SAFETY EVALUATION REPORT

DOCKET NO. 71-9341

CERTIFICATE OF COMPLIANCE NO. 9341, REVISION NO. 11

MODEL NO. BEA RESEARCH REACTOR PACKAGE

PACKAGE IDENTIFICATION NO. USA/9341/B(U)F-96

SUMMARY

By letter dated August 26, 2024 (Agencywide Documents Access and Management System Accession No. ML24239A866), Orano Federal Services LLC (Orano FS or the applicant) submitted an application to renew and amend Certificate of Compliance (CoC) No. 9341 for the Model No. Battelle Energy Alliance (BEA) Research Reactor (BRR) package to the U.S. Nuclear Regulatory Commission (NRC). Specifically, Orano FS requested to add the following as authorized contents: 1) the low enriched uranium (LEU) versions of the currently authorized high enriched uranium (HEU) research reactor type Advanced Test Reactor (ATR), Massachusetts Institute of Technology Research Reactor (MITR-II), and Missouri Research Reactor (MURR) fuels; and 2) encapsulated and segmented pressurized water reactor (PWR) and boiling water reactor (BWR) fuel rod segments within a Rods-In-Tubes Canister (RITC) basket, for the Model No. BRR Package. The applicant's August 26, 2024, submittal included BEA Research Reactor package Safety Analysis Report (SAR), Revision 19, which provided relevant details for the proposed revisions. By letter dated February 13, 2025 (ML25044A485), the applicant provided an updated SAR, BEA Package SAR, Revision 20, in response to the NRC staff's request for additional information (ML25010A389).

The NRC staff reviewed the application, as supplemented, using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," August 2020 (ML20234A651). The NRC staff reviewed the performance of the package under normal conditions of transport (NCT) as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 71.71 and the performance of the package under hypothetical accident conditions (HAC) as required by 10 CFR 71.73. The NRC staff finds that the analyses performed by the applicant demonstrate that the package provides adequate structural and thermal protection and meets the containment, shielding, and criticality requirements after being subject to the tests for normal conditions of transport and hypothetical accident conditions.

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 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." In addition, the NRC staff noted that the applicant provided a timely renewal request for CoC No. 9341 regarding the Model No. BRR Package. The NRC staff's analyses are included in the sections below.

STAFF EVALUATION

1.0 GENERAL INFORMATION

The BRR Package 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 applicant submitted the amendment request to add to the list of approved contents with two general groups of payloads for the BRR package: 1) LEU versions of currently authorized HEU research reactor type ATR, MITR-II, and MURR fuels; and 2) encapsulated and segmented PWR and BWR fuel rod segments within a new basket known as the RITC basket.

1.1 Packaging Description

The BRR package consists of a payload basket or canister, 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 (in.) long and 38 in. 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-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 or canisters 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.

There is a new SAR drawing, drawing 1910-01-05- SAR, which adds the RITC basket and other necessary hardware components.

1.2 Contents

The applicant is requesting to add two new content types: (1) research reactor type ATR, MITR-II, and MURR Low Enriched Uranium versions of currently authorized HEU versions of the same fuels; and (2) encapsulated and segmented PWR and BWR fuel rod segments within a new basket known as the RITC basket.

1.3 Drawings

The applicant is adding one new SAR drawing, Drawing No. 1910-01-05-SAR, which includes the RITC Basket and other necessary hardware components via August 26, 2024, application request, as supplemented.

1.4 Evaluation Findings

The NRC staff reviewed proposed amendment and renewal application related to CoC No. 9341 for the Model No. BRR Package, and found the following:

F1-1 The application describes the package in sufficient detail to provide an adequate basis for its evaluation.

F1-2 Drawings contain information that provides an adequate basis for evaluation against 10 CFR Part 71 requirements. Each drawing is identified, consistent with the text of the application, and contains keys or annotations to explain and clarify information on the drawing.

F1-3 The application for package approval includes either a description of the quality assurance program or a reference to the applicant's approved quality assurance program.

F1-4 The application for package approval identifies applicable codes and standards for the package design, fabrication, assembly, testing, maintenance, and use.

F1-5 Drawings submitted with the application provide a detailed packaging description that can be evaluated for compliance with 10 CFR Part 71 for each of the technical disciplines.

F1-6 The application specifies any restrictions on the use of the package.

F1-7 The description of the contents meets the requirements in 10 CFR 71.63.

Based on the review, the NRC staff concludes that the modified transport package continues to comply with the regulations in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

2.0 STRUCTURAL EVALUATION

Orano FS submitted an application to amend the NRC CoC No. 9341, "Model No. BEA Research Reactor Package, USA/9341/B(U)F-96, Revision 10," with a proposed BRR SAR, Revision 19 (Reference 1). In SAR, Revision 19, the applicant proposed two changes to the previously approved package. The NRC staff reviewed the proposed changes and found that one of two proposed changes described below is relevant to the structural performance of the package system and required structural evaluations. The change reviewed by the NRC staff involved the addition of a new basket system, known as the RITC basket system, in the BRR package.

This section of the SER provides NRC staff's evaluation of the structural adequacy and integrity of the RITC basket system. The structural design features are reviewed together with the evaluations of the structural analyses performed by the applicant to demonstrate the structural safety performance of the RITC basket system under the NCT and HAC (Reference 2).

2.1 Description of the Rods-In-Tube Canister Basket System

The RITC basket system has three major components: (1) Rods-In-Tube (RIT), (2) RITC, and (3) RITC basket. The details of each component are provided in the drawings in section 1.3.3, "Packaging General Arrangement Drawings," of the SAR.

<u>RIT</u>: The RIT is a PWR or BWR commercial fuel rod segmented and placed within an encapsulated tube. As shown in figure 1.2-24 of the SAR, the RIT is an either 3/8-in. or ½-in. stainless steel, Schedule 40 pipe with either internal or external threaded ends enclosed using pipe-end fittings. The RITs may range in length from a minimum of 10.1 in. to a maximum of 51.1 in.

<u>RITC</u>: The RITC is placed in the smaller 3.5-in. OD tubes of the RITC basket, and is used as holder for the RITs. The RITC is constructed of aluminum 6061 alloy in the T6 or T651 temper. The height of the RITC may vary to accommodate the RITs being transported. As shown in figure 1.2-23 of the SAR, each RITC consists of three carrier tubes for the RIT, arranged concentrically 120° apart. The RITC is nominally 3.2-in. in diameter and varies in height (12-in. minimum height to a maximum 53-in. height). The height depends on the required RIT length as shown in figure 1.2-24 of the SAR. Each of the three RITC carrier tubes is 1.13-in. OD by 1/8-in. thick. Lateral stiffness for the RITC is provided by intermittent tube support plates with the bottom of each RITC consisting of a thicker base plate. Each RITC has a centrally located lifting pintle welded to the uppermost tube support plate which can be used for lifting the RITC directly out of the RITC basket.

<u>RITC Basket</u>: The applicant described the RITC basket in section 1.2.1.4.8, "Rods-in-Tubes Canister (RITC) and RITC Basket," of the SAR. The RITC basket is designed to carry irradiated LEU fuel rod segments within encapsulated tubes. The overall dimensions of the RITC basket are approximately 15.5-in. diameter and 53.6-in. height. The RITC basket is constructed of aluminum 6061 alloy in the T6 or T651 temper with three flat circular plates which support four tubes that span the height of the basket. Among the four tubes, the two larger tubes are fabricated from nominal 6.0-in. schedule 10 pipe, while the two smaller tubes are constructed from 3.5-in. outside diameter (OD), 1/8-in. thick tube stock.

Those two larger 6.0-in. basket pipes are empty and function only to provide structural rigidity to the entire basket. Additionally, six 1/4-in.-thick vertical support ribs are equally spaced around the perimeter of the RITC basket and provide additional axial stiffness to the basket. The purpose of the RITC basket is to position the RITC in the smaller 3.5-in. tubular openings. Figure 1.2-22 of the SAR shows a configuration of the RITC basket.

The NRC staff reviewed the structural design descriptions of the RITC basket structural system in the SAR and determined that the descriptions provided sufficient information with the bases for structural evaluations. Therefore, the staff concludes that the descriptions of the basket structural system satisfy the requirements of 10 CFR 71.33.

2.2 NCT

The applicant evaluated the RIT, RITC and RITC basket for the NCT free drops as required by 10 CFR 71.71(c)(7). The applicant provided the evaluations in section 2.6, “Normal Conditions of Transport,” and appendix section 2.12.8.9, “RITC Basket, RITC, and RIT,” of the SAR to demonstrate their performance under NCT.

2.2.1 Evaluations of the RIT

The applicant evaluated two RITs (A3 and A4 RITs) using the information provided in Drawing 1910-01-05 of the SAR. Both RITs were analyzed for buckling failure due to axial loading during the end drop. With respect to the side drop, the A3 RIT was only evaluated for the side drop. The A4 RIT was not analyzed for the side drop because the A4 RIT is fully supported by the RITC over its full length as discussed further below.

<u>End Drop</u>: The applicant calculated the axial stress over the section area of the A3 RIT by hand calculation using the NCT acceleration of 40 g. The calculated axial stress was 1,707 psi. Additionally, the applicant calculated the effective radius of gyration and actual slenderness ratio of the A3 RIT and then calculated the critical buckling stress of 9,434 psi using the Euler beam column formula. Considering the factor of safety (FS) of 2.0 corresponding to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), NCT Level A service condition, the allowable stress of the A3 RIT was calculated as 4,717 psi. Based on these two calculated and allowable stresses, the applicant calculated a margin of safety (MS) of +1.76, where the MS is defined as the FS minus one (MS = FS -1.0) indicating that a structural component has an adequate strength and is safe against the applied load if the value of MS is greater than zero. Based on the results of the structural analyses, the applicant concluded that there is no failure of the A3 RIT under the NCT end drop.

The applicant also calculated the axial stress, allowable stress and MS of the A4 RIT for the end drop by the same methodology used for the A3 RIT. The calculated axial stress, allowable stress and MS were 1,587 psi, 7,464 psi and +3.70, respectively. Based on the results of these structural analyses, the applicant concluded that there is no failure of the A4 RIT under the NCT end drop.

The NRC staff reviewed the applicant’s calculations and results of the structural analyses of the A3 and A4 RITs under the NCT end drop and found them acceptable because it demonstrated that the RIT structural components (A3 and A4 RITs) have adequate strength and safety margin under the NCT end drop.

<u>Side Drop</u>: The applicant analyzed the A3 RIT by hand calculation using the closed form solutions with the NCT acceleration of 40 g. The applicant modeled the A3 RIT as a beam structure and analyzed it with three simple supports at the center and both ends. The applicant applied uniformly distributed load on the RIT and calculated bending moments in the RIT using the beam formulas provided in Reference 4. The calculated maximum bending stress and shear stress were 530.5 psi and 13.4 psi, respectively. By applying the NCT acceleration of 40 g, the maximum bending stress of the RIT became 21,220 psi. The applicant obtained the maximum allowable primary membrane and bending stress intensity of 30,000 psi from table 2.12.8-1 of

the SAR. Based on these two calculated and allowable stresses, the applicant calculated a MS of +0.41 and concluded that there is no failure of the A3 RIT under the NCT side drop.

Regarding the structural evaluation for the side drop of the A4 RIT, the applicant did not analyze the A4 RIT because the A4 RIT is fully supported by the RITC over its full length. As a result, the bending and shear stresses due to the side drop are negligible and there is no failure of the A4 RIT under the NCT side drop.

The NRC staff reviewed the applicant's calculations and results of the structural analyses of the A3 and A4 RITs under the NCT side drop and found them acceptable because it demonstrated that the RIT structural components (A3 and A4 RITs) have adequate strength and safety margin under the NCT side drop.

Center of Gravity-Over Corner Drop and Oblique Angle Drop: The applicant performed the structural analyses of the RIT with two free drop orientations (end drop and side drop) under NCT. However, the applicant did not perform the structural analyses with other free drop orientations (i.e., center of gravity (CG)-over corner drop and oblique angle drops) for the RIT under NCT. As required by 10 CFR 71.71(c)(7), a package needs to be demonstrated for structural adequacy by a free drop through a distance specified in 10 CFR 71.71(c)(7) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected. Thus, the NRC staff issued a request for additional information (RAI) to the applicant to provide technical justifications for not performing the structural analyses of the RIT with other drop orientations of oblique angles (30°, 45°and 60°) and CG-over corner to find the location and magnitude of the maximum damage expected in the BRR package (Reference 6). The applicant submitted responses to the RAI with a revised SAR, Revision 20 (Reference 7). The applicant stated that both CG-over corner drop and oblique angle drops were previously considered in the overall performance of the package during the earlier certification tests in order to maximize potential damage as discussed in Appendix section 2.12.3 of the SAR, "Certification Test Plan."

Additionally, the applicant stated that evaluations of the RIT, RITC and RITC basket at the limiting accelerations were considered and determined based on the previous evaluations for other fuel baskets, the results of the evaluations for both end and side drop orientations bound the evaluations of possible structural failures at oblique drop angle orientations, including the CG-over-corner impact under both NCT and HAC. Furthermore, the applicant stated that the revised SAR, Revision 20, which was part of the RAI responses, contains these justifications of how other drop orientations are bounded by the end and side drop cases.

The NRC staff reviewed the RAI responses and revised SAR, Revision 20, and determined that the applicant's technical justifications for not performing the structural analyses of the RIT, RITC and RITC basket with oblique angle drop orientations including the CG-over corner are acceptable because the NRC staff verified that: (i) the results of the evaluations for both end and side drop orientations bound the evaluations of possible structural failures at oblique drop angle orientations including the CG-over-corner impact, and (ii) the technical justifications has been properly incorporated in the revised SAR, Revision 20.

2.2.2 Evaluations of the RITC

The applicant analyzed the load placed in the carrier tubes due to the 40 gravity (g) NCT load for the end and side drops.

End Drop: The applicant selected A4 RIT for the end drop analysis because it has a smaller end diameter and a greater weight and bounds the stresses of the A3 RIT.

The applicant calculated the maximum shear stress of 606.1 psi in the plate by hand calculation with the applied 40 g NCT load. The calculated maximum shear stress was compared with the allowable shear stress of 4080 psi, which is from table 2.1-4 of the SAR. Using these two stresses, an MS of +5.73 was calculated. Based on the result of the analysis, the applicant concluded that there is no structural failure of the RITC under the NCT end drop.

The NRC staff reviewed the applicant's calculations and results of the structural analyses of the RITC under the NCT end drop condition and found them acceptable because it demonstrated that the RITC structural component has adequate strength and safety margin under the NCT end drop.

Side Drop: The applicant analyzed the RITC by hand calculation using the similar methodology used for the side drop analysis of the RIT. The applicant calculated a maximum bending stress of 1,273 psi after applying the NCT acceleration of 40 g. The applicant obtained the maximum allowable primary membrane and bending stress intensity of 10,200 psi from table 2.1-4 of the SAR. Based on these two calculated and allowable stresses, the applicant calculated a MS of +7.01 and concluded that there is no failure of the A4 RIT under the NCT side drop.

The NRC staff reviewed the applicant's calculations and results of the structural analysis of the RITC under the NCT side drop and found them acceptable because it demonstrated that the RITC structural component has adequate strength and safety margin under the NCT side drop.

2.2.3 Evaluations of the RITC Basket

The applicant analyzed the RITC basket structure with the 3.5-in. OD tube for the end and side drops under the 40 g NCT load.

End Drop: The 3.5-in. OD basket tubes were loaded in compression during an NCT end drop. The applicant calculated the axial stress over the section area of the basket tube by hand calculation using the NCT acceleration of 40 g. The calculated axial stress was 528.3 psi.

The applicant also calculated the allowable buckling stress of 11,005 psi using the ASME BPVC Code Case N-284-4 (Reference 5). The FS of 2.0 corresponding to the ASME BPVC, NCT Level A service condition was considered in the calculation. Based on these two calculated and allowable stresses, the applicant concluded that there is no failure of the RITC basket under the NCT end drop because the basket has significant strength (11,005 psi) to resist the applied load (528.3 psi).

The NRC staff reviewed the applicant's calculations and results of the structural analysis of the RITC basket under the NCT end drop condition and found them acceptable because it demonstrated that the RITC basket has adequate strength and safety margin under the NCT end drop.

Side Drop: The applicant analyzed the RITC basket by hand calculation using the similar methodology used for the side drop analyses of the RIT and RITC. The applicant calculated maximum bending stress of 2,564 psi in the 3.5-in. OD basket tube and compared this stress to the allowable stress of 10,200 psi, which is from table 2.1-4 of the SAR. Based on these two stresses, a MS of +2.98 was calculated. Based on the results of the analysis, the applicant concluded that there is no failure of the RITC basket under the NCT side drop.

The NRC staff reviewed the applicant's calculations and results of the structural analysis of the RITC basket under the NCT side drop condition and found them acceptable because it demonstrated that the RITC basket has adequate strength and safety margin under the NCT side drop.

2.2.4 Conclusion

The NRC staff reviewed the applicant's evaluations for the RIT, RITC and RITC basket under the NCT and concludes that the proposed addition of the new RITC basket system to the BRR package satisfies the regulatory requirements of 10 CFR 71.71(c)(7).

2.3 HAC

The applicant evaluated the RIT for the HAC free drops as required by 10 CFR 71.73(c)(1). It provided the evaluations in section 2.7, "Hypothetical Accident Conditions," and Appendix section 2.12.8.9, "RITC Basket, RITC, and RIT," of the SAR to demonstrate its performance under HAC. However, the applicant did not evaluate the RITC and RITC basket under HAC because the RITC and RITC basket are not required to control the separation of the fuel rod segments during the HAC event and, as a result, the RITC and RITC basket are not classified as important to safety components (ITS) during the HAC events in the shielding analysis, as discussed in section 5.10.4, "Shielding Evaluation," of the SAR. Therefore, only the RIT was evaluated with the HAC free drop event. Chapter 5, "SHIELDING EVALUATION," of this SER provides the NRC staff's evaluations on shielding.

2.3.1 Evaluations of the RIT

The applicant analyzed two RITs (A3 and A4 RITs) for a buckling failure due to axial loading for the end drop under HAC. With respect to the side drop, the A3 RIT was only evaluated for the side drop. The A4 RIT was not analyzed for the side drop because the A4 RIT is fully supported by the RITC over its full length.

<u>End Drop</u>: The applicant calculated the axial stress over the section area of the A3 RIT by hand calculation using the HAC acceleration of 120 g. The calculated axial stress was 5,122 psi. Additionally, the applicant calculated the effective radius of gyration and actual slenderness ratio of the A3 RIT and then calculated the critical buckling stress of 9,434 psi using the Euler beam column formula. Considering the FS of 1.34 corresponding to the ASME BPVC, HAC Level D service condition, the allowable stress of the A3 RIT was calculated as 7,040 psi. Based on these two calculated axial and allowable stresses, the applicant calculated the MS of +0.37 and concluded that there is no failure of the A3 RIT under the HAC end drop.

The applicant also calculated the axial stress, allowable stress and MS of the A4 RIT for the end drop using the same methodology used for the A3 RIT. The calculated axial stress, allowable stress and MS were 4,762 psi, 11,140 psi, and +1.34, respectively. Based on the results of the structural analyses, the applicant concluded that there is no failure of the A4 RIT under the HAC end drop.

The NRC staff reviewed the applicant's calculations and results of the structural analyses of the A3 and A4 RITs under the HAC end drop and found them acceptable because it demonstrated an adequate safety margin in the RITs.

Side Drop: The applicant analyzed the A3 RIT by hand calculation using the similar methodology used for the side drop analyses under NCT. The applicant calculated maximum bending stress of 63,660 psi after applying the HAC acceleration of 120 g. The applicant obtained the maximum allowable primary membrane and bending stress intensity of 72,000 psi from table 2.1-4 of the SAR. Based on these two calculated and allowable stresses, the applicant calculated a MS of +0.13 and concluded that there is no failure of the A3 RIT under the NCT side drop.

Regarding the structural evaluation for the side drop of the A4 RIT, the applicant did not analyze the A4 RIT because the A4 RIT is fully supported by the RITC over its full length. As a result, the bending and shear stresses due to a side drop are negligible and there is no failure of the A4 RIT under the HAC side drop.

The NRC staff reviewed the applicant's calculations and results of the structural analyses of the A3 and A4 RITs under the HAC side drop condition and found them acceptable because it demonstrated adequate safety margin in the RITs.

2.3.2 Conclusion

The NRC staff reviewed the applicant's evaluations for the RIT under HAC and concludes that the proposed addition of the new RITC basket system to the BRR package satisfies the regulatory requirements of 10 CFR 71.73(c)(1).

2.4 Evaluation Finding

The NRC staff reviewed and evaluated the applicant's statements and representations in the application. Based on the review and evaluations, the NRC staff concludes that the RITC basket structural system is adequately described, analyzed, and evaluated to demonstrate that its structural capability and integrity meet the regulatory requirements of 10 CFR Part 71.

2.5 References

1. Orano Federal Services LLC Letter to U.S. Nuclear Regulatory Commission, "Submittal of BEA Research Reactor Package CoC Amendment and CoC Renewal Request, Docket No. 71-9341, EPID L-2023-LLA-0090," dated August 26, 2024 (ML24239A866).

2. 10 CFR 71, "Packaging and Transportation of Radioactive Material."

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 7.6, Revision 1, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," 1978.

4. The Aluminum Association Inc., "Aluminum Design Manual 2015," Arlington, VA, 2015.

5. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1, Class MC, Code Case N-284-4, Metal Containment Shell Buckling Design Methods, 2012 Edition.

6. U.S. Nuclear Regulatory Commission Letter to Orano Federal Services LLC, “Amendment Request for Certificate of Compliance No. 9341 for the Model No. BEA Research Reactor Package – Request for Additional Information,” dated January 15, 2025 (ML25010A388).

7. Orano Federal Services LLC Letter to U.S. Nuclear Regulatory Commission, “Submittal of BEA Research Reactor Package CoC Amendment RAI Responses, Docket No. 71-9341, EPID L-2023-LLA-0090,” dated February 13, 2025 (ML25044A486).

3.0 THERMAL EVALUATION

The objective of the staff evaluation here is to ensure that (1) the packaging components of the BRR package remain within their limits under loading operations (e.g., vacuum drying), NCT and HAC; (2) the maximum accessible surface temperature of the package meets the requirements of 10 CFR 71.43(g); and (3) the maximum pressures of the package remain below the maximum normal operating pressure (MNOP) under NCT and the design pressure under HAC, in accordance with acceptance criteria of the NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material: Final Report,” and in compliance with thermal requirements of 10 CFR Part 71 under NCT and HAC.

For this renewal and amendment application, the applicant included thermal analyses for the RITC basket in SAR section 3.7, “Thermal Evaluation for RITC Payload,” including analyses of the HEU configurations of the MURR fuel, the MITR-II fuel, and the ATR fuel. The amendment application also performed thermal analyses for the LEU fuel configuration. The analysis for the LEU fuels is described in SAR section 3.8, “Thermal Evaluation of LEU Fuels.”

3.1 Thermal Design

The BRR package, as described in SAR chapter 1.0, consists of a lead-shielded cask body, a separate/removable upper shield plug, a bolted closure lid, closure bolts, upper and lower impact limiters containing polyurethane foam, a payload basket specific to the type of payload being transported, and an optional personnel barrier which is only included with the isotope target payloads.

3.1.1 Heat Dissipation

The applicant stated, in SAR section 1.2.1.8, “Heat Dissipation,” that the dissipation of heat from the BRR package is entirely passive: (a) the impact limiters are painted white to reduce the absorption of solar heat, (b) a thermal shield is attached to the outside of the outer shell to form a thin air gap that inhibits heat transfer into the package and is used on the package body to limit the temperature of the lead gamma shield in the HAC fire event, and (c) a personnel barrier is used when transporting isotope production targets.

The staff reviewed the decay heat loadings of all fuels provided in SAR section 3.1.2 and confirmed that heat loads of both MURR HEU fuel and MITR-II HEU fuel exceed heat loads of other types of fuels, and therefore the peak component temperatures achieved by the packaging components for transporting other types of fuels will be bounded by those predicted for either the MURR HEU fuel or MITR-II HEU fuel.

3.1.2 Minimum Temperatures

The applicant stated in SAR section 3.3.1.2, “Minimum Temperatures,” that all package components will eventually achieve the -40°F temperature under steady-state cold conditions with zero decay heat load and an ambient air temperature of -40°F per 10 CFR 71.71(c)(2).

The staff referred to SAR section 3.2.2, “Technical Specifications of Components,” for packaging components of the BRR package used to transport HEU fuel, RITC payload, or LEU fuel, and confirmed that a temperature of -40°F is within the allowable operating temperature range for all package components, including the containment seals which are sustainable down to -75°F.

3.2 Thermal Evaluation of HEU Fuels

3.2.1 Contents Decay Heat

The applicant provided, in SAR section 3.1.2, “Content’s Decay Heat,” the package design basis decay heat loads of 1,264 watts (W) per basket for MURR HEU fuel, 1,200 W per basket for MITR-II HEU fuel, 240 W per basket for ATR HEU fuel, and 40 W per ATR basket for Irradiated fuel rods for this amendment application.

3.2.2 NCT

The applicant performed thermal evaluations using MURR HEU fuel of 1,264 W and MITR-II HEU fuel of 1,200 W because their maximum decay heat loads exceed the heat loads of other types of fuels. The applicant presented the NCT temperatures for BRR packaging with MURR HEU fuel in SAR table 3.3-1 and MITR-II HEU fuel in SAR table 3.3-2, respectively, and summarized the maximum NCT temperatures in SAR table 3.1-1 when transporting irradiated HEU fuels. The applicant used the governing bulk average backfill gas temperature of 259°F for MURR HEU fuel and calculated an NCT cavity pressure of 5.2 psig which is below the MNOP of 15 psig.

The staff finds that (1) the thermal evaluations of MURR HEU fuel of 1,264 W and MITR-II HEU fuel of 1,200 W bound thermal evaluations of other types of fuels, and (2) the maximum fuel and packaging component temperatures, shown in SAR table 3.1-1, are below the allowable limits for NCT. Therefore, the staff confirmed that the maximum fuel and packaging component temperatures when transporting irradiated fuels proposed in this amendment will be below the allowable limits for NCT, and the maximum package cavity pressures will be below MNOP for NCT. Therefore, thermal analyses of NCT and HAC are not necessary when transporting other types of HEU fuel.

3.2.3 Vacuum Drying

The applicant performed thermal evaluation of the vacuum drying operations with MURR HEU fuel with a bounding heat load of 1,264 W, as described in SAR section 3.3.3, “Cask Draining and Vacuum Drying Operations,” and presented the peak temperatures of the packaging components in SAR table 3.3-6 and figure 3.3-10.

The applicant stated, in SAR section 3.2.2, that aluminum has a melting point of approximately 1,100°F, and therefore the peak HEU and LEU aluminum clad fuels shall be at or less than 530°C (986°F) for package draining and vacuum drying operations. The applicant selected the lower bounding permissible cladding temperature limit of 932°F (500°C), instead of 400°F (204 °C) limit for structural strength considerations, to cover both steel and aluminum cladding, as described in SAR section 3.2.2, because the aluminum cladding is not relied upon for structural strength of the fuel during vacuum drying operations.

The NRC staff reviewed SAR sections 3.2.2 and 3.3.3, including table 3.3-6 and figure 3.3-10, and confirmed that (a) the aluminum cladding has a melting point up to 1,100°F (593°C) and use of 932°F (500°C) as the aluminum cladding limit is acceptable for HEU aluminum clad fuel for vacuum drying operations, and (b) the peak temperatures of the packaging components, including fuel plate, lead, and vent/drain port seals, are below their maximum allowable limits under a minimum vacuum drying pressure of 1 torr to 3 torr.

3.2.4 HAC

The applicant performed the HAC thermal evaluation, as described in SAR section 3.4, with the bounding total decay heat of 1,264 watt for MURR HEU fuel. The applicant presented the HAC peak temperatures in SAR table 3.4-1 and then calculated the maximum package cavity pressure of 8.8 psig using the ideal gas law.

The staff reviewed SAR section 3.4, including initial conditions and fire test conditions for HAC, and finds that (1) use of the MURR HEU fuel (decay heat of 1,264 W) will bound thermal evaluations of other types of fuel, and (2) the maximum temperatures of the package components (e.g., fuel plates, lead, and containment seals at lid/vent port/drain port) when transporting the MURR HEU fuel are below the corresponding allowable HAC limits (SAR table 3.4-1). The staff also confirmed that the maximum package cavity pressure of 8.8 psig is below the design pressure of 25 psig for HAC. Therefore, thermal evaluations of the HAC are not necessary when transporting other types of HEU fuel.

3.3 Thermal Evaluation of RITC Payload

3.3.1 Contents Decay Heat

The applicant requested approval for the addition of the segmented fuel rods to the BRR package using the RITC basket. The decay heat of the RITC payload is 180 W distributed between six fuel segments, as described in SAR section 3.7, “Thermal Evaluation for RITC Payload.”

The applicant performed a thermal evaluation to verify that the BRR package, loaded with RITC payload, will meet 10 CFR Part 71 requirements and that the maximum temperatures of all packaging components, the RITC payload, and the accessible package surfaces will remain within their respective limits under NCT, HAC, and vacuum drying operations, as described in SAR section 3.7, “Thermal Evaluation for RITC Payload.”

3.3.2 NCT

The applicant performed NCT thermal analysis using the same 3-D thermal model of the BRR package used to license the BRR package. The applicant stated, in SAR section 3.7, that with the personnel barrier not included, the updated model of the BRR package includes the RITCs, Rod-in-Tubes (RITs, 180 W) and fuel rods in the model, as detailed in SAR section 3.7.5, “RITC Analytical Thermal Model.”

The applicant presented, in SAR table 3.7-1, the predicted NCT component temperatures of the BRR package, loaded with RITC payload and demonstrated that (1) thermal margins exist for all packaging components and payload components for transporting the RITC payload under NCT, and (2) the package maximum surface temperature of 114°F is below 185°F limit in an exclusive-use shipment, in compliance with 10 CFR 71.43(g), and (3) the maximum pressure of 3.1 psig within the package cavity is bounded by the MNOP of 15 psig.

The staff accepts that the maximum fuel and packaging component temperatures for the BRR package loaded with RITC payload (180 W) are below the allowable limits for NCT, and the maximum package cavity pressure is below the MNOP for NCT.

3.3.3 Vacuum Drying and HAC

The applicant stated, in SAR section 3.7, that the NCT payload and packaging component temperatures and the package cavity pressure of the BRR package, when transporting the RITC payload, are bounded by those found in other payloads (e.g., HEU fuels) by a wide range. Therefore, there will be no significant impact on the packaging during HAC and vacuum drying operations (e.g., draining and drying operations) with the RITC payload.

The staff accepts that based on the NCT analysis, the BRR package loaded with RITC payload will meet thermal requirements under loading operations and HAC. The staff confirmed that thermal evaluation of the package, loaded with RITC payload of the lower decay heat (180 W), will be bound by other payloads of the higher decay heat (e.g., HEU fuels) under loading operations and HAC.

3.4 Thermal Evaluation of LEU Fuels

3.4.1 Contents Decay Heat

The applicant described the thermal evaluation of the BRR package loaded with LEU fuel in SAR section 3.8, “Thermal Evaluation of LEU Fuels,” and stated that the BRR package has the maximum decay heat loadings of 1,264 W per shipment for MURR LEU fuel, 1200 W per shipment for MITR LEU fuel, and 960 W per shipment for ATR LEU fuel.

3.4.2 NCT

The applicant performed thermal analysis for each fuel type (MURR LEU fuel, MITR-II LEU fuel, and ATR LEU fuel), as described in SAR section 3.8, to verify that the BRR package will have the maximum temperatures of packaging components, payload, and accessible package surfaces below their respective limits under NCT.

The applicant presented the maximum NCT package temperatures in SAR table 3.8.3-1 for MURR LEU fuel, table 3.8.3-2 for MITR-II LEU fuel, and table 3.8.3-3 for ATR LEU fuel, and summarized the maximum fuel and component temperatures and the maximum package cavity pressure in SAR table 3.8.1-1 and table 3.8.1-2, respectively, when transporting the irradiating LEU fuels under NCT. The applicant assumed that the cavity gas reaches a bulk average temperature that is equal to the mean of the average inner shell temperature and the average fuel basket temperature and then calculated the maximum cavity pressure of 5.7 psig per ideal gas law.

The NRC staff reviewed SAR section 3.8, including initial conditions and fire test conditions, and accepts that (1) use of the ATR LEU fuel (decay heat of 1,264 W) will bound thermal evaluations of other types of fuel, and (2) the predicted NCT packaging component temperatures are below the required limits for transportation of LEU fuels. The staff finds that the maximum accessible surface temperature of the package loaded with LEU fuel is below the limit of 185°F for an exclusive use shipment, in compliance with 10 CFR 71.43(g). The staff also confirmed that the calculated NCT maximum package cavity pressures of 5.7 psig for ATR LEU fuel and 5.2 psig for both MURR LEU fuel and MITR-II LEU fuel are below the MNOP of 15 psig.

3.4.3 Vacuum Drying

The applicant conducted a thermal evaluation of the vacuum drying operations by assuming air as the backfill gas during draining and vacuum drying operations and removing the impact limiters to simulate a bare cask inside the reactor facility, as described in SAR section 3.8.3.3, "Vacuum Drying Operations." The thermal evaluation (a transient analysis) was conducted for a period of eight hours followed by a steady state evaluation to illustrate the heat-up rate and establish the peak temperatures that would occur if the helium back-fill is not established.

The applicant presented the bounding transient heat-up during vacuum drying in SAR figure 3.8.3-16 which shows that more than 10 hours are required for the peak fuel plate temperature to reach a transient temperature of 400°F and approximately 18 hours are needed for the peak fuel plate to achieve a steady-state temperature of 552°F.

The NRC staff recognized that a steady-state temperature of 552°F is still below the limit of 620°F for the fuel plate, and therefore, the indefinite operation with air-filled or nitrogen-filled conditions of vacuum drying is permissible. The staff also noted, in SAR figure 3.8.3-16, that the containment seals remain well below their long-term limit of 250°F.

The applicant stated, in SAR section 3.8.3.4, "Cask Cavity with Helium Gas," that after the fuel drying is complete, the package cavity is backfilled with helium to cool down the package component temperatures from the peak level under vacuum drying conditions to their associated limits for NCT. The applicant performed a cool-down transient analysis after helium backfill, and found that a period of less than 6 hours is required to lower the peak fuel plate temperature to below 400°F and a period of approximately 8 hours is required to reduce the fuel and package component temperatures to those reported in SAR section 3.8.3.1.1.3, " ATR Fuel Element Payload" and SAR table 3.8.3-3.

The staff accepts that the temperatures of the package and payload will be at or near those computed for the NCT hot (ambient 100°F) without insolation before transport, as the final leak testing, package closure, and preparation for transportation will take much longer than eight hours. The staff confirmed that the results presented in SAR section 3.8.3.4 demonstrate that the steady-state operations under package draining and vacuum drying conditions are permissible without exceeding the maximum allowable component temperature limits for LEU fuels (MURR LEU fuel, MITR-II LEU fuel and ATR LEU fuel).

3.4.4 HAC

The applicant performed thermal analysis for each fuel type (MURR LEU fuel and ATR LEU fuel), as described in SAR section 3.8.4, to verify that the BRR package will have the maximum temperatures of packaging components, payload, and package cavity pressures below their respective limits under HAC. The applicant assumed an initial temperature distribution equivalent to the package at steady-state conditions with 100°F, ambient and insolation, in compliance with the requirement of 10 CFR 71.73(b).

The applicant simulated the worst-case damage arising from the postulated HAC free drop and puncture testing, increased the emissivity/absorptivity to have more fire heat into the package, and added heat transfer via radiation within the impact limiter enclosures with an emissivity of 0.95 to account for potential loss of polyurethane, etc. The applicant presented the maximum package temperatures in SAR table 3.8.4-1 for ATR LEU fuel and table 3.8.4-2 for MURR LEU fuel, when transporting the irradiating LEU fuels under HAC.

The NRC staff reviewed SAR section 3.8, including initial conditions and fire test conditions, and accepts that (1) use of the ATR LEU fuel and MURR LEU fuel will bound thermal evaluations of MITR-II LEU fuel, and (2) the predicted fuel and packaging component temperatures are below the required limits for transportation of LEU fuels under HAC. The staff confirmed that the calculated HAC maximum package cavity pressures of 8.5 psig for ATR LEU fuel and 9.0 psig for MURR LEU fuel are below the design pressure of 25 psig.

3.5 Differential Thermal Expansions

The applicant described evaluations of the NCT and HAC differential thermal expansions in SAR section 2.7.4.2 and stated that differential thermal expansions can be neglected due to safety margins of the package component temperatures under NCT and the clearance between the fuel and the basket will not be significantly affected by the package component temperatures resulting from the HAC fire event.

The NRC staff reviewed evaluation of the differential thermal expansions (SAR section 2.7.4.2) and thermal evaluations of HEU fuels (SAR sections 3.3 and 3.4), RITC payload (SAR section 3.7) and LEU fuels (SAR section 3.8). The staff accepts evaluation of the differential thermal expansions for the BRR package loaded with HEU fuels, RITC payload, or LEU fuels because of the safety margins in fuel and package component temperatures under NCT and HAC.

3.6 Evaluation Findings

Based on review of the statements and representations provided in the application, as supplemented, the NRC staff found the following:

F3-1 All fuel and packaging component temperatures of the BRR package loaded with HEU fuels remain well below their allowable limits under NCT, vacuum drying operations, and HAC. The maximum package cavity pressure is well below the MNOP for NCT and below the design pressure for HAC.

F3-2 All fuel and packaging component temperatures of the BRR package loaded with LEU fuels remain well below their allowable limits under NCT, vacuum drying operations, and HAC. The maximum package cavity pressure is well below the MNOP for NCT and below the design pressure for HAC.

F3-3 All fuel and packaging component temperatures of the BRR package loaded with RITC payload remain well below their allowable limits under NCT, vacuum drying operations, and HAC. The maximum package cavity pressure is well below the MNOP for NCT and below the design pressure for HAC.

F3-4 The maximum accessible surface temperature of the BRR package, loaded with HEU fuels, LEU fuels or RITC payload will be below the 185°F limit in an exclusive use shipment, in compliance with 10 CFR 71.43(g).

Therefore, NRC staff concludes that (1) the BRR package thermal evaluations for vacuum drying operations, NCT, and HAC have been adequately described by the applicant; and (2) the BRR package, loaded with the proposed HEU fuels, LEU fuels or RITC payload, meets the acceptance criteria of NUREG-2216 and meets the regulatory requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

There is no change in containment system for the package loaded with HEU fuels, RITC payload, or LEU fuels proposed in this amendment. All package operations, reviewed by the NRC in the previous application, are still applicable to this amendment.

5.0 SHIELDING EVALUATION

This section of the SER documents the NRC staff's evaluation of shielding design changes to determine if the package continues to meet the dose rate requirements of 10 CFR Part 71 under the conditions described in 10 CFR 71.71 and 71.73. The package is designed to be transported under exclusive use. The staff's evaluation follows the guidance of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material" (SRP).

The NRC staff evaluated the proposed changes and determined the following have an impact on the package radiation safety design:

- A new RITC and RITC basket designed to accommodate LEU irradiated and segmented fuel rods with updated specifications

- New MURR, MITR-II, and ATR LEU fuel element payloads

5.1 Proposed Change No. 1: New RIT, RITC, and RITC basket and Updated LEU Irradiated and Segmented Fuel Rods Specifications

5.1.1 Description of Shielding Design

The contents include a new RIT, RITC, and RITC basket, however, the applicant did not include these new components in its shielding models. The applicant made no significant changes to the packaging for this proposed change.

5.1.1.1 Summary Tables of Maximum External Radiation Levels

Summaries of maximum irradiated fuel rod segments dose rates under NCT and HAC are shown in SAR tables 5.10-1 and 5.10-2, respectively.

5.1.2 Radioactive Materials and Source Terms

For this revision, the irradiated commercial LEU contents type remains unchanged. The applicant proposed a new RIT, RITC, and RITC basket design. Segmented, irradiated rods are contained within stainless steel pipes with the ends enclosed via standard pipe fittings. The

whole assembly of an enclosed stainless steel pipe containing rods is considered a RIT. Up to three RITs are then placed within the three carrier tubes inside of a RITC, which varies from 12 to 53 in. in height, depending on the maximum length of the irradiated rod segments the RITC is designed to hold. Each RITC carrier tube shall contain only one RIT regardless of length (i.e., stacking of RITs within a RITC carrier tube is not allowed). There are two tubes within a RITC basket, each designed to hold a maximum of two RITCs each (i.e., stacking of RITCs within a RITC basket tube is allowed). The RITC basket tubes are nominally 3.5 in. outer diameter and fabricated from 1/8-in.-thick tube stock. Other tube structures in the RITC basket shall not be used to transport radioactive material and are for structural support only.
With the limits listed above, the maximum capacity payload consists of six full RITs containing 8.40 kg uranium. The contents are limited to enrichments between 0.711 and 5.0 weight percent U-235 and with maximum burnup determined by the equations in section 5.10.2 of the application.

5.1.2.1 Source Term Calculation Methods

The applicant relied on predictive analyses conducted by Idaho National Laboratory (INL), which are documented in Reference 2. INL staff calculated isotopic inventories with TRITON and subsequent decay with ORIGEN, both which are part of the SCALE 6.2.4 code suite. The verification and validation of the INL SCALE installation is documented in Reference 3.
The neutron dose rates are considered limiting for the BRR package (2), and the applicant communicated to INL staff the maximum neutron source strength that would reach the surface dose rate limit in 10 CFR 71.47. Per the applicant's instructions, INL staff opted to calculate with an intensity of 85 percent of the bounding value to provide additional margin. This new bounding value was evenly divided between the maximum six RITs and then again divided by the length of each fuel rod to yield a final bounding neutron source expressed as neutrons per second per inch of fuel.

INL chose the BWR and PWR assemblies with the lowest water/fuel ratio to determine bounding neutron source terms. A lower water/fuel ratio increases the production of Cm-244 and Cm-252 and thus increases the neutron source. These nuclides dominate the neutron source in irradiated fuel at higher burnups. Given the bounding burnup-enrichment combinations occur above 50.0 GWd/MTU, the staff finds the selection of bounding assemblies acceptable.
INL staff calculated isotopic inventories for assemblies with 0.71, 2.0, 3.0, 4.0, and 5.0 weight percent U-235 initial enrichment at 10 GWd/MTU intervals from 0 to 100 GWd/MTU. INL staff split the burnup calculations from 0 to 10 GWd/MTU into 10 intermediate steps (i.e., 1 GWd/MTU per step), and each subsequent 10 GWd/MTU interval into five intermediate steps (i.e., 2 GWd/MTU per step). INL staff then decayed the isotopic inventories for 1 year to determine the post cooling-time inventories. INL staff determined a bounding burnup for each initial enrichment by interpolating between the nearest calculated inventories. INL staff calculated gamma and neutron source spectra from these 1-year cooled burnup/enrichment combinations.

The calculation process described above largely follows the recommendations in Reference 11 for source term calculations, and the staff finds it acceptable to use here. Given the margin provided by the 15 percent reduction in maximum source intensity and the selection of bounding energy groups (see 5.1.2.2 and 5.1.2.3 below), the staff finds reasonable assurance that the source spectra and intensity are conservative and actual dose rates will remain below the limits of 10 CFR 71.47.

5.1.2.2 Gamma Sources

In calculating the bounding gamma source spectrum, the applicant selected the highest source strength for each energy bin from any of the burnup and enrichment combinations, both BWR and PWR, and combined the bounding bins into a single bounding gamma source spectrum. The staff finds this acceptable as it will conservatively maximize calculated dose rates.

5.1.2.3 Neutron Sources

In calculating the bounding neutron source spectrum, the applicant selected the highest source strength for each energy bin from any of the burnup and enrichment combinations, both BWR and PWR, and combined the bounding as a single bounding neutron source spectrum. The staff finds the above acceptable since it will conservatively maximize calculated dose rates.

5.1.3 Shielding Model and Model Specifications

5.1.3.1 Configuration of Source and Shielding

The applicant used a shielding model in this application that is the same from previously approved revisions. The applicant modeled lead slump and radial lead shrinkage to the same maximum extent expected. The NRC staff found these values acceptable for prior revisions. The applicant omitted the encapsulation tubes, RITCs, and RITC basket from its shielding model. These components would provide some shielding effects, and this will maximize calculated dose rates. Therefore, the staff finds this simplification conservatively acceptable. The applicant credits the spacing provided by the impact limiters under NCT but does not include the impact limiter material. Prior review has shown the presence of impact limiters under NCT credible, and omitting the impact limiter material maximizes calculated dose. Under HAC, the applicant omits the impact limiters completely. This will conservatively maximize calculated dose rates since some material will remain. As a result, the staff finds the applicant’s treatment of the impact limiters in its NCT and HAC models acceptable.

The sources are modeled as constant activity, homogenous UO_2 cylinders for both gamma and neutron sources. For NCT, the applicant’s model relies on the RITCs to keep the RITs stacked and within the RITC basket opening. Since the applicant showed, no credible damage will occur to the RITC basket under NCT, the staff finds this location to be acceptable. The applicant grouped the source cylinders together closest to the package wall. This will maximize calculated dose rates, which the staff finds acceptable.

5.1.3.2 Material Properties

The components of the BRR package that contribute to radiation shielding are comprised of lead and stainless steel. The staff reviewed the applicant’s compositions in SAR tables 5.10-8 and 5.10-9 and noted they remain largely unchanged from revisions that the NRC staff previously found acceptable. The biggest changes are a 1 percent increase in stainless steel density and the inclusion of separate isotopes for each element. The change in density is small, and the total content of each remained consistent. As a result, staff finds the material properties consistent with previously approved evaluations.

5.1.4 Shielding Evaluation

5.1.4.1 Methods

The applicant used MCNP6.3 for its dose rate analyses for rod segments. The applicant used two cross-section libraries based on EPICS2014 and ENDF/B-VIII.0 for photons and neutrons, respectively. These are the default libraries distributed by Los Alamos National Laboratory for use with MCNP6.3. MCNP is a three-dimensional, Monte Carlo transport code that has been well vetted through a long history of use in radiation safety applications. As a result, the staff finds its use here by the applicant acceptable. The EPICS2014 library is based on ENDF/B-VII.1 nuclear data, which has been approved by NRC staff in previous applications. The ENDF/B-VIII.0 has been approved by NRC staff in previous applications. Therefore, the NRC staff finds the applicant's cross-section libraries acceptable.

Except for conical surfaces, the applicant used segmented mesh tallies to determine dose rates. For conical surfaces, the applicant used cylindrical surface tallies segmented into eight circumferential segments. The staff reviewed the spatial aspects of the applicant's tallies and found they are sufficient to cover the package and determine the location of the maximum dose rates.

5.1.4.2 Code Input and Output Data

The applicant provided sample input in SAR section 5.10.5.1.

5.1.4.3 Fluence-Rate-to-Radiation-Level Conversion Factors

The applicant used the American National Standards Institute/American Nuclear Society (ANSI/ANS) 6.1.1-1977 dose conversion factors. This follows SRP recommendations, and the staff finds the use of these factors acceptable.

5.1.4.4 External Radiation Levels

The maximum dose rates are shown in SAR table 5.10-10.

5.1.5 Confirmatory Analyses

The staff used SCALE 6.3 in its analyses, with the 28 and 19 group ENDF/B-VII cross-section libraries for neutrons and photons, respectively. The staff modeled the steel and lead components of the BRR packaging. The staff simplified or omitted some small features of the geometry that have a little, if any, significant effect on calculated dose rates. For the staff's general model, those typically involved omitting drains, vents and other small features. The staff performed additional calculations with drains and vents modeled as voided cylinders to verify that the dose rates at 1 m and 2 m are not significantly impacted even though there are some localized increases in dose rates. The staff modeled the RITC basket outer shell and RITC tubes. The staff modeled the RITC basket material as void to match the applicant's model. The staff placed the irradiated fuel rods on the edge of a single RITC tube, matching the applicant's NCT model. For its HAC model, the staff moved the irradiated fuel rods to the inside of the BRR cavity, which also matches the applicant's assumed configuration under HAC. The staff sampled the source particles uniformly along the fuel rod contents. The staff results largely follow those of the applicant and provide reasonable assurance that the applicant's modeling is conservatively bounding.

5.2 Proposed Change No. 2: MURR, MITR-II, and ATR LEU Fuel Elements

5.2.1 Description of Shielding Design

The packaging and fuel baskets remain unchanged from previous, NRC-approved revisions.

5.2.1.1 Summary Tables of Maximum External Radiation Levels

Summaries of maximum LEU fuel elements dose rates under NCT and HAC are shown in tables 5.9-1 and 5.9-2 of the application, respectively.

5.2.2 Radioactive Materials and Source Terms

The proposed contents for this amendment are MURR LEU, MITR-II LEU, and ATR LEU fuel elements. These fuel elements are LEU versions of existing HEU fuel elements that have been approved by NRC staff for transport in the BRR package. The LEU variants of the fuel is aluminum clad uranium-molybdenum alloy (U-10Mo) plate fuel. The overall size and shape of each fuel element remains unchanged, however the number of fuel plates and spacing is different.

5.2.2.1 Source Term Calculation Methods

5.2.2.1.1 ATR Source Term Calculation

The applicant relied on analyses conducted by INL, which are documented in Reference 4. INL staff calculated isotopic inventories with MCNP5 and subsequent decay with ORIGEN2. The verification and validation of this INL MCNP5 and ORIGEN2 installation is documented in Reference 5.

INL staff calculated a series of isotopic inventories burned for 60 days with power histories that varied from 1.0 to 9.0 MW per single LEU ATR fuel element. This range encompasses the typical irradiation conditions in the ATR core. INL staff analyzed an additional power history of a single LEU ATR fuel element through three cycles to evaluate the maximum practical burnup. This roughly corresponds to the 8 MW single-element evaluation which remains bounded by the 9 MW study. INL staff also calculated a full 40-element LEU ATR core exposure to 15,000 MWd. INL staff selected a power history from Reference 6, which describes the maximum core burnup calculation used to support the ATR SAR. The average element in this study ended up with a final irradiation of 375 MWd, equivalent to 6.25 MW for 60 days, which falls within the bounds of the single-assembly evaluations. Prior studies (4) showed that this burnup exceeds what is possible in the HEU-fueled ATR due to excessive 235U depletion. Therefore, staff finds reasonable assurance that this burnup remains bounding for LEU ATR fuel elements.
INL staff calculated source term data at 13 different decay times from 0.0 hours (i.e., shutdown) to 1 year. The applicant showed the 90-day, post-irradiation isotopic composition from the depletion calculations for the LEU ATR fuel in SAR table 5.9-3. To meet the thermal limits of the BRR package, additional cooling time is required, and the applicant performed additional decay calculations. The applicant used the ORIGEN code included with the SCALE 6.2.4 distribution. The applicant used decay libraries based on ENDF/B-VII.1 nuclear data. Both the nuclear data and ORIGEN code have a long history of use in irradiated fuel composition calculations and the staff finds the applicant’s use of the code and cross-section libraries acceptable.

5.2.2.1.2 MITR-II Source Term Calculation

The applicant relied on analyses conducted by Argonne National Laboratory (ANL), which are documented in Reference 7. ANL staff selected a bounding fuel element with the greatest predicted fission density, which would also maximize the calculated source term. ANL staff modeled MITR depletion with ORIGEN2.2 with a custom cross-section library representative of the bounding fuel element that was generated as part of the preliminary design verification (8). INL staff calculated isotopic inventories with ADDER v1.0.1, which uses pre-calculated data from ORIGEN2.2 to determine isotopic composition. ORIGEN2.2 has a long history of use with irradiation and depletion calculations, and the staff finds its use acceptable. ANL staff evaluated two irradiation histories, a “split” power case that is more representative of actual operations, and a constant power over the entire irradiation history. The constant power case resulted in larger decay heat and activity across all irradiation time points (7), and the staff finds the selection of this bounding case for the LEU MITR-II depletion calculations acceptable. The applicant showed the 90-day, post-irradiation isotopic composition from the depletion calculations for the LEU MITR fuel in SAR table 5.9-4. To meet the thermal limits of the BRR package, additional cooling time is required, and the applicant performed additional ORIGEN (SCALE 6.2.4) decay calculations.

5.2.2.1.3 MURR Source Term Calculation

The applicant relied on analyses conducted by ANL, which are documented in Reference 9. ANL staff used ORIGEN2.2 to calculate isotopic compositions following irradiation and decay. The thermal LWR libraries included with ORIGEN2.2 are representative of conditions in the MURR core. However, the plate fuel geometry with monolithic U-10Mo alloy differ from the typical UO2 pin fuel. Using neutron flux and reaction rate tallies computed with MCNP5 for previous safety analyses (10), ANL staff created substitute cross-sections for actinides and fission products considered most important to calculate reaction rates and isotopic content needed for transportation activities (9). Evaluations of the HEU MURR fuel elements showed that the smallest fuel plate, plate 1, had the highest discharge fission density. Subsequent evaluations showed this remained the case for LEU MURR fuel elements. Based on plate burnup data, the peak to average fission density ratio for plate 1 is 1.27. ANL staff assumed a 27 percent increase in element discharge burnup, roughly equivalent to 33,000 MWd/MTU, which still keeps plate 1 within the licensed fission density limit for U-10Mo fuel. The applicant showed the 90-day, post-irradiation isotopic composition from the depletion calculations for the LEU MURR fuel in table 5.9-5 of the application. To meet the thermal limits of the BRR package, additional cooling time is required, and the applicant performed additional ORIGEN (SCALE 6.2.4) decay calculations.

5.2.2.2 Gamma Sources

The applicant showed the gamma source spectrum and strength for the LEU fuel elements in SAR table 5.9-10. These values correspond to the maximum loading of eight LEU fuel elements, which maximizes the calculated dose.

5.2.2.3 Neutron Sources

The applicant showed the neutron source spectrum and strength for the LEU fuel elements in SAR table 5.9-11. These values correspond to the maximum loading of eight LEU fuel elements, which maximizes the calculated dose.

5.2.3 Shielding Model and Model Specifications

5.2.3.1 Configuration of Source and Shielding

The shielding components of the BRR packaging under NCT and HAC, including the lead slump and shrinkage, remain in the same configuration as previously approved by NRC staff and discussed above in section 5.1.3.1. The applicant modeled the fuel baskets, which provide some shielding, in the same configuration as approved in previous revisions by NRC staff. The applicant modeled the entire source within the outermost fuel plate of the MURR LEU and ATR LEU fuel elements pressed against the outer wall of the basket locations. This will move the source term from inner fuel plates closer to the package wall, conservatively increasing calculated dose rates. For MITR-II fuel, the applicant modeled source term that was distributed throughout the fuel element volume. As discussed below in section 5.2.4, the applicant modeled the source regions as void. Any non-conservative dose rate impacts due to the source volume location will be more than offset by the lack of self-shielding by the MITR-II fuel elements. For these reasons, the staff finds the applicant's source configuration acceptable.

5.2.3.2 Material Properties

The components of the BRR packaging and fuel baskets that contribute to radiation shielding are comprised of lead and stainless steel. The applicant's compositions for stainless steel and lead in SAR tables 5.9-13 and 5.9-15, respectively, are the same as SAR tables 5.10-8 and 5.10-9 which staff found acceptable as discussed in section 5.1.3.2 above.

5.2.4 Shielding Evaluation

5.2.4.1 Methods

The applicant used MCNP 6.2 with the default cross-section libraries that are based on ENDF/B-VII.1 and ENDF/B-VI.8 for neutrons and photons, respectively. The code and cross-section libraries have a long history of use calculating dose rates for transportation packages and the staff finds the use here acceptable. For each LEU fuel type, the applicant evaluated neutrons and photons separately in individual calculations. The applicant captured secondary photons from neutron interactions with a third calculation.

The applicant credits the spacing provided by the impact limiters under NCT but does not include the impact limiter material. Prior review has shown the presence of impact limiters under NCT credible, and omitting the impact limiter material maximizes calculated dose. Under HAC, the applicant omits the impact limiters completely. This will conservatively maximize calculated dose rates since some material will remain. As a result, the staff finds the applicant's treatment of the impact limiters in its NCT and HAC models acceptable.

Except for the source regions, the applicant modeled the empty space within the BRR package as dry air. The applicant modeled any source region as void. This conservatively removes attenuation due to self-shielding of the U-10Mo fuel alloy, cladding, and fuel element structural components.

The applicant evaluated the dose rate effect of shifting the fuel elements maximally both upward and downward within the basket locations.

5.2.4.2 Code Input and Output Data

The applicant provided sample inputs of the decay and shielding calculations in SAR sections 5.9.5.1 and 5.9.5.2, respectively.

5.2.4.3 Fluence-Rate-to-Radiation-Level Conversion Factors

The applicant used the ANSI/ANS-6.1.1-1977 dose conversion factors. This follows SRP recommendations, and the staff finds the use of these factors acceptable.

5.2.4.4 External Radiation Levels

The maximum dose rates for ATR LEU, MITR LEU, and MURR LEU are shown in SAR tables 5.9-16, 5.9-17, and 5.9-18, respectively.

5.2.5 Confirmatory Analyses

The staff used SCALE 6.3 in its analyses, with the 28 and 19 group ENDF/B-VII cross-section libraries for neutrons and photons, respectively. The staff modeled the steel and lead components of the BRR packaging. The staff simplified or omitted some small features of the geometry that have a little, if any, significant effect on calculated dose rates. For the staff's general model, those typically involved omitting drains, vents and other small features. The staff performed additional calculations with drains and vents modeled as voided cylinders to verify that the dose rates at 1 m and 2 m are not significantly impacted even though there are some localized increases in dose rates. The staff modeled the MURR, MITR-II, and ATR fuel element baskets. The staff's basket model included the inner and outer shells, the bottom support plate, and the steel that separates the fuel element cavities. The staff modeled the LEU fuel plates, cladding, and structure, but modeled the non-source material as void to match the applicant's configuration. In order to maintain a unique material number to sample source particles in a specific fuel plate, the staff included the material properties of the plate and reduced the density by a factor of 1.0E-8 to minimize any self-shielding effects. The staff sampled the source particles from the position of the outermost fuel plate in each of the LEU fuel elements and sampled source particles from the entire MITR basket location to match the applicant's configuration. The staff results largely follow those of the applicant and provide reasonable assurance that the applicant's modeling is conservatively bounding.

5.3 Evaluation Findings

As described in sections 5.1 and 5.2 above, the staff considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, and its own confirmatory analysis in reaching the following findings:

F5-1 The staff has reviewed the BRR revision application and finds that it adequately describes the package contents and the package design features that affect shielding in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b), and provides an evaluation of the package's shielding performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a). The descriptions of the packaging and the contents are adequate to allow for evaluation of the package's shielding performance. The evaluation is appropriate and bounding for the packaging and the package contents as described in the application.

F5-2 The staff has reviewed the BRR revision application and finds that prior NRC review remains applicable to demonstrate the package has been designed so that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), and in compliance with 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), the external radiation levels do not significantly increase.

F5-3 The staff has reviewed the BRR revision application and finds that it demonstrates that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), external radiation levels do not exceed the limits in 10 CFR 71.47(b) for exclusive-use shipments.

F5-4 The staff has reviewed the BRR revision application and finds that prior NRC review remains applicable to demonstrate that under the tests specified in 10 CFR 71.73 (hypothetical accident conditions), external radiation levels do not exceed the limits in 10 CFR 71.51(a)(2).

F5-5 The staff has reviewed the BRR revision application and finds that it identifies codes and standards used in the package's shielding design and in the shielding analyses, in compliance with 10 CFR 71.31(c).

F5-6 The staff has reviewed the BRR revision application and finds that it includes RITC basket loading and unloading operations descriptions, acceptance tests, and maintenance programs that will ensure that the package is fabricated, operated, and maintained in a manner consistent with the applicable shielding requirements of 10 CFR Part 71. For ATR LEU, MITR-II LEU, and MURR LEU Fuel Elements, the operations, acceptance tests, and maintenance programs remain unchanged from the HEU variants that NRC staff previously found acceptable.

Based on the review of the information and representations provided in the application, along with the independent and confirmatory calculations, the NRC staff has concluded that there is reasonable assurance the proposed contents satisfy the shielding requirements, and the radiation level limits in 10 CFR Part 71.

5.4 References

1. NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Materials,” U.S. Nuclear Regulatory Commission, August 2020.

2. P.F. O'Donnell, “Determination of Loading Cures for PWR and BWR Rod Segments in the BRR Transportation Package,” ECAR-7744, Revision 0, Idaho National Laboratory, April 2024.

3. B. Carbno, “Verification and Validation Testing of SCALE6/ORIGENS Application,” Idaho National Lab. (INL), Idaho Falls, ID, Technical Evaluation TEV-3686 Revision 5, July 2023.

4. J.W. Sterbentz, “Calculated Source Terms for LOWE Fuel Elements Irradiated in ATR,” ECAR-4505, Revision 0, Idaho National Laboratory, May 2019.

5. PLN-5650, “Test Plan for the MCNP and ORIGEN2 Application,” Revision 0, Idaho National Engineering Laboratory, July 2018.

6. M.L. Carboneau, “ORIGEN2 Calculated Core Inventories and Photon Source Term for the ATR SAR,” TRA-ATR-784, Revision 0, Idaho National Engineering Laboratory, May 1993.

7. K. Anderson, W. Cowherd, “Massachusetts Institute of Technology Reactor (MITR) Low-Enriched Uranium Fuel Element Preliminary Shipping and Backend Information,” ANL/RTR/TM-21/24, Revision 1, Argonne National Laboratory, August 2023.

8. W. Cowherd, “Alternate Neutronics Calculations for Preliminary Design Verification of Massachusetts Institute of Technology Reactor Low-Enriched Uranium Conversion, ANL/RTR/TM-21/7, Argonne National Laboratory, June 2021.

9. J.A. Stillman, et al., “University of Missouri Research Reactor (MURR) Low-Enriched Uranium Fuel Element Preliminary Shipping and Backend Information,” ANL/RTR/TM-20/14, Argonne National Laboratory, February 2021.

10. J.A. Stillman, et al., “Transition Core Planning and Safety Analysis in Support of LEU Fuel Conversion of the University of Missouri Research Reactor (MURR),” ANL/RTR/TM-19/18, Argonne National Laboratory, September 2020.

11. G. Radulescu, “Updated Recommendations Related to Spent Fuel Transport and Dry Storage Shielding Analyses,” NUREG/CR-7302, Oak Ridge National Laboratory, May 2023.

6.0 CRITICALITY EVALUATION

This section of the SER documents the NRC staff’s evaluation of criticality design changes to determine if the package continues to meet the criticality safety requirements of 10 CFR Part 71 under the conditions described in 10 CFR 71.71 and 71.73. The package is designed to be transported under exclusive use. The staff’s evaluation follows the guidance of NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material” (SRP).

6.1 Description of Criticality Design

In its SAR 6.1.1, Design Features, the applicant stated that five basket types are used to properly position the fuel within the cask cavity, and that these baskets limit the number of fuel elements that may be shipped at a given time in addition to controlling the spacing between the fuel elements. There are no poisons utilized in the package.

The applicant requested the addition of two different sets of authorized contents: irradiated fuel rod segments within encapsulation tubes in the RITC and LEU versions of the ATR, MITR, and MURR fuel elements within their corresponding fuel baskets. Previously, the package was authorized for full-length segments only, along with the HEU versions of the above fuel elements.

6.2 Segmented Rods Evaluation

6.2.1 RITC Model

The RITC contents include irradiated fuel rod segments transported in RITs within the RITC basket. The fuel rod segments will contain UO2 fuel with a maximum enrichment of 5 weight percent U-235, with a maximum burnup of 100 GWd/MTU. These rods will be contained within tubes and placed into RITCs, then loaded into the RITC basket. The total combined length of segments will not be greater than 125.75 cm. The maximum payload will be 6 RITs, with a total uranium content not exceeding 8.40 kg. Although the fuel will be irradiated, no burnup credit is taken by the applicant during the criticality evaluation.

To show compliance with 10 CFR Part 71 criticality requirements, the BRR package and contents were modeled by the applicant. Instead of modeling the rods and tubes explicitly within the contents, the applicant modeled a sphere of a uranium and water mixture within the package. To determine the most reactive configuration, the applicant did a study in which the uranium mass was kept constant while the radius of the circle increased. The study found the radius with the highest k_{eff} values, which in turn determined the most reactive concentration of moderator within the mixture.

While the applicant did not model the specific geometry of the RITC contents, the most reactive geometry for neutron multiplication is a sphere due to low neutron leakage. The applicant applied this concept to create a worst-case conservative model of the fuel rod segment contents. To demonstrate this, the staff analyzed two different optimally moderated geometries of the same uranium mass. The staff confirmed these assumptions and finds reasonable assurance that the applicant identified the most reactive condition.

When modeling the package, the addition of impact limiters to the model would increase spacing within the array cases and make reflection less effective during the single package cases. Consequently, to maintain conservative modeling, the impact limiter was neglected in the applicant's model. The cask surrounding the fuel elements was modeled by the applicant to provide neutron absorption and spacing parameters. For the contents, all structural aspects of the elements were ignored, therefore no credit was taken for neutron absorption of the structural material in the applicant's model.

The staff has reviewed the applicant's analysis of the package and concludes that the applicant used packaging features, content configurations, and material properties in the criticality safety analysis that are consistent with and bounding for the package's design basis.

6.2.2 Methods and Nuclear Data

The applicant used the MCNP6.2 Monte Carlo particle transport code to perform the criticality calculation with continuous-energy cross-sections based on ENDF/B-VII.1 and ENDF/B-VII.0. The staff reviewed the calculation parameters and found them to be acceptable.

6.2.3 Single Package and Package Array Evaluation

As stated in 10 CFR Part 71, under NCT and HAC, the package must stay subcritical. The applicant did not model NCT conditions due to the bounding conditions of the HAC model. For NCT, the model would keep the geometry of the fuel elements, creating more surface area for neutrons to escape without interaction, decreasing the k_{eff} values. This is discussed in the summary of the staff's review above.

The applicant's HAC model used the spherical fuel geometry described above. For the single package cases, the applicant modeled the package surrounded by 12 in. of full density water for full reflection. For the HAC array case, the applicant modeled the packages packed within an infinite hexagonal lattice with the smallest packing fraction possible. The small gaps between the packages are filled with water and a sensitivity study was conducted by the applicant to determine the density of the water between the packages that demonstrates maximum reactivity. The results are summarized below, where each k_{eff} value includes bias and uncertainty margins that will be discussed in the next section. The k_{eff} values for NCT and HAC are the same because they use the same HAC bounding model. The staff finds reasonable assurance that the HAC model bounds the NCT conditions. These k_{eff} values are all well below the applicant's calculated Upper Subcritical Limit (USL) of 0.92259.

Table 6.1 Summary of Criticality Calculations for Irradiated Fuel Segments

NCT	**k_{eff}**
Single Package	0.75427
Array	0.75426
HAC	**k_{eff}**
Single Package	0.75427
Array	0.75426

The Criticality Safety Index (CSI) determines the number of packages that can safely be transported within a single shipment. As stated in 10 CFR Part 71.59(b), "The value of the CSI may be zero provided that an unlimited number of packages are subcritical." This irradiated rod HAC array model uses an infinite array of packages, which the applicant demonstrated is subcritical, therefore the applicant determined the CSI to be zero. The staff has reviewed the applicant's analysis and concludes that the application specifies the number of packages that may be transported in the same vehicle through provision of an appropriate CSI in compliance with 10 CFR 71.35(b).

6.2.4 Benchmark Evaluations for RITC

The applicant used a non-parametric code, Whisper, to select benchmarks and evaluate the bias and bias uncertainty to ultimately determine a USL. NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," recommends the use of both parametric and non-parametric methods; however other sources recommend parametric methods over non-parametric since they "allow the analyst to generate a bias and bias uncertainty that are more reflective of the system being analyzed than [non-parametric] methods" (2). Since both the methodology and the code are called "Whisper," the staff will be specific in its naming for clarity.

The Whisper Code is a statistical analysis package developed by Los Alamos National Laboratory and is distributed with MCNP6.2. The Whisper Code uses sensitivity/uncertainty (S/U) techniques for benchmark selection and trending analysis, and is designed to provide "repeatable, quantitative, physics-based information to nuclear criticality safety analysts for determining USLs" using the Whisper Methodology (3). The Whisper Code uses the same MCNP6.2 code and continuous-energy neutron cross-section library based on ENDF/B-VII.1

nuclear data. This follows NUREG-2216 guidance that benchmarking use the same criticality code and cross-section libraries as the criticality analysis. The applicant's implementation of the Whisper Code (4) shows it uses the same dedicated hardware for both benchmarking and analyses. This follows the recommendations in NUREG-2216 for code validation. Whisper has been shown by the applicant to comply with the recommendations of ANSI/ANS 8.1 and 8.24 (5). Since NUREG-2216 also recommends following ANSI/ANS 8.1 and 8.24, the NRC staff finds the applicant's implementation of the Whisper software acceptable.

6.2.4.1 Experiments and Applicability

The applicant listed the 52 experiments that the Whisper Code selected and used to determine the bias and bias uncertainty in its analyses in table 6.10-10 of the application. NUREG-2216 recommends the selection of benchmark experiments that have the same materials, neutron spectrum, and configurations as the package evaluations. In this case, the applicant evaluated a moderated and reflected, homogenous sphere of uranium and water. The staff reviewed the list of selected benchmarks in table 6.10-10 of the application and noted that all are thermal uranium systems. The staff also noted that more than half are solution experiments, and all but two of the remaining benchmarks are heterogeneous arrays of rods, which are also systems with distributed uranium in moderator. This is expected given the evenly dispersed fissile material in the applicant's bounding criticality configuration. As a result, the staff concludes that there is reasonable assurance that the selected benchmark experiments have similar materials and configurations as the applicant's bounding criticality model. For neutron spectrum similarity, NUREG-2216 suggests evaluating the correlation of individual neutronic and materials parameters (e.g., EALF, H/X ratio, enrichment). Given that the parameters evaluated (e.g., EALF and H/X ratio) also contribute to neutronic similarity, it is reasonable to conclude that systems that are parametrically similar will also show a high degree of correlation with S/U techniques. All of the applicant's selected benchmark experiments show a high degree of correlation with a c_k value [correlation coefficient] greater than 0.94. Therefore, the staff finds reasonable assurance that the applicant has selected a sufficient number of benchmark experiments that are similar to the bounding configuration evaluated.

The staff has reviewed the application and concludes that the applicant's evaluations include an adequate benchmark evaluation of the calculations. The applicant identified and evaluated experiments that are relevant and appropriate for the package analyses.

6.2.4.2 Bias Determination

The applicant also provided a comparative analysis showing the calculated USL from Whisper and USLSTATS. For other Orano packages containing water-moderated uranium, the applicant showed that the Whisper Code would have yielded equivalent or more conservative (i.e., lower) USL values than the ones calculated by USLSTATS, which the NRC staff found acceptable for those packages (4). In addition, the proposed RITC contents consist of irradiated fuel, and the applicant does not rely on burnup credit for criticality safety. This adds additional conservative margin.

The applicant presented the results of the USL calculation for RITC contents in table 6.10-11 of the application. When compared to the maximum calculated k_{eff} shown in table 6.10-9 of the application, there is significant margin between the conservatively determined maximum k_{eff} and the USL.

6.3 Fuel Elements

6.3.1 ATR, MITR, and MURR Fuel Element Model

The contents of the LEU versions of the ATR, MITR, and MURR fuel elements have the same geometries and layouts as previously approved HEU versions. The fuel is a uranium-molybdenum core (U-10Mo) with an aluminum alloy cladding. The fuel has an enrichment of 19.75 ± 0.20 weight percent U-235.

The ATR, MITR, and MURR fuel elements were modeled by the applicant to determine compliance with 10 CFR Part 71 criticality requirements. The fuel elements are U-10Mo alloy plates surrounded by aluminum cladding. In each of the applicant's models, the "meat" of the fuel element, the fissile fuel center, is maximized by modeling the tolerances that minimize the aluminum cladding and maximize the volume and density of the U-10Mo metal. Within the design of the MURR fuel element the channel thickness is specified. For the conservative model, the channel thickness is increased due to the minimization of cladding thicknesses. For each fuel element, the mass of uranium modeled by the applicant was roughly 10 percent over the allowable, maximum mass. The staff reviewed these assumptions and determined them to be conservative.

Similar to the segmented rods case, when modeling the package, the impact limiter was neglected by the applicant. The structural elements of the package were included in the applicant's model. However, within the fuel elements, non-fuel components were not credited by the applicant and, instead, replaced with either air or water with a density to establish maximum reactivity.

The staff reviewed the applicant's isotopic composition for the U-10Mo metal, lead, aluminum alloy, stainless steel, dry air, and water and found them to be acceptable.

The staff has reviewed the package and concludes that the applicant used packaging features, content configurations, and material properties in the criticality safety analysis that are consistent with and bounding for the package's design basis.

6.3.2 Methods and Nuclear Data

The applicant used MCNP6.2 to perform the calculation with continuous-energy cross-sections based on ENDF/B-VII.1 and ENDF/B-VII.0. The staff reviewed the calculation parameters and found them to be acceptable.

6.3.3 Single Package and Package Array Evaluation

To show compliance with the subcriticality requirements of 10 CFR Part 71, the applicant evaluated each fuel element's NCT and HAC model for a single package and an array of packages. The NCT results are summarized in table 6.2. For the applicant's NCT model, the internal cavity is filled with air and all free volumes outside the internal cavity are filled with full density water, along with 12 in. of water surrounding the package for full neutron reflection. For the package arrays, the boundary conditions of the applicant's model are mirrored, effectively creating an infinite array of packages. All k_{eff} values are well below the corresponding USL of each element.

Table 6.2 NCT Single Package and Package Array Results

Fuel	k_{eff} Single Package	k_{eff} Package Array	USL
ATR	0.14137	0.24131	0.91964
MITR	0.13959	0.19701	0.91954
MURR	0.18104	0.25705	0.92006

The applicant's HAC model differs from the NCT by filling the internal cavity with water. A study was performed by the applicant to determine the density of water resulting in the greatest reactivity. Full-density water was found by the applicant to be the most reactive. For the array cases, mirrored boundary conditions were used by the applicant to model an infinite lattice. HAC results are summarized below in table 6.3. All k_{eff} values are well below their corresponding USLs.

Table 6.3 HAC Single Package and Package Array Results

Fuel	k_{eff} Single Package	k_{eff} Package Array	USL
ATR	0.70743	0.71727	0.91964
MITR	0.59563	0.61591	0.91954
MURR	0.79245	0.81621	0.92006

The applicant's CSI calculation is the same as above for the segmented irradiated fuel rod case. Each fuel element package array was modeled by the applicant to be infinite. Therefore, the CSI for each fuel element package is zero. The staff has reviewed the applicant's analysis and concludes that the application specifies the number of packages that may be transported in the same vehicle through provision of an appropriate CSI in compliance with 10 CFR 71.35(b).

6.3.4 Benchmark Evaluations for LEU MITR-II, LEU ATR, and LEU MURR Fuel

The applicant used the same non-parametric code, Whisper, to select benchmarks and evaluate the bias and bias uncertainty to ultimately determine a USL for the bounding configurations for the LEU MITR-II, LEU ATR, and LEU MURR fuel elements. The staff's evaluation on the use of the Whisper Code for a thermal, uranium system without strong absorbers is described in section 6.1.6 above. The BRR package does not rely on fixed neutron absorbers for the proposed LEU fuel elements, and the applicant modeled the bounding configurations as unirradiated fuel (i.e., no burnup credit), and the staff's discussion above remains applicable for LEU fuel elements.

6.3.4.1 Experiments and Applicability

The applicant listed the experiments that the Whisper Code selected and used to determine the bias and bias uncertainty in its analyses in table 6.11-24, 6.11-25, and 6.11-26 of the application for the LEU ATR, LEU MITR, and LEU MURR fuel elements, respectively. The Whisper Code identified and selected 60, 57, and 52 cases for the LEU ATR, LEU MITR, and LEU MURR fuel, respectively. The staff reviewed the list of selected benchmarks and noted that all are thermal uranium systems. The most highly correlated are LEU and IEU heterogeneous systems, and roughly 30-35 percent are HEU solution experiments for each of the LEU fuel element types. This is expected given the evenly spaced fuel plates in the applicant's bounding criticality configurations and the higher level of enrichment of the fuel elements compared to most LEU. As a result, the NRC staff finds that the selected benchmark experiments have similar materials and configurations as the applicant's bounding criticality models, and that the applicant has selected a sufficient number of benchmark experiments that are similar to the bounding configuration evaluated.

6.3.4.2 Bias Determination

The applicant presented the results of the USL calculation for LEU ATR, LEU MITR, and LEU MURR fuel elements in table 6.11-27 of the application. When compared to the maximum calculated k_{eff} shown for each fuel element type in table 6.11-23 of the application, there is significant margin between the conservatively determined maximum k_{eff} and the USL.
The staff has reviewed the package and concluded that the applicant's evaluations include an adequate benchmark evaluation of the calculations. The applicant identified and evaluated experiments that are relevant and appropriate for the package analyses. The applicant has determined an appropriate bias and bias uncertainties for the criticality evaluation of the package.

6.4 Evaluation Findings

Based on review of the statements and representations in the application, the NRC staff concluded that the proposed package design and contents satisfy the nuclear criticality safety requirements in 10 CFR Part 71. In making this finding, the staff considered the regulatory requirements, relevant regulatory guides, applicable codes and standards, accepted engineering practices, and independent confirmatory calculations.

6.5 References

1. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," NUREG-2216, August 2020.

2. U.S. Nuclear Regulatory Commission, "Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods," NUREG/CR-7311, April 2025.

3. F. B. Brown, et al. "User Manual for Whisper-1.1," LA-UR-17-20567, Los Alamos National Laboratory, January 2017.

7.0 PACKAGE OPERATIONS

There is no change in package operations for the package loaded with HEU fuels, RITC payload, or LEU fuels proposed in this amendment. All package operations, reviewed by the NRC in the previous application, are still applicable to this amendment.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

There is no change in acceptance tests and maintenance program for the package loaded with HEU fuels, RITC payload, or LEU fuels proposed in this amendment. All acceptance tests and maintenance program reviewed by the NRC in the previous application, are still applicable to this amendment.

In addition, based on thermal analyses of HEU fuels, RITC payload, and LEU fuels, no thermal tests are necessary to ensure continued performance of the BRR package loaded with HEU fuels, RITC payload or LEU fuels.

CONDITIONS

The following revisions were made to CoC No. 9341:

Item No. 3(a) was revised to show the new address of the applicant.

Item No. 3(b) was revised to reference the application dated August 26, 2024, as supplemented on February 13, 2025.

Condition No.5(a)(2) was revised to describe the new contents as HEU and LEU, aluminum clad plate fuel, loose fuel plates, PULSTAR reactor fuel, and TRIGA fuel of varying enrichments. Also, payload contents include isotope production targets, commercial irradiated fuel rod segments and irradiated metal. Fuel is loaded into a payload basket while irradiated metal is placed within a canister. Also, if required, loose fuel plates are contained within a loose plate box and irradiated fuel rod segments within encapsulated tubes. There are now seven baskets and one canister for irradiated metal. The RITC basket is made of 6061-T6/T651 aluminum. RITs are PWR or BWR commercial fuel rods segmented and placed within encapsulated tubes. RITC. RITC which is placed in the smaller 3.5-in. OD tubes of the RITC basket and which is used as holders for the RITs.

Condition No. 5(a)(3) was revised to show that the drawings are now referenced as ORANO drawing, and a new drawing was added 1910–01–05–SAR, “BRR Package RIT, RITC and RITC Basket SAR Drawing,” Sheets 1-3, Rev. 1

Condition No. 5(b)(1)(ii) was added to include irradiated MURR LEU Fuel Elements as authorized contents and describe them.

Condition No. 5(b)(1)(iv) was added to include the Irradiated MITR-II LEU Fuel Element as authorized contents.

Condition No. 5(b)(1)(v) was revised to clarify the cooling time after discharge from the reactor and that, although the ATR HEU fuel has 19 plates, only 18 of the plates contain fuel. A second YA fuel element design has the side plate width reduced by 15 mils.

Condition No. 5(b)(1)(vii) was added to specify the characteristics of the ATR LEU fuel element.

Condition No. 5(b)(1)(xi) was modified to clarify that up to four rods are trisected and sealed in an encapsulation tube for a total of up to 12 segments, each being up to 51 in. long. The ATR basket is used to support the 12 segments, with up to six segments placed in each of a maximum two ATR basket openings. A table, “Table 1.10. Irradiated Fuel Rod Gamma and Neutron Spectrum Per Full Length Rod in the ATR,” was added.

Condition No. 5(b)(1)(xiii) was added to account for the Irradiated and Segmented RITs. RITs consist of stainless steel pipes containing segmented fuel rods and closed off with pipe endcaps. RITs may range in length from 10.1 to 51.1 in., with fuel rod segment lengths from 8.5 to 49.5 in. per RIT. Up to 297 in. of fuel segments (6 x 49.5-in.) may be placed in up to four RITCs per RITC basket per shipment.

A new condition No. 5(b)(2)(xii) was added to specify that for the contents described in 5(b)(1)(xiii), up to 297 in. of fuel segments fits into 12 RITs.

Condition No. 6(a)(viii) was added to specify that for fuel segments in RITs, RITCs shall be used and RITC spacer pedestals may be used, both as described in section 7.1.2 of the application. Up to two RITCs shall be used within each of only the smaller openings of the RITC basket.

Condition 9 was revised to indicate that Revision No. 10 of the certificate may be used until June 11, 2025.

Condition 10 was revised to renew the CoC for an additional 5 years, with CoC expiring on June 11, 2030.

The REFERENCES section was revised to include the amendment and renewal request from Orano FS.

CONCLUSION

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the NRC staff finds that the Model No. BRR package has been adequately described and evaluated, therefore concludes that the package meets the requirements of 10 CFR Part 71. It is renewed for a 5-year term.

Issued with CoC No. 9341, Revision No. 11, dated June 11, 2025.