NRC FORM 618 (8-2000) 10 CFR 71	**CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES**			U.S. NUCLEAR REGULATORY COMMISSION
a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE / PAGES
9341	2	71-9341	USA/9341/B(U)F–96	1 OF 6

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)*

AREVA FEDERAL SERVICES LLC
505 336th ST Suite 400
Federal Way, WA 98003

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

AREVA Federal Services LLC
application dated March 25, 2009.

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

A package used to transport fuel elements that have been irradiated in various test and research reactors. The package is comprised of a lead–shielded cask body, payload basket, an upper shield plug, a closure lid, upper and lower impact limiters, and utilizes ASTM Type 304 stainless steel as its primary structural material. The cask is a right circular cylinder 77.1 inches long 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 are attached to each end, having essentially identical design. Each limiter is 78 inches in diameter and 34.6 inches long overall, with a conical section 15 inches long towards the outer end. The impact limiter design consists of ASTM Type 304 stainless steel shells and approximately 9 lb/ft^3 polyurethane foam. There are four 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 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 long, 78 inches in diameter (over the impact limiters), and weighs 32,000 lb.

5.(a) Packaging (continued)

(3) Drawings

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

- 1910–01–01–SAR, BRR Package Assembly SAR Drawing, Sheets 1-4, Rev. 4
- 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 1-3, Rev. 4

(b) Contents

(1) Type and form of material

(i) Irradiated MURR fuel element to a maximum burnup of 180 MWD or a U-235 depletion of 30.9%. The minimum cooling time is 180 days after reactor shutdown. Each MURR element contains 24 fuel plates. Each fresh MURR element contains 775.0 ± 7.8 g U–235. The enrichment range is 93 ±1 wt.% U-235. The MURR element overall length, including irradiation growth, is 32.75 inches. The maximum decay heat per fuel element is 158 W. The maximum number of fuel elements per basket is 8. The bounding weight of one element is 15 lb. Pre-irradiated MURR fuel element dimensions are in Table 1.1.

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	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	782.8
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	46.0

(ii) Irradiated MITR–II fuel element to a maximum burnup of 165 MWD or a U–235 depletion 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 U–235, which is 500 - 513 g U-235. The enrichment range is 93 ±1 wt.% U-235. 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. Pre-irradiated MITR–II fuel element dimensions are in Table 1.2.

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

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	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	513
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	34.3

(iii) Irradiated ATR fuel element to a maximum burnup of 480 MWD or a U–235 depletion of 58.6%. The minimum cooling time is 1,670 days (4.6 years) after reactor shutdown. Each ATR fuel element contains 19 plates. The YA fuel element has 19 plates, but only 18 contain fuel. There are two general classes of ATR fuel element, XA and YA. The enrichment range is 93 ±1 wt.% U-235. The XA fuel element has a fresh fuel loading of 1,075 ± 10 g U–235. The YA fuel element has a fresh fuel loading of 1,022.4 ± 10 g U–235. A second YA fuel element design (YA–M) has the side plate width reduced by 15 mils. The ATR element overall length, after removal of the end box structures, 51.0 inches max. 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. Pre-irradiated ATR fuel element dimensions are in Table 1.3.

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	U-Al (x)
Cladding material	Aluminum
Maximum U-235 per element (g)	1,085
Maximum enrichment (wt.%)	94.0
Maximum U-235 per fuel plate (g)	85.2

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

(iv) Irradiated TRIGA fuel elements. Pre-irradiated TRIGA fuel element dimensions are in Table 1.4. The TRIGA fuel matrix is uranium mixed with zirconium hydride. The BRR package is limited to five specific TRIGA fuel types:

1. 8 wt% uranium in the fuel matrix, U - U/Zr with uranium aluminum clad element (General Atomics catalog number 101).
2. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 103).
3. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element, high enriched uranium (General Atomics catalog number 109). This fuel element is sometimes referred to in the literature as a Fuel Life Improvement Program (FLIP) element.
4. 20 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element (General Atomics catalog number 117). This fuel element is sometimes referred to in the literature as a FLIP–LEU–I element.
5. 8.5 wt% uranium in the fuel matrix, U - U/Zr with uranium stainless steel clad element, instrumented (General Atomics catalog number 203).

Table 1.4
TRIGA - Fresh Fuel Element Characteristics

Parameter	GA Cat. # 101	GA Cat. # 103	GA Cat. # 109	GA Cat. # 117	GA Cat. # 203
Maximum Active Fuel Length (in)	14	15	15	15	15
Fuel Pellet OD (in)	1.41	1.44	1.44	1.44	1.44
Overall Element Length (in)	28.37	28.9	28.9	29.68	45.25
Cladding OD (in)	1.48	1.48	1.48	1.48	1.48
Minimum Cladding Thickness (in)	0.0285	0.0185	0.0185	0.0185	0.0185
Graphite Reflector Length Top/Bottom (in)	4.0 / 4.0	2.6 / 3.7	2.6 / 3.7	2.6 / 3.7	3.1 / 3.4
Maximum Zr Mass in Fuel Matrix (g)	2,070	2,088	2,060	2,060	2,088
Maximum U-235 Mass (g) per element	36	39	137	101	39
Maximum U-235 Enrichment (wt%)	20	20	70	20	20
Maximum H/Zr atom ratio	1.0	1.7	1.6	1.6	1.7

The maximum length of a TRIGA fuel element, including irradiation growth, is 45.50 inches. For all fuel elements, 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. TRIGA fuel parameters are in Table 1.5.

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

Table 1.5

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

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.

(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.

(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. Expiration date: January 22, 2015.

REFERENCES

AREVA Federal Services LLC application dated March 25, 2009.

Supplements dated August 6, 2009, November 5, 2009, June 4, 2010, December 16, 2010 and June 24, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: August 22, 2011

Safety Evaluation Report
Model No. BRR Package
Docket No. 71-9341
Certificate of Compliance No. 9341
Revision 2

SUMMARY

By application dated June 24, 2011, AREVA Federal Services, LLC (AREVA) submitted an application to the U.S. Nuclear Regulatory Commission for an amendment to Certificate of Compliance (CoC) No. 9341 for the BEA Research Reactor (BRR) package.

The revision adjusts the assumed minimum cladding thickness in the criticality analysis for irradiated University of Missouri Research Reactor (MURR), Massachusetts Institute of Technology Research Reactor (MITR-II) and Advanced Test Reactor (ATR) fuel types. Previously, the minimum cladding thickness did not envelop the minimum that may be encountered in actual fuel components. TRIGA fuel types remain unaffected. The staff's evaluation of the criticality analysis of the BRR package with the adjusted cladding thickness is discussed in the safety evaluation report.

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

References

AREVA Federal Services LLC, application dated June 24, 2011.

GENERAL INFORMATION

Minimum cladding thicknesses are reduced and maximum local channel spacing at time of fabrication are now provided for MURR, MITR-II, and ATR fuel. For MURR, MITR-II, and the majority of ATR fuel plates, the minimum cladding thickness is 0.008-inch. For two of the ATR fuel plates, the minimum cladding thickness is 0.018-inch. The maximum local channel spacing for MURR and MITR-II fuel is 0.090-inch. For ATR fuel, the spacing is 0.087-inch.

CRITICALITY EVALUATION

Criticality Design Criteria and Features

Staff reviewed the changes and information in the proposed SAR revision and verified that the information is consistent as well as all descriptions, drawings, figures, and tables are sufficiently detailed to support an in-depth staff evaluation.

Packaging and Design Features

The package consists of a payload basket, a lead-shielded cask body, an upper shield plug, a closure lid, and upper and lower impact limiters. The body is cylindrical, 77.1 inches long (cm) and, excluding the impact limiters, 38 inches (cm) in diameter. The end structures are cast, with lead sheets firmly packed into place. The upper and lower end structures are connected by the inner and outer shells, with lead cast-in-place between the two shells.

Since the cask is designed for use in a pool or hot cell environment, a drain port is present in the lower end structure with a vent in the top end structure extending through the closure lid. These are intended for use with a drying system to ensure water is not present during transport. Both the drain and vent are closed with a threaded plug sealed with a butyl rubber washer.

The BRR package provides a single level of leak-tight containment. This consists of two butyl rubber O-ring seals, with the inner seal providing the containment boundary, and the outer being a test seal.

The payload cavity is 16 inches (cm) in diameter with a length of 54 inches (cm). The contents are held within welded stainless steel baskets containing multiple cavities unique to each fuel type carried.

The BRR package maintains criticality control by means of mass limits. There is no component or material whose primary function is neutron absorption.

Codes and Standards

The applicable regulations considered in the review of the criticality safety portion of this application include the fissile material requirements in 10 CFR Part 71, specifically the general requirements for fissile material packages in §71.55, and the standards for arrays of fissile material packages in §71.59. The staff also used the review guidance contained in NUREG-1617.

Summary Table of Criticality Evaluations

Table 1

Normal Conditions of Transport (NCT)				
	MURR	**MITR-II**	**ATR**	**TRIGA**
Case	k_s	k_s	k_s	k_s
Single Unit Maximum	0.085	0.058	0.088	0.417
Infinite Array Maximum	0.197	0.144	0.234	0.539
Hypothetical Accident Conditions (HAC)				
	MURR	**MITR-II**	**ATR**	**TRIGA**
Case	k_s	k_s	k_s	k_s
Single Unit Maximum	0.784	0.574	0.704	0.709
Infinite Array Maximum	0.827	0.609	0.721	0.720
USL = 0.9209				

Criticality Safety Index (CSI)

The applicant demonstrated that an infinite array of Areva-BRR packages with the most reactive fuel in both NCT and HAC remains adequately subcritical. Therefore, the CSI is 0.0 in accordance with 71.59(b).

Spent Nuclear Fuel Contents

The applicant evaluated the fuel types for which the cladding thickness changes applied. The change only applies to plate-type fuel and not the previously reviewed TRIGA fuel types.

The wedge-shaped MURR fuel elements contain 24 fuel elements, with element 1 having the smallest radius. The fuel meat is uranium aluminide (UAl), while the cladding and structural materials are an aluminum alloy. The side plates are fabricated of ASTM B 209, aluminum alloy 6061-T6 or 6061-T651. Overall length, including irradiation growth is 32.75 in; the bounding weight of a single assembly is 15 lbs.

Fresh MURR elements contain up to 782.8 g U-235 with an enrichment of 93 wt%. The isotope weight percents are shown in Table 1. The MURR fuel element may be irradiated for a maximum of 218,196 MWD/MTU with a minimum of 180 days cooling. The maximum allowed decay heat is 158 W per assembly.

MITR-II fuel is box-shaped and consists of 15 flat fuel plates. The fuel meat is UAl, while the cladding and structural materials are an aluminum alloy. The side plates are fabricated of ASTM B 209 or aluminum alloy 6061-T6. Overall length including irradiation growth is 26.52 in; the bounding weight of one assembly is 10 lbs.

Fresh MITR-II elements contain up to 513.0 g U-235 with an enrichment of 93 wt%. The isotope weight percents are presented in Table 1. The MITR-II element may be irradiated for a maximum of 309,900 MWD/MTU with a minimum cooling time of 120 days. Maximum decay heat allowed is 150 W per assembly.

ATR fuel elements are wedge-shaped and contain up to 19 curved fuel plates with element 1 having the smallest radius. The fissile material is UAl and the clad is an aluminum alloy. The side plates are fabricated of ASTM B 209, aluminum alloy 6061-T5 or 6061-T651. Overall length of the assembly after removal of the end box structures is 51.0 in. The bounding weight for one assembly is 25 lb.

ATR fuel elements fall into two general classes, XA and YA. The XA fuel element is further subdivided into types 7F, 7NB, and 7NBH. The fuel element with the greatest reactivity is the 7NB type which contains no burnable poison. Fuel type 7NBH is similar to the 7NB except that it contains one or two borated plates. The 7F type fuel has burnable poison in the 4 inner- and outer-most plates. The XA type fuel elements have up to 1085 g U-235 with an enrichment of 93 wt%. YA type fuel elements are identical to the 7F elements except that plate 19 is an aluminum alloy plate containing neither fuel nor burnable poison. The YA elements have up to 1032.4 g of U-235. ATR fuel elements may be irradiated up to 491,155 MWD/MTU with a minimum cooling time of 1670 days. The maximum decay heat allowed is 30 W per assembly.

The staff viewed the information in Section 1 and 6 of the SAR and verified that the description of the fuel used in the criticality analysis bounds that of the allowable fuel contents.

General Considerations for Criticality Evaluations

Model Configuration

The BRR cask is modeled using conservative simplifying assumptions. The impact limiters are not included in the model, and the cask is fully reflected with 12 in. (30.48 cm) of water. The omission of the impact limiters results in a minimized separation of casks when modeling an infinite array, increasing the system reactivity. The cask body is modeled as nested steel-lead-steel cylinders. Minor cask details having negligible effect on reactivity are also omitted.

The cask upper region in the model is simply representative of the shield plug thickness, and the steel lid is not included. This omission also reduces separation in the cask array models.

Each fuel type has its own unique basket design. With the basket models, structure details that capture relevant criticality near the active fuel region are included.

All structures are modeled without damage or deformity as prior analysis has shown that the structures remain elastic and the fuel undamaged during HAC.

Material Properties

The fuel, clad, and structural materials are presented. Aluminum structures that are part of the fuel assembly are modeled as pure aluminum with a density of 2.7 g/cm^3. The inner and outer tubes of the package are constructed from stainless steel. The SCALE standard composition for stainless steel 304 was used in the criticality analysis.

Water density is varied from 0.0 to 1.0 g/cm^3 to determine optimum reflection and moderation.

Computer Codes and Cross Section Libraries

The applicant used MCNP5 v1.30 for the criticality analysis. The uranium isotopes use the ENDF/B-VII cross section data. ENDF/B-VI cross section data is used for all other isotopes unless it is unavailable, in which case the ENDF/B-V cross sections are used. The applicant applied the appropriate S(a,b) data (LWRT.60T) for the ambient temperature of the system. Sample input decks are provided by the applicant.

Demonstration of Maximum Reactivity

The fuel assemblies impacted by the scope of this amendment are bounded by the TRIGA fuel during NCT due to its hydrogenous fuel matrix.

During HAC, the package is assumed to be flooded and fully water reflected. The applicant evaluated the displacement of fuel elements and a shifting of assemblies. Moving assemblies toward the center of the package increased reactivity. Water density was varied to determine optimum moderation and reflection. Under optimum flooded conditions, the MURR fuel is the most reactive.

A parametric study of the fuel assemblies is included to determine the impact of various tolerances on reactivity. In the most reactive fuel type (MURR), reactivity is maximized by maximizing the arc length of the fuel meat and the channel spacing. The channel spacing is increased by modeling an artificially reduced clad thickness.

Confirmatory Analyses

Staff developed a model using the KENO-VI module in SCALE 6. The model consists of MURR fuel assembly plates and aluminum structure with the clad thickness minimized to increase moderation to the fullest extent.

Staff analysis used multigroup ENDF/B-VII cross section data. The KENO-VI model maximized the uranium enrichment and mass within the range of uncertainty provided by the applicant. The configuration used in the analysis was determined from the limiting case presented by the applicant in the single package evaluation. Both a single package and an infinite hexagonal array were examined, and staff results were within reasonable agreement with those from the applicant.

Single Package Evaluation

Configuration

The fuel assemblies are shifted radially inward, outward, and upward in conservative manners. The configuration of the model is often impossible to achieve in actual practice due to structural materials that are not included in the analysis. This yields a more conservative result.

Results

In all cases, the most reactive system k_{eff} under both NCT and HAC remains below the stated USL.

Evaluation of Package Arrays under Normal Conditions of Transport (NCT)

Configuration

In the array configurations, the most reactive NCT single package model for each fuel type was utilized. A hexagonal reflective surface is added to simulate a close-packed infinite hexagonal array of packages. There is a reflective boundary on the top and bottom surfaces as well.

Water density between the packages is varied from 0.25 to 1.0 g/cm^3. In each case, reactivity is maximized with no water between the packages.

Results

Table 1 summarizes the most reactive case for each fuel type in an infinite array of packages under NCT.

Evaluation of Package Arrays under Hypothetical Accident Conditions (HAC)

Configuration

An infinite hexagonal array is modeled in the same manner as the NCT array. The most reactive single package configuration under HAC is used in the array. The changes to this amendment under HAC include flooded cases with the cladding thickness changed as appropriate in the model. Water density inside the package is varied, as is the water externally, to determine the maximum reactivity of the system.

Results

Table 1 summarizes the most reactive case for each fuel type in an infinite array of packages under HAC.

Benchmark Evaluations

The applicant used MCNP5 v1.30 for benchmark analysis. The USLSTATS code developed at Oak Ridge National Laboratory (ORNL) was used to establish a USL for the analysis. Both the methodology and software are appropriate for determining criticality safety limits.

Experiments and Applicability

The MURR, MITR-II, and ATR fuel types analyzed in this amendment all fall under the category of high-enriched aluminum plate fuel. The critical experiment benchmarks are selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments.

Thirty five benchmarks meet the criteria of high-enriched uranium plate fuel in a thermal system. One benchmark is derived from the ATR itself, so the benchmark is essentially the same as the ATR fuel modeled in the package analysis. Of those 35, 17 meet the criteria of having a UAl and aluminum fuel matrix, aluminum cladding and no absorbers and are directly applicable. The other 18 are applicable to a lesser degree. The USL selected is the minimum of both benchmark sets.

Bias Determination

The calculated upper subcritical limit for fresh fuel evaluations includes both the bias and uncertainty in the bias, with the most limiting of these presented in Table 1.

Burnup Credit

No credit is taken for fuel reactivity reduction due to burnup. All analyses are modeled using fresh, un-irradiated fuel.

Evaluation Findings

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

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

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

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

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

The staff has reviewed the criticality evaluation of an infinite array of the most reactive configuration in under both NCT and HAC and concludes that it is subcritical under these conditions.

The staff has reviewed the benchmark evaluation of the calculations and concludes that they are sufficient to determine an appropriate bias and uncertainty for the criticality evaluation.

CONDITIONS

- Condition 5.(b) was revised to reduce the minimum cladding thicknesses for MURR, MITR-II, and ATR fuel. For MURR, MITR-II, and the majority of ATR fuel plates, the minimum cladding thickness is now 0.008-inch. For two of the ATR fuel plates, the minimum cladding thickness is now 0.018-inch.

CONCLUSION

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

Issued with Certificate of Compliance No. 9341, Revision No. 2,
on August 22, 2011.