

The applicant states that the currently authorized design of the BRR package shows a dimension of the inner shell of 1.0 inch as shown in Drawing 1910-01-01-SAR, Revision 8, Sheet 4, Zone B-6. The tolerance is ± 0.06 inches. According to the applicant, this proposed change to the inner shell thickness is necessary due to an error in machining the inner diameter of one of the two BRR packages currently being fabricated. The inner diameter has been partially machined such that part of the inner shell has a thickness less of than 1.0 inches. The as built inner shell has a minimum thickness of 0.94 inches (1.00 ± 0.06 inches).

5.2 Evaluation

The staff reviewed the shielding effect of the reduced steel thickness for both gammas and neutrons in the analyses submitted by the applicant in document CALC-3022582-000 (ADAMS Accession No. ML19220A336). The applicant performed shielding calculations using a simplified MCNP shielding model. The most significant gamma and neutron sources (in terms of resulting total dose) were selected by the applicant from Revision 16 of the SAR. The applicant built a spherical model of the BRR side wall with a point source at the center. Surface dose rates at the most limiting location per the SAR, were calculated for the currently-licensed BRR side wall material thickness and compared to surface dose rates using the reduced inner shell wall thickness.

The shielding analysis performed by the applicant shows the effect of reducing the inner shell of 0.94 inches results in a 1% increase in the maximum side dose rates reported for the fuel payloads in SAR Table 5.1-1, Summary of Maximum Total Dose Rates (Exclusive Use) for Irradiated Fuel Payloads, and a 6% increase in the maximum side dose rates reported for isotope production target payloads in SAR Table 5.6-1, Summary of Maximum Total Dose Rates (Exclusive Use) for Isotope Production Target Payloads. Shielding calculations presented by the applicant in document CALC-3022582-000 showed that the dose rates from all currently-licensed BRR payloads will remain less than applicable limits. The applicant applied the maximum gamma and neutron fractional increases (6% and 0.7%, respectively) to the most limiting fuel payload and location dose rates (TRIGA fuel, side surface) which resulted in a maximum dose rate of 68.7 mrem/hr on the side surface for the case with a reduced inner steel thickness of 0.94 inches, which is significantly less than the limit of 200 mrem/hr. Similarly, applying the applicant applied the maximum gamma fractional increase to the most limiting Co-60 payload and location dose rate (side surface) which resulted in a maximum dose rate of 183.3 mrem/hr for the case with a reduced inner steel thickness of 0.94 inches, which is less than the limit of 200 mrem/hr.

The staff reviewed the applicants' simplified model and considered whether the applicant could have neglected important details such as streaming paths. Additionally, the staff considered if the simplified spherical geometry would accurately represent the source term and the geometry of the package in terms of shielding because a detector will receive more radiation from a line source than it does from a point source with the same total strength. The impact in the difference in the modeling is dependent on the geometry of the sources and the distance between the source and the point of interest. As such, the staff determined that the applicant model of the source in the shielding design analyses may not be conservative. However, based on the large margin of the calculated dose rates at the most limiting fuel payload location to the dose rate limits, the staff has reasonable assurance that the regulatory limits will not be exceeded. Thus, the staff found the reduction of inner shell thickness acceptable.

5.3 Findings

Based on the review of the statements and representations in the application request, the staff finds that the BRR transportation package has been adequately described and evaluated and that the package design with the reduction in the inner shell wall thickness still meets the shielding design requirements of 10 CFR 71.47 and 10 CFR 71.51.

6.0 CRITICALITY EVALUATION

There were no changes that affected the package's criticality evaluations.

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. This amendment also incorporates revised operating procedures approved in a letter authorization dated August 5, 2019 (ADAMS Accession No. ML19217A124) stemming from a request for this authorization.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

Chapter 8 of the application identifies the acceptance tests and maintenance programs to be conducted on the Model BRR package and verifies its compliance with the requirements of 10 CFR Part 71.

9.0 CONDITIONS

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

Condition No. 3.(b), "Title And Identification Of Report Or Application," was updated to include reference to the most recent application, which included Revision 16 of the SAR.

Condition No. 5.(a)(3), "Drawings," was updated to reflect one revised drawing.

1910–01–01–SAR, Sheets 1-5, Rev. 8 Assembly

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

AREVA Federal Services LLC application dated June 13, 2019. (Model No. BRR Safety Analysis Report, Revision 15).

Orano Federal Services LLC supplements dated: July 23 and July 30, 2019 (Safety Analysis Report, Revision 16).

10.0 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. In the consolidated SAR, OFS incorporated all supplements previously approved by NRC.

Issued with CoC No. 9341 for the Model No. BRR, Revision No. 8.