UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9341
Model No. BRR
Certificate of Compliance No. 9341
Revision No. 10

Summary

By letter dated November 8, 2021 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML21313A227), as supplemented on June 27, 2022 (ADAMS Accession No. ML22180A240), Orano Federal Services, LLC (Orano or the applicant) requested that the U.S. Nuclear Regulatory Commission (NRC) revise Certificate of Compliance (CoC) No. 9341 for the Model No. BEA Research Reactor (BRR) package per the details of the submitted revision of the safety analysis report (SAR or the application), Revision 18. Orano provided a fully updated SAR in response to the request for additional information.

The NRC staff (the staff) reviewed the application, as supplemented, including relevant information in the attachment to the application, using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spend Fuel and Radioactive Material," August 2020 (ADAMS Accession No. ML20234A651). Based on the statements and representations in the application, as supplemented, and the "conditions" section of this safety evaluation report (SER), the staff concludes that the package meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, "Packaging and Transportation of Radioactive Material."

The staff reviewed the performance of the package under normal conditions of transport (NCT) as required by 10 CFR 71.71 and the performance of the package under hypothetical accident conditions (HAC) as required by 10 CFR 71.73. Their analyses are included in the sections below.

1.0 GENERAL INFORMATION

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

The applicant submitted the amendment request for the BRR package, Docket No. 71-9341 to seek approval for adding two new payload types in the BRR package irradiated fuel rods and irradiated metal and includes the addition of a new canister to house irradiated metal, as shown in License Drawing 1910-01-03-SAR, Revision 8.

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

The principal components of the BRR are:

1) a lead-shielded package body,

2) a separate, removable upper shield plug,

3) a bolted closure lid,

4) upper and lower impact limiters containing polyurethane foam,

5) various payload baskets 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.

1.2 Contents

The applicant is requesting to add two new content types: (1) irradiated fuel rods and (2) irradiated metal.

1.2.1 Irradiated Fuel Rods

The irradiated fuel rod payload consists of up to the equivalent of four commercial irradiated fuel rods. Each full-length rod may contain up to 1,573.5 grams of uranium, and has an active fuel length of 152.9 inches. Cooling time is a minimum of four years. The fuel pellet diameter is 0.315 inches, and the cladding material is Zircaloy-4. Each rod is cut into three approximately equal lengths and placed in stainless steel tubes which are sealed at each end. The maximum length of an encapsulated rod segment, including the end fittings, is 51.0 inches. The encapsulated rod segments will be transported using the ATR basket. Up to six encapsulated rod segments is proposed to be placed into any single opening of the ATR basket. A removable stainless steel mesh screen will be placed in the bottom of each cavity which is used to hold the encapsulated rods. SAR Table 1.2-5 “Irradiated Fuel Rod Gamma Spectrum Per Full Length Rod” provides gamma source strength limits at given gamma energies likely in the payload. SAR Table 1.2-6 “Irradiated Fuel Rod Neutron Spectrum Per Full Length Rod” provides neutron source strength limits at given neutron energies likely in the payload.

1.2.2 Irradiated Metal

The irradiated metal payload consists of up to 350 lb of radioactive metal and may include both activation of the solid material and a layer of surface contamination. The payload is proposed to potentially include commercial reactor surveillance samples containing small dosimeter wires made using uranium and neptunium. The irradiated metal payload must be placed in the canister described in Section 1.3 of this SER. SAR Table 1.2-7 provides Irradiated Metal Gamma Source limits, the gamma source must be bounded by the energy spectrum in this table. Neutron emission for each payload must be less than 1×10^5 neutrons/sec.

1.3 Canister

The canister consists of a circular shell with a base and a bolted lid and is designed to house irradiated metal pieces. The lid is attached with four remotely-operated bolts. A 14-inch diameter circular shell forms the lower skirt of the canister. A lifting bail is welded to the lid. The lid features a vent hole and the baseplate includes a drain hole. The canister is shown in SAR Figure 1.2-21.

1.4 Drawings

Orano revised one drawing associated with the proposed changes.

1910-01-03-SAR, Sheets 1-5, Rev. 8	BRR Package Impact Limiter SAR Drawing

The change adds a canister for irradiated metal which is a cylindrical shell with flat ends which ultimately fills the payload cavity.

1.5 Evaluation Findings

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

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.

2.0 STRUCTURAL EVALUATION

The objective of this structural evaluation is to verify that the applicant has adequately evaluated the structural performance of the package (packaging together with contents) so that it meets the regulations in 10 CFR Part 71.

2.1. Structural Design

In this amendment the applicant has included two new payloads along with their criticality control structures for use with the certified BRR package. The cask in this package serves as the confinement structure. In the revised SAR, the applicant has addressed only those features of the new payload that can affect the structural behavior of the certified package.

The applicant states in SAR Section 2.1.1 that an additional basket has been added to hold the isotope production targets, and an additional canister has been added to hold the irradiated metal. Both have been added for use with the certified BRR package.

2.2. Evaluation

The weights and the center of gravity of the cask components and new payloads with their criticality control structures are shown in SAR Table 2.1-2 and compared with those of the certified package. The comparison illustrates that the weights and center of gravity locations are enveloped by those considered in the original design.

The design criteria used for the basket holding the targets and the canister for the irradiated metal is stated in SAR Section 2.1.2 as ASME, Section III Subsection NG.

In SAR section 2.1.2.1 the applicant states that the criticality control structures in the form of the basket for the targets and the canister for the irradiated metal continue to meet the allowable stresses of SAR Table 2.1-1 for the certified BRR package.

Changes in the thermal environment within the cask that can result from additional payload considered in this amendment can affect the structural design of the cask. The applicant in SAR Section 3.6.1 has addressed the thermal changes brought about by the introduction on the new payloads and in SAR Table 3.6-1. For the structural review it was concluded that as the thermal loading has not changed from the certified design loading there is no specific need for a structural analysis for a change in thermal load has shown that the thermal loads of the certificated design envelope the thermal demand of the new payloads. For the structural review it was concluded that as the thermal loading has not changed from the certified design loading there is no specific need for a structural analysis for a change in thermal load. In SER section the staff further evaluates the applicant’s assessment.

The need for any additional shielding resulting for the inclusion of the new payloads are evaluated by the applicant in SAR Section 5.6. The applicant in SAR Section 5.1.1 states that no additional shielding is required. For the structural review it was concluded that as the shielding need has not changed from the certified design there is no specific need for a structural analysis to account for a change in shielding load. In SER Section 5 the staff further evaluates the applicant’s assessment.

The applicant states that the size and the weights of the two new containers of irradiated material is within the envelope of the containers considered for the certified package. In addition, the thermal and shielding demand has not changed. Prior analysis has demonstrated that the confinement boundary provided by the cask remains intact under NTC and HAC conditions. The canister being surrounded by the cask cavity has no failure mode that can compromise the containment or shielding functions of the cask. Hence for this amendment, no additional structural evaluation was conducted.

2.3. Evaluation Findings

The staff in their structural evaluation has reviewed the information presented in the revised SAR and find reasonable assurance that the safety of the BRR package continues to be maintained with the inclusion of the two proposed payloads. The staff concludes that the BRR package continues to meet the structural requirements of 10 CFR 71.

F2-1 The staff has reviewed the package structural design description and concludes that the contents of the application satisfies the requirements of 10 CFR 71.31(a)(1) and (a)(2) as well as 10 CFR 71.33(a) and (b).

F2-2 The staff has reviewed the structural codes and standards used in package design and finds that they are acceptable and therefore satisfy the requirements of 10 CFR 71.31(c).

F2-3 The staff has reviewed the lifting and tie-down systems for the package and concludes that they satisfy the standards of 10 CFR 71.45(a) for lifting and 10 CFR 71.45(b) for tie-down.

F2-4 The staff has reviewed the package description and finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size.

F2-5 The staff reviewed the package closure description and finds that the package satisfies the requirements of 10 CFR 71.43(b) for a tamper-indicating feature.

F2-6 The staff reviewed the package closure system and the applicant's analysis for normal and accident pressure conditions and concludes that the containment system is securely closed by a positive fastening device and cannot be opened unintentionally or by a pressure that may arise within the package and therefore satisfies the requirements of 10 CFR 71.43(c) for positive closure.

F2-7 The staff reviewed the package description and finds that the package valve, the failure of which would allow radioactive contents to escape, is protected against unauthorized operation and provides an enclosure to retain any leakage and therefore satisfies the requirements of 10 CFR 71.43(e).

F2-8 The staff reviewed the application and finds that the package was evaluated by subjecting a specimen or scale model to the specific tests, or by another method of demonstration acceptable to the Commission, and therefore satisfies the requirements of 10 CFR 71.41(a).

F2-9 The staff reviewed the structural performance of the packaging under NCT required by 10 CFR 71.71 and concludes that there will be no substantial reduction in the effectiveness of the packaging that would prevent it from satisfying the requirements of

10 CFR 71.51(a)(1) for a Type B package and 10 CFR 71.55(d)(2) for a fissile material package.

F2-10 The staff reviewed the structural performance of the packaging under HAC required by 10 CFR 71.73 and concludes that the packaging has adequate structural integrity to satisfy the subcriticality, containment, and shielding requirements of 10 CFR 71.51(a)(2) for a Type B package and 10 CFR 71.55(e) for a fissile material package.

F2-12 The staff reviewed the packaging structural performance under an external pressure of 2 MPa [290 psi] for a period of not less than 1 hour and finds that the package does not buckle, collapse, or allow the inleakage of water and therefore satisfies the requirements of 10 CFR 71.61.

3.0 THERMAL EVALUATION

The amendment request seeks approval for adding two new payload types in the BRR package (1) irradiated fuel rods with heat load limit of 40 watt and (2) irradiated metal with heat load limit of 120 watt. The amendment also includes addition of a new canister to house irradiated metal, as shown on sheet 1 and sheet 5 of License Drawing 1910-01-03-SAR.

3.1. Review Objective

The objective of this review is to verify that the BRR package design, with addition of two payload types of irradiated fuel rods and irradiated metal to the list of approved contents, meets the thermal requirements of 10 CFR Part 71 under NCT and HAC.

3.2. Evaluation

3.2.1 Thermal Expansions

The applicant stated in SAR Section 2.6.1.2.2 that irradiated fuel rods are cut into sections, encapsulated in stainless steel tubes with closed ends, and transported in the Advanced Test Reactor (ATR) fuel basket. The maximum decay heat for the irradiated fuel rods encapsulated in stainless steel tubes placed in a single basket is up to 20 watt, which is only 2/3 of the maximum decay heat of 30 watt of the ATR element. In addition, the material of the encapsulation tube is stainless steel, which has a lower coefficient of thermal expansion than the aluminum material of the ATR fuel element. Thus, because the irradiated fuel rod encapsulations are the same length as the ATR fuel element, used in the same basket, and with a lower heat and lower thermal expansion, the minimum gap for the irradiated fuel rods is larger than the minimum gap calculated for the ATR fuel elements. Therefore, thermal expansion of the fuel is not of concern.

The staff reviewed the lower decay heat load of the irradiated fuel rods, the lower thermal expansion coefficient of stainless steel when compared to aluminum, and the larger minimum gap for the irradiated fuel rods, when compared to approved content of the ATR fuel, and confirmed that thermal expansion of the irradiated fuel rods is bounded by the ATR fuel when loaded in the BRR package, and therefore the thermal expansion of the irradiated fuel rods in the BRR package is not of concern.

3.2.2 Irradiated Fuel Rods under NCT and HAC

The applicant stated in SAR Section 3.1.3 that the bounding total decay heat for the irradiated fuel rods is up to 40 watt, which is 17% of the total ATR fuel element heat load of 240 watt, but only about 3.2% of the heat load (1,264 watt) of the governing BRR case (MURR) approved by the NRC in the previous application. Under NCT, the temperatures of the irradiated fuel rods loaded in the BRR package will be lower than in the case of a full load of ATR fuel elements since there will be significantly less heat generation in the package. Therefore, thermal results for the irradiated fuel rods loaded within the BRR package are not calculated.

The applicant stated in SAR Section 3.1.3 that the fire heat dominates the evolution of payload and packaging temperatures under the HAC, and the package will enter the fire with a lower initial temperature due to lower decay heat of the irradiated fuel roads. In addition, the decay heat in the irradiated fuel rods will be relatively well distributed, the rods do not contain any thermally sensitive materials, and any gases will be released when the rods are sectioned. Therefore, thermal analysis of the HAC is not necessary when transporting the irradiated fuel rods.

The staff finds that (a) the decay heat load of the irradiated fuel rods is bounded by the heat load of the MURR fuel approved by the NRC as the content of the BRR package in the previous application, and (b) the maximum fuel and package component temperatures when transporting the MURR fuel are below the allowable limits for NCT and HAC (see SAR Table 3.1-1). Therefore, the staff confirmed that the maximum fuel and package component temperatures when transporting irradiated fuel rods will be below the allowable limits for NCT and HAC, and the maximum package cavity pressures will be below the maximum normal operating pressure (MNOP) for NCT and the design pressure for HAC, respectively. Therefore, thermal analyses of NCT and HAC are not necessary when transporting the irradiated fuel rods.

3.2.3 Irradiated Metal Payload under NCT and HAC

The applicant stated in SAR Section 3.1.3 that the bounding total decay heat for the irradiated metal is 120 watt. This is only about 9.5% of the total heat load of the governing BRR case (MURR) and the maximum temperatures of the BRR package under NCT and HAC will be significantly lower than in the governing MURR case approved by the NRC in the previous application. The applicant stated that thermal results for the irradiated metal payload are not calculated and thermal analysis of the irradiated metal payload is not necessary.

The staff reviewed the material properties and configuration of the stainless canister for containing irradiated metal payload, as shown on sheet 1 and sheet 5 of License Drawing 1910-01-03-SAR, and determined that the stainless container, used to host irradiated metal payload, has no negative impact to heat removal of the BRR package under NCT and HAC.

The staff finds that (a) the decay heat load of the irradiated metal payload loaded within the stainless container is still bounded by heat load of MURR fuel, approved by the NRC as the content of the BRR package in the previous application, and (b) the maximum fuel and package component temperatures when transporting the MURR fuel are below the corresponding allowable limits of NCT and HAC (see SAR Table 3.1-1). Therefore, the staff confirmed that the maximum fuel and package component temperatures when transporting irradiated metal payload will be below the corresponding allowable limits for NCT and HAC, and the maximum package cavity pressures will be below the MNOP for NCT and the design pressure for HAC, respectively. Therefore, thermal analyses of NCT and HAC are not necessary when transporting the irradiated metal payload.

3.2.4 Vacuum Drying

The applicant performed the thermal evaluation of the vacuum drying operation, as described in SAR Section 3.3.3, to ensure that the component temperatures will remain within their acceptable temperature limits. The thermal evaluation consisted of (1) assuming air as the backfill gas during draining and vacuum drying operations, (2) disconnecting the thermal connections between the cask end surfaces and the impact limiters to simulate a bare cask within the reactor facility, (3) assuming all cask components are at equilibrium with a maximum assumed pool water temperature of 80 °F at the start of the cask draining operation, and (4) a lowest vacuum pressure of 1 torr which categorizes the gas heat transfer everywhere can be characterized as being in the viscous state and independent of the gas pressure. The applicant presented the results in SAR Table 3.3-6 and Figure 3.3-10.

The staff reviewed SAR Section 3.3.3, including SAR Table 3.3-6 and SAR Figure 3.3-10, and accepts the assumptions used in thermal evaluation of the vacuum drying operations and confirmed that all peak temperatures of the packaging components, including fuel plate and vent/drain port seals, are below their maximum allowable limits under a minimum vacuum drying pressure of 1 to 3 torr.

3.3. Evaluation Findings

Based on review of the statements and representations provided in this amendment, the staff concludes that (1) the NCT and HAC thermal evaluations have been adequately described by the applicant, (2) the addition of two payload types of irradiated fuel rods and irradiated metal to the list of approved contents of the BRR package meets the thermal requirements of 10 CFR Part 71, and (3) the component temperatures will remain within their acceptable temperature limits during the vacuum drying operations under a minimum vacuum pressure of 1 to 3 torr.

F3-1 The staff has reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71.

F3-2 The staff has reviewed the material properties and component specifications used in the thermal evaluation and concludes that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

F3-3 The staff has reviewed the methods used in the thermal evaluation and concludes that they are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.

F3-4 The staff has reviewed the accessible surface temperatures of the package as it will be prepared for shipment and concludes that they satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle.

F3-5 The staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not extend beyond the specified allowable limits during NCT consistent with the tests specified in 10 CFR 71.71.

F3-6 The staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not exceed the specified allowable short-term limits during HAC consistent with the tests specified in 10 CFR 71.73.

4.0 CONTAINMENT EVALUATION

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

5.0 SHIELDING EVALUATION

The purpose of this evaluation is to verify that the BRR package continues to provide adequate protection against direct radiation from the proposed new contents by ensuring that the package design meets the external dose rate limit requirements of 10 CFR Part 71 under NCT and HAC. This proposed revision adds irradiated fuel rods and irradiated metal to the authorized payloads. The package is designed for exclusive use. In its review, staff followed the guidance provided in Chapter 5 of NUREG-2216.

5.1 Shielding Design Description

The principal shielding feature of the BRR package consists of a lead-filled upper shield plug, lead-filled side wall, and lead-filled bottom shield plug encased in stainless-steel shells. The lid consists of a top plug with approximately 9.5 inches of lead between a 1-inch stainless steel bottom plate and 0.5-inch stainless-steel top plate. The lid is constructed of 2-inch thick stainless-steel. The lead in the side wall of the package is 8 inches thick. The package bottom consists of 7.7 inches of lead through the centerline, with a 1-inch stainless-steel bottom cover plate, and approximately 1.2-inch stainless steel inner forging.

The applicant requests approval of two types of contents: (1) irradiated fuel rod and (2) activated metal.

The proposed irradiated fuel rod contents will be transported in sealed tubes within NRC-approved ATR fuel baskets and held in place with screens. Up to 6 segments may be placed in a single ATR fuel location, with a maximum of 12 total segments per package. The maximum mass of irradiated fuel rod contents is 6,294 grams of uranium. The applicant does not rely on the basket or screens to maintain the contents position to meet the dose rate requirements of 10 CFR 71.47.

The proposed activated metal payload is limited to a maximum of 350 lbs. transported within a metal canister. The applicant describes the canister in SAR Section 1.2.1.4.7. The applicant does not rely on internal components to maintain the position of the contents to meet the dose rate requirements of 10 CFR 71.47.

5.1.1 Summary Table of Maximum Radiation Levels

Due to the weight of the package, only one package will be transported. Therefore, the dose rate limits for exclusive use established in 10 CFR 71.47(b)(1) apply. A summary of the maximum dose rates for irradiated fuel are shown in SAR Table 5.7-1 for NCT and HAC. Under NCT, the applicant calculated a maximum package surface dose rate of 145.24 mrem/hr. For the vehicle surface and 2-meter from the surface of the vehicle dose rates, the applicant calculated a maximum of 26.19 and 3.64 mrem/hr, respectively. Under HAC, the applicant calculated a maximum dose rate of 17.69 mrem/hr at 1 meter from the package surface. A summary of the maximum dose rates for irradiated metal is shown in SAR Table 5.8-1 for NCT and HAC. Under NCT, the applicant calculated a maximum package surface does rate of 108.2 mrem/hr. The applicant calculated a maximum 2-meter dose rate of 5.6 mrem/hr under HAC.

5.2 Radiation Source Specification

5.2.1 Irradiated Fuel Rod Segments

The irradiated fuel rod payload may consist of up to the equivalent of four commercial irradiated fuel rods with Zircaloy 4 cladding. Each rod must have a minimum enrichment of 4.7385% ^{235}U by weight with a total uranium content not to exceed 1573.2 grams per full-length fuel rod. The maximum burnup allowed is 73.225 MWD/MTU with a minimum cooling time of four years. Each rod segment is encapsulated in stainless steel tubes, which are sealed at each end. The active fuel length of each full-length shall not exceed 152.9 inches, and the maximum length of each encapsulated fuel rod segment, including end fittings, is 51.0 inches. Up to six encapsulated rod segments may be placed into a single ATR basket opening with a removable stainless steel mesh screen to hold the encapsulated rods.

5.2.1.1 Source Term Calculation Methods

The applicant used ORIGEN module within the SCALE 6.2.4 code to determine the source term from the predicted isotopic composition of irradiated fuel rod segments. ORIGEN is a well-vetted code with a long history of use for calculating a source spectrum from a given isotopic inventory and staff finds its use here appropriate. This also is a code recommended by NUREG-2216. The applicant used multigroup cross-section libraries based on ENDF/B-VII.1. These cross-section libraries have an established history of use with SCALE 6.2.4 and staff finds their use here appropriate. The applicant provided no details on the code nor assumptions behind the isotopic source term calculation, and instead relies on the user to ensure the isotopic inventory of irradiated fuel are bounded by the predicted contents described in SAR Section 5.7-2. The applicant selected an isotopic composition intended to bound the actual composition of irradiated fuel rods given the enrichment, burnup and cooling time parameters. Staff reviewed the sample input provided by the applicant and finds the applicant appropriately modeled the inventory to determine a correlated source spectrum. Staff conducted its own depletion analysis (see Section 5.4.5 below) to determine whether the applicant's selected isotopic inventory can reasonably be expected to be bounding. Staff also considered the margin between the expected dose rates and the regulatory limits of 10 CFR 71.47(b). As a result of these considerations, staff finds the applicant's selected isotope inventory can be reasonably met with the given enrichment, burnup, and cooling time limits specified above.

5.2.1.2 Gamma Source

The gamma rays in irradiated fuel rods will come from the decay of fission products, actinides, and secondary gamma radiation from the interaction of the neutron source term with the fuel rod and other packaging materials (e.g., stainless steel components). The applicant scaled the source strength to the equivalent of four irradiated fuel rods, which staff finds acceptable since this will increase the calculated dose rate. The applicant presented the gamma spectrum from irradiated fuel rod segments in SAR Table 5.7-3.

5.2.1.3 Neutron Source

Neutrons emitted from irradiated fuel rods will come primarily from spontaneous fission with additional neutrons induced via subcritical multiplication. The applicant used the maximum k_{safe} from its criticality evaluation to determine a subcritical multiplication factor. This k_{safe} is derived from an infinite array of homogeneous spheres of fresh fuel and water as described in SAR Chapter 6. The applicant then applied this factor to neutron and neutron induced photon fluxes. Since staff review found the applicant's k_{safe} acceptable in Section 6 of this SER, staff also finds the applicant's subcritical multiplication factor acceptable. The applicant determined a bounding neutron source spectrum for irradiated fuel rods and scaled the source term to the equivalent of four irradiated fuel rods. The applicant presented the neutron spectrum from irradiated fuel rod segments in SAR Table 5.7-4.

5.2.2 Irradiated Metal

The that irradiated metal contents will primarily consist of bulk steel material and may include both activation of the solid material and a layer of surface contamination. The contents will be transported within a metal canister described in SAR Section 1.2.1.4.7. A maximum of 350 lbs. of irradiated metal is allowed in the package and the decay heat shall not exceed 120 W.

5.2.2.1 Source Term Calculation Methods

The applicant used the ORIGEN module within the SCALE 6.2.4 code to determine the source term from the predicted isotopic composition of irradiated fuel rod segments and irradiated metal. ORIGEN is a well-vetted code with a long history of use for calculating a source spectrum from a given isotopic inventory and staff finds its use here appropriate. The applicant provided three expected isotopic inventories of contaminated and activated metal materials. The applicant provided no details on the source of this inventory nor did the applicant describe any activation calculations. The applicant relies on the user to calculate the isotopic inventory of irradiated metal and ensure that it is bounded by the predicted contents described in SAR Tables 5.8-3, 5.8-4, and 5.8-5. The applicant selected the highest quantity of each activation and contamination isotope from among the three materials and combined this into a single source term for its dose rate calculations. Since this will increase the calculated source term, staff finds this condensed source term is bounding of the provided inventories and finds the applicant's calculation acceptable. Staff conducted its confirmatory calculations with the provided source spectra and confirmed the applicant's calculated dose rates are reasonable.

The applicant expects little to no neutron radiation from irradiated metal contents. The applicant determined a minimum significant neutron source strength, in neutrons per second, that is spectrum independent by taking the maximum-value, flux-to-dose rate conversion factor and minimum distance from source to package surface with no credit for shielding materials. The flux-to-dose rate conversion factors are based on the ANSI/ANS-6.1.1-1977 values and are presented in SAR Table 5.4-1. This method is consistent with prior, NRC-approved revisions.

5.2.2.2 Gamma Source

The applicant presented the expected isotopic inventory consisting of activation and contamination materials of three materials. These inventories are shown in SAR Tables 5.8-3, 5.8-4, and 5.8-5. The applicant showed the bounding unified gamma source and spectrum in SAR Table 5.8-6. The applicant's source term includes a 50% conservative factor.

5.2.2.3 Neutron Source

The applicant did not develop a shielding model for a neutron source with irradiated metal. The applicant limited irradiated metal to a maximum neutron source strength of 1.0×10^5 neutrons per second regardless of spectrum. The applicant determined a total payload neutron source term with additional 50% conservative margin to be 6.2126×10^2 neutrons per second. This falls under the applicant's calculated minimum significant neutron source strength of 1.0×10^5 neutrons per second. As a result, staff finds the applicant's conclusion that any neutron source term from meeting the limit for this contents type will result in negligible dose acceptable.

5.3 Shielding Model and Model Specifications

The applicant's MCNP model is largely unchanged from prior revisions. The applicant maintained the same inconsistencies in the model from the actual design in this analysis, which NRC staff approved in prior revisions.

Empty space is modeled as air, and impact limiters (ILs) are modeled as void. The applicant modeled the spacing provided by the ILs under NCT. Staff finds this acceptable as the ILs are designed to remain in place under NCT and modeling the space as void will remove the shielding provided by the ILs and maximize calculated dose rates.

The applicant removed the ILs from the HAC model. Staff finds this acceptable since this assumption is consistent with the typical 30-foot drop accident under which the IL will separate from the main body of the package and will increase calculated dose rates.

5.3.1 Configuration of Source and Shielding

The applicant proposed no changes to the packaging design or materials, and the only changes are to contents and the associated source terms. The applicant omitted all components and support structures from the package cavity in its evaluations. Since this material would provide additional shielding and reduce calculated dose rates, staff finds the removal acceptable. Outside of the package cavity, the applicant maintained all the same modeling simplifications and assumptions as prior, NRC-approved revisions. For NCT dose rates, the applicant credits the spacing provided by the ILs but models them as void. Staff finds this acceptable as the ILs credibly remain in place and modeling as void omits any shielding provided by the ILs and increases the calculated dose rates. Under HAC, the applicant removed the ILs from its models.

The applicant modeled the irradiated fuel rod segments as two, optimally packed bundles of 6 rod segments located as near the top vent port as possible. The volume of each cylinder is the equivalent of six times the volume of a single rod segment. The length of the cylinders modeled is 48 inches, which is less than the maximum length of the encapsulated rods. The concentration of the source volume and the placement close to the potential streaming path will increase the calculated dose rates. The applicant modeled the fuel rods as void. This ignores self-shielding, which will increase the calculated external dose rates. For these reasons, staff finds the applicant's irradiated fuel rod segment source configuration acceptable.

The applicant modeled the irradiated metal as cylinders of carbon steel with a volume that gives an effective density of 7.75 g/cm^3. The applicant increased the source term in the shielding calculation by 50% above the activation calculation which will conservatively increase the calculated dose rate. The applicant modeled the cylinder as close to the tallies being used to calculate external dose rates. In the case of the top and the bottom, the applicant modeled the source as a relatively flat cylinder that filled the package cavity at the top and bottom, respectively. For the side dose rates, the applicant modeled the irradiated metal source as a long cylinder shifted toward the side of the package cavity and aligned over the drain port. Staff finds the configuration of the irradiated metal source acceptable as it moves the source as close as possible to the location where the dose is being evaluated, which will increase calculated dose rates and ensure the dose rate calculations to be conservative.

5.3.2 Material Properties

The applicant proposed no changes to the properties of the packaging materials.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant used MCNP 6.2 to calculate the dose rates for irradiated fuel rod segments. The applicant used the default cross-section libraries based on ENDF/B-VI.8 for photons, and ENDF/B-VII.1 for neutrons. MCNP is a three-dimensional, Monte Carlo code which, along with its cross-section libraries, have a long history of use in fixed-source shielding applications, and the staff finds their use appropriate.

For irradiated metal, the applicant calculated the dose rates at the top, bottom, and side of the package with MCNP 6.2 and ENDF/B-VI.8 photon cross-section libraries. The applicant included a two-inch offset at the surface of the top shield plug for the top- and side-shifted source location. The irradiated metal payload will be contained within its own canister and the applicant showed this canister credibly maintains this spacing under NCT and HAC. Staff calculations confirmed the applicant’s position that this offset precludes direct streaming through the vent port.

5.4.2 Code Input and Output Data

Applicant provided sample ORIGEN and MCNP input for both the irradiated rod segments and irradiated metal in Sections 5.7.5 and 5.8.5, respectively. Staff evaluated the sample input and finds the applicant appropriately modeled the geometry, materials, and source specifications.

5.4.3 Dose Rate Conversion Factors

Applicant used ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors. Staff finds this acceptable since it follows the guidance of NUREG-2216.

5.4.4 External Radiation Levels

The applicant presented the maximum dose rates from irradiated fuel rod segments in SAR Table 5.7-1. The applicant presented the maximum dose rates from irradiated metal in SAR Tables 5.8-1 and 5.8-2. The applicant calculated the dose rates at the following locations: the cask side surface, the top surface of the top IL, the bottom surface of the bottom IL, the IL side surfaces, and the IL “underside” surfaces. The applicant considered the IL “underside” surfaces as part of the package side surface. The applicant used two segmented surface tallies on the conical surface of the top and bottom surfaces of the packages. The applicant considered the top and bottom surfaces of the package to be the same as the vehicle top and bottom for dose rate calculations. The applicant calculated vehicle side surface dose rates with one mesh tally at the projected transport vehicle side surface with a cylindrical tally 4 feet from cask centerline. The applicant calculated 2-meter dose rates with one mesh tally 2 meters from the vehicle projected side surface. For 1 meter dose rates under HAC, the applicant calculated the dose rates with three mesh tallies 1 meter from the cask side, 1 meter from the cask top, and 1 meter from the cask bottom. Staff reviewed the applicant's tally locations and spatial segmentation and determined they are sufficient to determine the location of maximum dose rate around the package to a reasonable degree of accuracy.

5.4.5 Confirmatory Analysis

To evaluate the applicant's isotope contents, staff modeled a single fuel pin with the same diameter and enrichment as the intended contents. Staff used TRITON in SCALE 6.2.3 with the 252 multi-group cross sections based on ENDF/B-VII data. TRITON is a two-dimensional discrete ordinates flux solver that applies the flux solution through time steps to calculate material depletion. Staff implemented reflective boundary conditions to model an infinite array of fuel pins. Staff depleted the fuel to 75 GWd/MTU in a single "cycle" and then cooled for 4 years. The staff's depletion calculation tracked all the nuclides available for depletion in SCALE 6.2.3. Staff sorted the nuclides from SAR Table 5.7-2 by activity and decay. Using NUREG/CR-6701 as a reference, staff further narrowed down the list of nuclides based on likely contribution to external dose. Staff results showed that the dose rate from the source term calculated from isotope quantities in SAR Table 5.7-2 can reasonably be expected to bound those of Zircaloy 4 clad fuel rods with the burnup, enrichment, and cooling time limits specified in Section 5.2.1 above.

Staff created a model of the BRR package using the Monaco/MAVRIC module within SCALE 6.2.3. MAVRIC applies Denovo, a three-dimensional, discrete-ordinates solver, which uses multi-group cross-sections and a spatial mesh to calculate adjoint flux as a function of position and energy. The code uses those calculated fluxes to determine an importance map for weight windows and a biased source for variance reduction with Monaco. Monaco is a 3-dimensional, Monte Carlo radiation transport code used to calculate dose rates from a fixed input source. Staff used continuous energy neutron and photon cross-section libraries based on ENDF/B-VII nuclear data.

Staff's model consisted of the steel and lead components of the side wall and upper and lower shield plugs and omitted the ILs. Staff modeled the drain in the lower end structure and the vent in the upper shield plug to evaluate potential radiation streaming paths. Staff evaluated the positioning of the spent fuel rods and irradiated metal around the upper shield plug vent and the lower shield plug drain. Staff results confirm the applicant's positioning of the source in its shielding analyses with irradiate metal and irradiated fuel rods results in the maximum calculated dose rates. Staff used mesh tallies and subsequent point detectors to evaluate any potential location of maximum dose not considered by the applicant. Staff results did not show peak dose rates at a location that differed significantly from the applicant's results. Therefore, staff finds the applicant acceptably determined the maximum dose locations.

5.5 Evaluation Findings

Based on its review of the information and representations provided in the application and the staff's independent, confirmatory calculations, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements, and the radiation level limits in 10 CFR Part 71. The staff also considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices, in reaching the following findings:

F5-1 The staff has reviewed the application and finds that it adequately describes the package contents in compliance 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 evaluation is appropriate and bounding for the packaging and the package contents as described in the application.

F5-2 The staff has reviewed the 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.

F5-3 The staff has reviewed the 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).

6.0 CRITICALITY EVALUATION

The objective of this review is to verify that the BRR spent fuel transportation package design containing the new contents as specified in Chapter 1 of the SAR meets the criticality safety requirements of 10 CFR Part 71 under NCT and HAC, respectively. The staff reviewed the description of the package design and criticality safety analyses presented in SAR Chapters 1 and 6 for the package. The staff performed its review following the guidance provided in NUREG-2216. The following sections of this report document the staff's criticality safety evaluation for this package design.

The applicant requested the addition of two new materials as authorized contents in the BRR package, irradiated commercial nuclear fuel and irradiated metal. The irradiated metal is non-fissile, and as such will not be discussed as part of the criticality safety review. The irradiated commercial nuclear fuel is fissile and will be the focus of this safety evaluation.

6.1 Package Design

6.1.1 Description of Criticality Safety Design

The BRR packaging system is a cylindrical lead lined cask designed for transportation of Type B quantity radioactive materials and/or solid fissile materials. The packaging consists of a payload basket, a lead–shielded cask body, an upper shield plug, a closure lid, and upper and lower impact limiters. The package is of conventional design and utilizes ASTM Type 304 stainless steel as its primary structural material. The package is designed to provide leak tight containment of the radioactive contents under normal conditions of transport.

6.1.2 Fissile Material Contents

The irradiated commercial nuclear fuel rods contain 1573.5 grams of uranium per rod enriched to 4.7385 wt% ^{235}U, with an active fuel length of 152.9 inches and a fuel pellet diameter of 0.315 inches. Each fuel rod is cut into thirds, not to exceed 51 inches in length per cut length and encapsulated in a tube to prevent material loss. A maximum of six fuel segments are allowed in an ATR basket.

The applicant evaluated post-burnup fissile material inventory in the fuel rods and based on a comparison with the fresh fuel fissile material inventory, as show in SAR Tables 6.10-2 and 6.10-3, and has elected to use the fresh fuel assumption as the basis for their modeling. This is a conservative assumption that the staff finds acceptable.

6.2 General Considerations

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

6.2.1 Model Configuration

As outlined in the SAR Table 6.10.4, the applicant identified key model dimensions for the BRR package loaded with the 304 stainless steel ATR basket containing the irradiated fuel rod sections. Models were created by the applicant for a single package as well as arrays of packages. The applicant based its criticality models upon the shielding models and utilized a model that differs slightly from the drawings in a conservative way, most notably in the diameter of the lead at the bottom of the BRR, which is modeled as 9.75 inches as opposed to the actual diameter of 10.25 inches. This is a conservative assumption with regards to the multiplication factor and increases neutron transmission between packages in an array. For the criticality analysis the impact limiters are neglected in both the single package and array configurations. Staff finds this acceptable since it eliminates those absorption materials from the models as well as allowing for closer packing of packages in the array evaluation.

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

6.2.2 Computer Codes and Cross-Section Libraries

The applicant used MCNP6.2 in its criticality safety analyses for the BRR package, utilizing continuous energy neutron cross-sections based on the ENDF/B-VII libraries. All cases were run using 300 generations with the first 25 generations skipped for calculation of k_{eff} and 5,000 neutrons per generation to ensure proper convergence of the modeled systems. The staff reviewed the applicant's selection of computer code and cross sections. The staff finds that the MCNP computer code is a widely used and rigorously validated computer code for criticality safety analyses and is adequate for this application. The cross sections selected encompassed the energy and temperature ranges of the system. On this basis, the staff determined that the computer code and cross sections selected are acceptable and adequate for criticality safety analyses of this application.

6.3 Demonstration of Maximum Reactivity

Utilizing the fresh fuel assumption, with a minimum enrichment of 4.7385 wt% 235U at the maximal mass of uranium contained in four rods (i.e., the maximum payload configuration), the applicant made several very conservative assumptions, including: no cladding on the fuel rods to eliminate the parasitic absorption of neutrons from the system; structural components of the fuel and the ATR basket are excluded; and any resulting volumes that are omitted are filled with water. The applicant assumed the tubes containing the fuel rods fail and the resultant fissile mass forms a homogeneous sphere of uranium and water mixture, maximizing the reactivity of the uranium mass. Staff finds this acceptable because these assumptions provide the most reactive configuration possible of the fuel mixture.

For the single package case, the applicant modeled the package as reflected by 12 inches of water. For the NCT and HAC cases, the applicant modeled the packages as a close packed hexagonal lattice, which minimizes the distance between packages in the array, thereby yielding the highest k_{eff} for the system. The applicant did not request air transport of this package configuration and is not modeled as part of this evaluation.

6.4 NCT Single Package Evaluation

The NCT configuration for a single package maintains the rod segments contained in intact encapsulation tubes. Since this configuration is much less reactive than the HAC conditions of failed fuel rods and material in an optimum geometry configuration, the NCT case is bounded by the HAC case and is not modeled as part of this application.

6.5 HAC Single Package Evaluation

As described above, the applicant assumes that the encapsulation tubes have failed and that the fissile material forms a homogeneous sphere of fresh fuel and water under HAC. The applicant performed multiple parametric studies on different configuration of the fissile mass to determine the most reactive configuration. Fuel mixture radius was varied while holding the total uranium mass constant and optimum moderation was determined as illustrated in SAR Figure 6.10-3 and shown in SAR Table 6.10-9. The applicant identified the most reactive conditions of the homogeneous sphere as a 12.5 cm sphere at a uranium to water mixture density of 1.729 g/cm3 reflected by 12 inches of full density water. This resulted in a k_{eff} = 0.68751, which is substantially below the USL of 0.92097. Therefore, staff finds reasonable assurance that the NCT and HAC single packages will remain subcritical.

6.6 HAC Package Array Evaluation

Building upon the single package analysis, the applicant assumes an infinite, tightly packed hexagonal array of the most reactive packages determined in SER Section 6.5 above with variable density water between packages. The cavity of the ATR basket is filled with a spherical uranium/water mixture centered in the cavity, with the remainder of the cavity filled with full density water. In order to determine the most reactive configuration of packages, the applicant performed a parametric study of variable density water between the array lattices from void to full density water as shown in SAR Table 6.10-10. Staff finds that this yielded the most reactive configuration possible for the array evaluation.

Based on this analysis, the largest multiplication factor was found to be at a water density of 0.8 g/cm3, yielding a maximum k_{eff} = 0.68788, which is substantially lower than the applicant's USL. Therefore, staff finds reasonable assurance that the loaded BRR with commercial irradiated fuel rod sections will remain subcritical during HAC.

6.7 Confirmatory Analysis

Staff performed a confirmatory analysis on the most reactive HAC case identified by the applicant using the SCALE 6.3 computer code system, with the KENO VI three-dimensional Monte Carlo code and the continuous-energy ENDF/B-VII cross section library. The staff's confirmatory analysis confirmed the applicant's criticality analysis results and conclusion that the package remains subcritical with sufficient safety margin under hypothetical accident conditions and continues to meet the requirements of 10 CFR Part 71.

6.8 Benchmark Evaluations

The applicant selected critical experiments to bound a range of parameters covered by the criticality design and contents of the BRR package. The applicant analyzed the k_{eff} results for trends and determined the maximum bias and bias uncertainty for the BRR criticality safety analyses consistent with recommendations of NUREG-2216. These results are presented in SAR Section 6.10.8. Staff finds that the benchmark evaluations and bias determination are for the types of fuel to be transported under this amendment to the BRR package are representative and appropriate and demonstrate a conservative bias for their calculations.

6.9 Evaluation Findings

Based on the information presented in the amended SAR and the staff's independent calculations, staff found with reasonable assurance that the package with the new content of irradiated commercial fuel rod segments as allowable contents in the BRR package meets the regulatory requirements of 10 CFR Part 71.

F6-1 The staff has reviewed the package and concludes that the application adequately describes the package contents and the package design features that affect nuclear criticality safety in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b) and provides an appropriate and bounding evaluation of the package's criticality safety performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a).

F6-2 The staff has reviewed the package and concludes that the application identifies the codes and standards used in the package's criticality safety design in compliance with 10 CFR 71.31(c).

F6-3 The staff has reviewed the package 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). The applicant specifies an appropriate CSI for each type of fissile content.

F6-4 The staff has reviewed the package and concludes that the applicant used packaging features and package contents configurations and materials properties in the criticality safety analyses that are consistent with and bounding for the package's design basis, including the effects of the NCT and the relevant accident conditions in 10 CFR 71.55(f). The applicant has adequately identified the package configurations and material properties that result in the maximum reactivity for the single package and package array analyses.

F6-5 The staff has reviewed the package and concludes that the criticality evaluations in the application of a single package demonstrate that it is subcritical under the most reactive credible conditions, in compliance with 10 CFR 71.55(b), 71.55(d), and 71.55(e). The evaluations in the application also demonstrate that the effects of the normal conditions of transport tests do not result in a significant reduction in the packaging's effectiveness in terms of criticality safety, in compliance with 10 CFR 71.43(f) and 10 CFR 71.55(d)(4) and, for Type B fissile packages, 10 CFR 71.51(a)(1). The evaluations in the application also demonstrate that the geometric form of the contents is not substantially altered under the normal conditions of transport tests, in compliance with 10 CFR 71.55(d)(2).

F6-6 The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 5N undamaged packages demonstrates that the array of 5N packages is subcritical under NCT to meet the requirements in 10 CFR 71.59(a)(1).

F6-7 The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 2N packages subjected to the tests in 10 CFR 71.73 demonstrates that the array of 2N packages is subcritical under HAC in 10 CFR 71.73 to meet the requirements in 10 CFR 71.59(a)(2).

F6-8 The staff has reviewed the package 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 and performed appropriate trending analyses of the benchmark calculation results. The applicant has determined an appropriate bias and bias uncertainties for the criticality evaluation of the package.

F6-9 The staff has reviewed the package and concludes that the application identifies the necessary special controls and precautions for transport, loading, unloading, and handling and, in case of accidents, compliance with 10 CFR 71.35(c). These controls include additional contents specifications (e.g., fuel loading curve(s), reactor operating parameters) and administrative procedures to prevent package misloads.

F6-10 The staff has reviewed the package and concludes that the evaluations in the application assume unknown properties of the fissile contents are at credible values that maximize neutron multiplication consistent with 10 CFR 71.83. This includes following the recommendations in Section 6.4.7 and Attachment 6A to this SRP chapter for crediting the burnup of the SNF contents.

7.0 MATERIALS EVALUATION

The purpose of the staff's materials evaluation is to assess whether the applicant adequately described and evaluated the materials for the Orano BAE Research Reactor (BRR) Package to ensure compliance with the applicable requirements of 10 CFR Part 71. The NRC staff performed its materials evaluation by following the technical guidance in NUREG-2216.

The amendment application for the BRR package includes the addition of new irradiated contents, including commercial irradiated fuel rods and irradiated metal pieces. The amendment also adds a new packaging component, an internal stainless steel canister (located inside the BRR cask in lieu of a fuel basket) that houses the irradiated metal pieces. Other than the addition of the stainless steel canister for housing the irradiated metal pieces, there are no changes to the BRR packaging components. Therefore, the NRC staff performed its materials review for the amendment application by evaluating the stainless steel canister materials and assessing the impact of the new irradiated fuel rod and irradiated metal contents on the BRR packaging component materials.

7.1 Drawings

The BRR package design drawings are provided in SAR Section 1.3.3. Drawing No. 1910 01-03-SAR, “BRR Package Fuel Baskets SAR Drawing,” is updated to include the stainless steel canister that houses the irradiated metal contents. The drawing includes new sheets depicting canister component geometries, assemblies, dimensions, tolerances, component material specifications, welds, and non-destructive examination (NDE) methods. The staff reviewed these changes and confirmed the updated drawing adequately specifies the materials, welds, and NDE methods for the new stainless steel canister. Therefore, the staff finds that the updates to the BRR package drawings to include the stainless steel canister are acceptable.

7.2 Codes and Standards

SAR Section 2.1.2 states that the new stainless steel canister is classified as ASME Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NG. The staff determined that the use of Subsection NG for the construction of the canister is appropriate since NG is generally recommended for construction of fuel baskets, which are relied on for subcriticality, whereas the new canister does not perform any function related to criticality safety. The material specifications for the BRR package are listed in the bill of materials of the design drawings in SAR Section 1.3.3. Drawing No. 1910-01-03-SAR includes the material specifications for the new stainless steel canister. The staff reviewed the stainless steel canister base material and noted that it is the same as the base material specified for the BRR cask and lid assembly, which form the structural containment boundary. The staff verified that the ASTM standard specification for the canister stainless steel base material, ASTM A240, Type 304 plate, has properties that are suitable for protecting the BRR cask interior during NCT and HAC. Drawing No. 1910-01-03-SAR also specifies generic “stainless steel” for the canister closure lid bolting material. The staff noted that the material information in the drawing does not include any consensus standard specification (such as an ASTM or ASME BPVC Section II standard specification) for the stainless steel bolting. The staff determined that the specification of “stainless steel” for the closure lid bolting material is reasonable since its safety function is to support the weight of the loaded canister during canister lifting operations, and stainless steel bolting is generally not susceptible to brittle fracture. The staff’s evaluation of the tensile properties of the stainless steel bolting for performing its structural support function is addressed below in Section 7.4 of this SER. Based on these considerations, the staff determined that the use of stainless steel canister materials that are the same as those used for the containment boundary and fuel baskets is conservative in light of the more limited safety function of the canister. Therefore, the staff determined that the codes and standards that are specified for the stainless steel canister materials are acceptable.

7.3 Weld Design and Inspection

SAR Chapters 1, 2, 4, and 8 include information concerning the design, fabrication, qualification, and NDE of the stainless steel canister welds. The staff reviewed this information and verified that the application specifies that the stainless steel canister welds must meet the applicable requirements of the ASME BPVC Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” ASME BPVC Section IX, “Welding, Brazing, and Fusing Qualifications,” and American Welding Society (AWS) D1.6, “Structural Welding Code—Stainless Steel”. The staff noted that the code requirements for the canister welds are the same as those that are specified for the internal fuel basket welds, which are relied upon in the structural and criticality evaluations to maintain the fissile material contents in the analyzed subcritical configuration.

Since the irradiated metal pieces are not fissile material, criticality analysis is not performed for irradiated metal content. The canister is designed to house the irradiated metal pieces and protect the interior of the BRR cask containment boundary from structural damage during NCT and HAC, but as previously noted, the canister is not relied on for containment or criticality control. Based on these considerations, the staff determined that the use of the same welding code requirements that are applied for the fuel baskets is conservative in light of its more limited safety function. Therefore, the staff determined that the codes and standards that are specified in the application for the canister welds are acceptable.

7.4 Mechanical Properties of Materials

Other than the addition of the stainless steel canister (located inside the BRR cask) for housing the irradiated metal, there are no changes to the BRR packaging components. The staff reviewed the mechanical properties of the canister base material, including the design stress intensity, yield strength, ultimate tensile strength, Young's modulus, Poisson's ratio, and mean coefficient of thermal expansion, and confirmed the properties are consist with those specified in the ASME Code, Section II, Part D for ASTM A240, Type 304 stainless steel plate. For the canister closure lid bolting, the staff reviewed the range of tensile properties for generic stainless steel bolting and confirmed that the four closure lid bolts have adequate tensile strength to support the full weight of the loaded canister with adequate safety margin. The staff also verified that the new commercial irradiated fuel rods and irradiated metal contents would have no adverse effect on the mechanical properties of the existing BRR packaging materials. Therefore, the staff determined that the mechanical properties of the packaging materials are acceptable for use in the structural evaluation of the BRR package.

7.5 Thermal Properties of Materials

For the new stainless steel canister, the staff reviewed the thermal properties, including the thermal conductivity and specific heat capacity, and confirmed the properties are consistent with those specified for Type 304 stainless steel in the ASME Code, Section II, Part D. The staff also verified that the new payloads would have no adverse effect on the thermal properties of the existing BRR packaging materials. Therefore, the staff determined that the thermal properties of the packaging materials are acceptable for use in the thermal evaluation of the BRR package.

7.6 Chemical and Galvanic Reactions

SAR Section 2.2 provides an evaluation of the potential for chemical, galvanic, and corrosive reactions of package components and contents. This section demonstrates that the materials of construction of the BRR package will not undergo significant chemical, galvanic, or corrosive reactions. The staff reviewed the potential impacts of the proposed new payloads and noted that the commercial irradiated fuel rods are sectioned into thirds and encapsulated in unreactive stainless steel tubes. The tubes containing the irradiated fuel rod segments are carried in the Advanced Test Reactor (ATR) fuel basket, which is contained inside the BRR cask. The irradiated metal payload is carried inside the internal stainless steel canister, and the canister is contained in the BRR cask. The BRR cask is designed and fabricated as a leak tight containment boundary, which maintains a chemically inert environment. The entire BRR package design ensures that physical contact occurs only between unreactive materials over the range of analyzed conditions for NCT and HAC. Other than the addition of the stainless steel canister for carrying the irradiated metal pieces, there are no changes to the design of the BRR packaging components. Therefore, the staff determined that the addition of the new payloads for the allowable contents will not have an adverse effect on the potential for chemical, galvanic, or corrosive reactions of package components and contents.

7.7 Radiation Shielding

SAR Sections 5.7 and 5.8 provide the radiation shielding evaluation of the commercial irradiated fuel rods and the irradiated metal payload that are proposed as allowable contents for the BRR package.

The commercial irradiated fuel rods are sectioned into thirds, and the segments are encapsulated in stainless steel tubes to prevent material loss. The encapsulated fuel rods segments are carried in the Advanced Test Reactor (ATR) fuel basket located inside the BRR cask. There are no changes to the design or construction of the ATR fuel basket, the BRR cask, the lead shielding of the cask, or any other BRR packaging components. SAR Section 5.7 evaluates the radiation shielding performance of the packaging components to ensure that the external radiation requirements of 10 CFR 71 will be satisfied for the proposed irradiated fuel rod contents. The staff verified that all packaging materials that are credited for radiation shielding are not prone to degradation or loss of structural integrity that would compromise their shielding performance, as demonstrated in the structural and thermal evaluations for NCT and HAC. Therefore, the staff determined that the material properties of the packaging components are acceptable for the shielding evaluation of the commercial irradiated fuel rods.

The irradiated metal payload is contained in a welded stainless steel canister located inside the BRR cask. The internal stainless steel canister is not credited in the radiation shielding analysis. The analysis of the irradiated metal payload relies only on the BRR cask for radiation shielding. SAR Section 5.8 evaluates the radiation shielding performance of the BRR cask to ensure the external radiation requirements of 10 CFR 71 will be satisfied for the proposed irradiated metal content. The staff verified that all packaging materials that are credited for radiation shielding are not prone to degradation or loss of structural integrity that would compromise their shielding performance, as demonstrated in the structural and thermal evaluation for NCT and HAC. Therefore, the staff determined that the material properties of the packaging components are acceptable for the shielding evaluation of the irradiated metal payload.

7.8 Criticality Safety

SAR Section 6.10 provides the criticality evaluation of the commercial irradiated fuel rods that are proposed as allowable contents for the BRR package. As with the other fissile material contents previously authorized for transport inside the BRR package, no dedicated neutron absorbers are used in the BRR package. The application identifies that the limitation on the quantity of fissile material contained in the package and the spatial arrangement of the fuel rods in the ATR fuel basket are sufficient to maintain the required subcritical assembly, as analyzed in Section 6.10. There are no changes to the design or construction of the ATR fuel basket or any other BRR packaging components. The staff confirmed that the structural materials used in the ATR basket are suitable for ensuring that the fuel rods are kept in a subcritical configuration based on the structural and thermal evaluations of the BRR package demonstrating acceptable package performance for NCT and HAC. Therefore, the staff determined that the material properties of the packaging components are acceptable for the criticality evaluation of the commercial irradiated fuel rods.

The application does not include a criticality evaluation of the irradiated metal payload. The staff verified that criticality requirements are not applicable to the irradiated metal payload since the irradiated metal is not fissile material.

7.9 Evaluation Findings

Based on review of the statements and representations in the application, the NRC staff concludes that the materials used in the transportation package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

F7-1 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff's evaluation.

F7-2 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.31(c). The applicant identified the applicable codes and standards for the design, fabrication, testing, and maintenance of the package and, in the absence of codes and standards, has adequately described controls for material qualification and fabrication.

F7-3 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a). The applicant demonstrated effective materials performance of packaging components under normal conditions of transport and hypothetical accident conditions.

F7-5 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(d). The applicant has demonstrated that there will be no significant corrosion, chemical reactions, or radiation effects that could impair the effectiveness of the packaging.

F7-6 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a) for Type B packages and 10 CFR 71.55(d)(2) for fissile packages. The applicant has demonstrated that the package will be designed and constructed such that the analyzed geometric form of its contents will not be substantially altered and there will be no loss or dispersal of the contents under the tests for normal conditions of transport.

8.0 PACKAGE OPERATIONS EVALUATION

The purpose of the package operations evaluation is to verify that the proposed changes to the operating controls and procedures of the transport package continue to meet the requirements of 10 CFR Part 71.

SAR Chapter 7 provides procedures for package loading, unloading, and preparation of the empty package for transport. SAR Section 7.1.2.2 provides revised operating procedures for loading the dry loading of irradiated fuel rods and irradiated metal into the BRR package.

The staff reviewed the Operating Procedures in SAR Chapter 7 to verify that the package will be operated in a manner that is consistent with its design evaluation. Based on its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the proposed dry loading of irradiated fuel rods and irradiated metal in accordance with 10 CFR Part 71. Further, the CoC is conditioned such that the package must be prepared for shipment and operated in accordance with the Operating Procedures specified in SAR Chapter 7.

F8-1 The NRC staff has reviewed the proposed special controls and precautions for transport, loading, unloading, and handling and [if needed] the proposed special controls in case of accident or delay, and finds that they satisfy 10 CFR 71.35(c).

F8-2 The NRC staff has reviewed the description of the operating procedures and finds that the package will be prepared, loaded, transported, received, and unloaded in a manner consistent with its design and evaluation for approval.

F8-3 The NRC staff has reviewed the description of the special instructions needed to safely open a package and concludes that the procedures for providing the special instruction to the consignee are in accordance with the requirements of 10 CFR 71.89.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

There were no changes that affected the package's acceptance tests and maintenance program evaluation.

10.0 CONDITIONS

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

Condition No. 3.(b), "Title And Identification Of Report Or Application," was updated to include reference to the most recent application, which included Revision 18 of the SAR.

Condition No. 5.(a)(2), "Description," was updated to include descriptions of the canister assembly show in Drawing 1910–01–03–SAR.

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

1910–01–03–SAR, Sheets 1-5, Rev. 8	BRR Package Fuel Baskets SAR Drawing

Condition No. 5.(b)(1), "Type and form of material," (viii) and (ix) were added to describe the new irradiated fuel rods and the new irradiated metal pieces payloads, respectively.

Condition No. 5.(b)(2), "Maximum quantity of material per package," (x) and (xi) were added to state the maximum quantities of the irradiated fuel rods and irradiated metal pieces payloads, respectively.

Condition No. 6.(a), requirements in addition to 10 CFR Part 71 Subpart G, (vi) and (vii) were added to include details of shipment preparations and operations for the new payloads.

Condition 9 was revised to authorize use of revision No. 9 of the certificate until October 31, 2023.

Revised the "References" section of the CoC to incorporate the application prompting this review.

11.0 CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated, and the Model No. BRR package meets the requirements of 10 CFR Part 71. In the consolidated SAR, Orano incorporated all supplements previously approved by NRC.

Issued with CoC No. 9341 for the Model No. BRR, Revision No. 10.