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

August 8, 2017

Mr. Scott P. Murray
Manager, Facility Licensing
Global Nuclear Fuel – Americas, LLC
3901 Castle Hayne Rd.
P.O. Box 780
Wilmington, NC 28402

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9309, REVISION NO. 11, FOR THE MODEL NO. RAJ-II TRANSPORTATION PACKAGE

Dear Mr. Murray:

As requested by your application dated September 30, 2016, [Agencywide Documents Access and Management System (ADAMS) Accession Number (No.) ML16274A097] and as supplemented on November 28, 2016, (ADAMS Accession No. ML16333A225) and April 7, 2017 (ADAMS Accession No. ML17097A102) enclosed is Certificate of Compliance No. 9309, Revision No. 11, for the Model No. RAJ-II transportation package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's safety evaluation report is also enclosed.

The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of Title 49 of the *Code of Federal Regulations* (49 CFR) 173.471. Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471.

Upon removal of Enclosure 3, this document is uncontrolled.

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

Sincerely,

/RA/

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

Docket No. 71-9309
CAC No. L25154

Enclosures:
1. Certificate of Compliance
   No. 9309, Rev. No. 11
2. Safety Evaluation Report
3. Registered Users

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

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9309, REVISION NO. 11, FOR THE MODEL NO. RAJ-II TRANSPORTATION PACKAGE, DOCUMENT DATE: AUGUST 8, 2017.

Distribution: DSFM r/f NMSS r/f

http://fusion.nrc.gov/nmss/team/sfst/sfst-licensing/10_cfr_71/71_raj-ii/Shared Documents/9309.r11.LTR_and_SER.doc
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Closes CAC No. L25154.
ADAMS Package No.: ML17222A008 LTR&SER: ML1722A010 Enclosure 1: CoC: ML17222A011 Enclosure 3: Registered Users: ML17222A012

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SAFETY EVALUATION REPORT
Docket No. 71-9309
Model No. RAJ-II
Certificate of Compliance No. 9309
Revision No. 11

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SAFETY EVALUATION REPORT
Docket No. 71-9309
Model No. RAJ-II
Certificate of Compliance No. 9309
Revision No. 11

SUMMARY

By letter dated September 30, 2016, (GNF 2016a) and as supplemented on November 28, 2016, (GNF 2016b) and April 7, 2017 (GNF 2017), Global Nuclear Fuel – Americas, LLC (GNF-A), requested an amendment to Certificate of Compliance (CoC) No. 9309, for the Model No. RAJ-II transportation package. GNF-A provided an application (safety analysis report (SAR)) to support the proposed changes to the content parameters related the 8 × 8, 9 × 9, GNF3 10 × 10, and 10 × 10 fuel assemblies.

The U.S. Nuclear Regulatory Commission (NRC) staff (the staff) reviewed the application, including relevant supplemental information to determine if the package still meets the regulatory requirements of Title10 of the *Code of Federal Regulation* (10 CFR) Part 71. The staff used the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," for its review. Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concludes that the package meets the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging Description

The applicant did not propose changes to the packaging design. The Model No. RAJ-II is a rectangular box used to transport a maximum of two fuel assemblies or a various number of individual rods. The package is used to transport unirradiated boiling water reactor (BWR) fuel assemblies, BWR fuel rods, CANDU fuel rods, and pressurized water reactor (PWR) fuel rods, containing Type B fissile material. The package consists of a stainless steel inner and outer container. The inner container includes polyethylene cushioning material that provides protection to the fuel assembly. Wood and honeycomb resin impregnated kraft paper serve as shock absorbers.

The applicant proposed a revision to the description of the containment boundary for the package, which it would incorporate in the application. Previously, the containment boundary description just mentioned the fuel rod clad and ceramic nature of the fuel pellets. The applicant updated the description of their containment boundary as follows:

> "[t]he fuel rod cladding and welded end plugs provide the primary containment of the radioactive material. The radioactive material is bound in ceramic pellets with very limited solubility and minimal propensity to suspend in air."

The staff discusses this revision in Chapter 4, "Containment Evaluation," of this safety evaluation report (SER).

The applicant provided a shielding analysis for completeness, and it is discussed in Chapter 5, "Shielding Evaluation," of this SER. The shielding analysis did not result in additional shielding features that should be added to the package design.

The applicant proposed changes to the tables in the application and CoC. The applicant revised Table 1-1, "Weights and Outer Dimensions of the Package," of the application to include the following:

a. the "loose rods pipe nominal mass per component" of 106 kilograms (kg) (234 pounds (lbs.)),

b. the "protective case nominal mass per component" of 87 kg (192 lbs.), and

c. the maximum gross shipping weight of 1,614 kg (3,558 lbs.).

The maximum gross weight did not change from previous revisions of the CoC.

1.2 Contents

The RAJ-II package is used to transport unirradiated BWR fuel assemblies, BWR fuel rods, CANDU fuel rods, and PWR fuel rods, containing Type B fissile material. The fissile material can be in the form of uranium dioxide or uranium carbide enriched up to 5.0 weight percent (wt.%) ^{235}U.

The applicant revised the application as it relates to the authorized contents in the Model No. RAJ-II:

a. updated Table 1-2, "Maximum Weight of Uranium Dioxide Pellets per Fuel Assembly," to be consistent with Table 1, "Maximum Weight of Uranium Dioxide Pellets per Fuel Assembly," in the CoC by removing some of the details of the fuel assembly that are stated elsewhere.

b. updated Table 1-3, "Maximum Concentrations," by including updated values for ^{232}U (5.00×10^{-8} grams per gram of uranium (g/gU) instead of 2.00×10^{-9} g/gU) and the gamma emitters (4.4×10^{5} Megaelectronvolts-Bequerel per kg of uranium (MeV-Bq/kgU) instead of 5.18×10^{5} MeV-Bq/kgU.)

c. relocate Table 1-4, "Isotopes and A_2 Fractions," in Chapter 4, "Containment," of the application. The applicant updated this table (i.e., Table 1-4) to provide typical fuel structural materials for various components in a fuel assembly.

The applicant also deleted Table 1-5, "Typical Dimensions of the Main Components of Fuel Assembly and Fuel Rod," of the application and replaced it with an updated version of

Table 1-7, “Density of Structural Materials.” The updated version of the table changes the UO_2 density to be less than 98% theoretical consistent with the criticality safety analysis. The “Conditions” section of this SER includes additional discussions about the changes to the CoC related to the authorized contents in the Model No. RAJ-II package.

1.3 Criticality Safety Index and Transport Index

The criticality safety index (CSI) for fuel assemblies is 1.0 and the CSI for fuel rods is 1.6. Previously, the CSI for fuel rods was 2.1. (See Section 6.1.3 of this SER.)

The transport index for this package is based on the shielding analysis and assessments described in Chapter 5, “Shielding Evaluation,” of this SER.

1.4 Drawings

The applicant provided primarily administrative updates to the licensing drawings. The applicant simplified the drawings and reorganized the notes. The applicant did not make changes to the packaging design or materials. The information below includes the new revision Nos. of the drawings pertaining to this licensing action.

<u>Outer Container Drawings</u>

105E3737, Rev. 8	“Outer/Inner Container Assembly Licensing Drawing”
105E3738, Sheet 1, Rev. 11	“Outer Container Main Body Assembly Licensing Drawing”
105E3738, Sheets 2-3, Rev. 10	“Outer Container Main Body Assembly Licensing Drawing”
105E3739, Rev. 6	“Outer Container Fixture Assembly Licensing Drawing”
105E3740, Rev. 6	“Outer Container Fixture Assy. Installation Licensing” “Drawing”
105E3741, Rev. 3	“Outer Container Shock Absorber Assy. Licensing Drawing”
105E3742, Rev. 5	“Outer Container Bolster Assembly Licensing Drawing”
105E3743, Rev. 7	“Outer Container Lid Assembly Licensing Drawing”
105E3744, Rev. 8	“Outer Container Marking Licensing Drawing”

<u>Inner Container Drawings</u>

105E3745, Sheets 1-4, Rev. 10	“Inner Container Main Body Assembly Licensing Drawing”
105E3746, Rev. 3	“Inner Container Parts Assembly Licensing Drawing”
105E3747, Rev. 6	“Inner Container Lid Assembly Licensing Drawing”
105E3748, Rev. 4	“Inner Container End Lid Assembly Licensing Drawing”
105E3749, Rev. 8	“Inner Container Marking Licensing Drawing”

<u>Contents Containers</u>

105E3773, Rev. 2	“RAJ-II Protective Case Licensing Drawing”
0028B98, Rev. 2	“Shipping Container Loose Fuel Rods”

The applicant also provided the importance to safety classification for the packaging components per the guidance in NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.” The staff, including inspection and materials staff, reviewed the classification of the components and found them to be acceptable.

2.0 STRUCTURAL EVALUATION

The objective of the structural review is to verify that the changes proposed by the applicant to the Model No. RAJ-II meet the requirements of 10 CFR Part 71 as these relate to the structural performance of the package, including the tests and conditions for normal conditions of transport (NCT) and hypothetical accident conditions (HAC). The following sections summarize the staff's structural evaluation.

2.1 Description of the Structural Design

The structural design of the Model No. RAJ-II package remained unchanged from the previously approved structural design described in Revision 7.1 of the application (GNF 2014). However, the staff evaluated proposed changes in the application to ensure the structural design remained unaffected and assumptions remained valid.

2.2 Materials Evaluation under Hypothetical Accident Conditions

The applicant did not propose any changes to the materials of construction for the Model No. RAJ-II package. However, during the staff's review of the thermal performance of the package, the applicant provided information in its responses to requests for additional information (RAIs) 3-1 and 3-2 related to the performance of the fuel rod cladding in a fire (GNF 2017). The staff's evaluation of that information follows.

In its response to RAI 3-1 (GNF 2017), the applicant evaluated the effects of fuel rod cladding deformation (rod bending) on the ability of the cladding to maintain containment during a subsequent fire event. The staff notes that the applicant observed rod bending in a dropped package certification test unit. The applicant estimated the degree of strain (cold work) in the bent rods and concluded that it would have a negligible effect on the mechanical properties of the zirconium cladding alloys. The applicant also noted that the heat from a hypothetical fire event would only have the effect of relieving any cold work in the cladding. Therefore, the thermal performance of the fuel rod cladding would be unaffected by the rod bending. The staff reviewed the applicant's analysis and verified that the applicant provided a conservative estimate of the maximum degree of strain in the certification test unit's bent fuel rods. The staff also reviewed processing and mechanical property data for zirconium cladding alloys (ASM 1990) and verified that:

a. the cold work in the bent rods would not significantly reduce the ductility of the cladding, and

b. cladding temperatures during the postulated fire event are similar to those experienced by zirconium alloys to relieve cold work during typical process annealing.

Therefore, because the staff verified that the strain in the fuel cladding in a package drop would have an insignificant effect on the mechanical properties of the fuel cladding during the hypothetical fire, the staff finds the applicant's analysis of this event to be acceptable.

In its response to RAI 3-2 (GNF 2017), the applicant provided an evaluation to show that its creep model can accurately predict fuel rod cladding failure during a fire event for the variety of fuel rod geometries authorized in the Model No. RAJ-II package. The applicant first showed that the model's predicted failure temperature for BWR fuel rods is consistent with laboratory test results. The applicant then stated that the robustness of the model justifies its use for non-BWR fuel rods (i.e., PWR, CANDU). The applicant based this assertion on the following:

a. the calculation methodology used in the model was previously found to be acceptable to the NRC, and the BWR rod lab tests further validate the methodology, and

b. the creep model shows a very strong correlation between the cladding failure temperature and the initial cladding hoop stress, regardless of the geometry of the fuel rod.

The applicant stated that this strong correlation justifies the use of the creep model on non-BWR fuel rod geometries that were not validated with laboratory testing. The staff reviewed the applicant's analysis and the NRC's previous approval of the cladding creep calculation methodology (NRC 2010). The staff also reviewed the creep model's predicted failure temperatures for non-BWR fuel rods, which are at least 117°C greater than rod temperatures expected in a fire event (Section 3.4.4.2 of the application). Based upon the applicability of the creep calculation methodology previously reviewed by the staff (NRC 2010), the validation of the methodology by the BWR fuel rod laboratory testing, and the significant margin between the predicted failure temperatures of non-BWR fuel rods and the rod temperatures expected in a fire, the staff finds the applicant's use of the creep model to be acceptable.

2.3 Structural Evaluation under Normal Conditions of Transport and Hypothetical Accident Conditions

In the application (GNF 2016a), the applicant requested that the payload within each RAJ-II package consists of a maximum of two unirradiated BWR fuel assemblies or individual rods (BWR, uranium carbide, or generic PWR) contained in a 5-inch stainless steel pipe, protective case or strapped together and positioned in one or both sides of the inner container. The currently approved payload within each RAJ-II package consists of a maximum of two unirradiated BWR fuel assemblies or individual rods (BWR, uranium carbide, or generic PWR) contained in a cylinder, protective case or bundled together and positioned in one or both sides of the inner container. The applicant indicated that a maximum gross shipping weight of 1,614 kg (3,558 lbs.) and a maximum packaging weight of 930 kg (2,050 lbs.) were not changed from the previously approved structural design. As a result, the previous analyses remain applicable and valid. The applicant concluded that the previous analyses and certified model tests are bounding and no further structural evaluations were needed.

The staff reviewed the structural analyses of the application and the resultant weight, pressure, and temperature loads due to the proposed change in the payload. Because the weight, pressure and thermal loads are bounded by the loading parameters previously approved for the structural design, the staff concludes that the previous analyses and certified model tests are applicable and valid for both NCT and HAC.

2.4 Evaluation Findings

Based on a review of the statements and representations in the application, the staff concludes that the applicant adequately described and evaluated the structural design, properties of the materials, and the structural integrity of the package in the application. Therefore, the package meets the requirements of 10 CFR Part 71 related to the structural design.

3.0 THERMAL EVALUATION

The purpose of this thermal evaluation is to verify that the proposed changes to the design of the Model No. RAJ-II:

1. provide adequate protection against the thermal tests specified in 10 CFR Part 71, and

2. meet the thermal performance requirements of 10 CFR Part 71 under NCT and HAC.

Regulations applicable to the thermal review include 10 CFR 71.31, 71.33, 71.35, 71.43, 71.51, 71.71, and 71.73. The following sections summarize the staff's thermal evaluation.

3.1 General Considerations

The staff reviewed the RAJ-II package to verify that the applicant adequately described and evaluated the RAJ-II package thermal design, relevant to the proposed changes to the CoC, as required per 10 CFR Part 71. According to Section 1 of the application, the content of the RAJ-II package includes unirradiated BWR fuel assemblies or unirradiated BWR, PWR, and CANDU fuel rods.

Even though there were no changes to the thermal design as part of this CoC revision, the applicant revised the safety analysis report to reflect editorial changes, a slight change in parameters of the 10 × 10 BWR fuel, and the addition of a creep-based thermal-stress criterion for non-BWR fuel. The staff focused its evaluation on the impacts of these changes into the thermal evaluation.

The applicant discussed the effect of a thermal HAC on the certification test unit's deformed cladding after undergoing the hypothetical accident test condition in its response to RAI 3-1 (GNF 2017). Section 2.2 of the SER includes an evaluation of the RAI response.

According to the applicant's response to request for supplemental information (RSI) 3-1 (GNF 2016b), the previously approved application, Revision 7.1, (GNF 2014) bounds the analysis in the thermal chapter of the application (Revision 9) (GNF 2016a) for the Model No. RAJ-II with the exception that Revision 9 of the application adds a thermal-stress criterion for non-BWR fuel. As part of the RAI 3-2 response (GNF 2017), the applicant provided an explanation that described the relevance of the thermal-stress criterion to the non-BWR fuel. Section 2.2 of the SER includes an evaluation of this thermal-stress criterion.

In addition, the staff notes that the applicant proposed changes to the 10 × 10 BWR content parameters that are listed in the CoC. According to the applicant's response to RSI 3-1(GNF 2016b), these parameters do not affect the thermal analysis provided in a previous RAJ-II application, Revision 7.1 (GNF 2014). Therefore, the staff concludes that the proposed changes do not significantly affect the decay heat of the system because the content is unirradiated fuel.

3.2 Evaluation Findings

Based on a review of the thermal chapter of the application, the staff concludes that the applicant has adequately described and evaluated the RAJ-II thermal design, relevant to the proposed changes to the CoC, and has reasonable assurance that the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The purpose of the containment evaluation is to:

1. verify that the proposed changes to the Model No. RAJ-II package meet the requirements for the containment and

2. ensure that the applicant adequately described and evaluated the containment design under NCT and HAC as required in 10 CFR Part 71.

The following sections summarize the staff's containment evaluation.

4.1 Description of the Containment System

The applicant updated the safety analysis report (SAR) (the application) for the Model No. RAJ-II to modify the authorized contents. The applicant did not request modifications to the previously approved packaging design. The applicant proposed changes to the CoC related to the description of the containment boundary for the package. In previous applications of the Model No. RAJ-II, the applicant described the containment boundary as the fuel rod cladding and ceramic nature of the fuel pellets. The applicant updated the description of the Model No. RAJ-II containment boundary as follows:

> "[t]he fuel rod cladding and welded end plugs provide the primary containment of the radioactive material."

The radioactive material is contained in ceramic sintered pellets and sealed in zirconium alloy or stainless steel cladding. The fuel is leak tested to demonstrate that it is leak tight to 1×10^{-7} reference cubic centimeters per second (ref-cm^3/s) per American National Standards Institute (ANSI) N14.5.

4.2 General Considerations

4.2.1 Type A Fissile Packages

The Model No. RAJ-II package can only be used as a Type B package.

4.2.2 Type B Packages

The applicant added Appendix 4.5 to the SAR to provide detailed calculations for the determination of allowable leak rates. The applicant demonstrated a release rate less than 10^{-6} A_2/hr. Therefore, satisfying the quantified release rate of 10 CFR 71.51.

4.3 Containment Under Normal Conditions of Transport

The welded containment boundary is not affected by any of the NCT as demonstrated by the structural and thermal evaluations. In Appendix 4.5 of the application, the applicant calculated the allowable release rate and associated allowable leak rate for NCT assuming 15% of the ceramic pellet material converts to an aerosol mixture. The post HAC drop test leak test results for an entire 10 × 10 fuel bundle is less than the NCT allowable leak rate for one rod. Therefore, the staff finds the analysis of the containment boundary to be acceptable for NCT.

4.4 Containment Under Hypothetical Accident Conditions

Following the HAC thermal and drop tests, the applicant helium leak tested the fuel rods. Two of the tests showed no change in meeting the ANSI N14.5 leak tight criteria (i.e., 1×10^{-7} ref-cm^3/s) and one shown to have a leak rate of 5.5×10^{-6} ref-cm^3/s. Although this does not meet the leak tight criteria, the staff finds this acceptable based on the calculations in Appendix 4.5, which demonstrate that this post HAC drop test actual leak rate of a 10 × 10 bundle is less than the calculated HAC allowable leak rate for one rod (5.39×10^{-1} ref-cm^3/s).

4.5 Leakage Rate Tests for Type B Packages

During manufacturing, each fuel rod and weld joint is helium leak tested to demonstrate leak tightness per ANSI N14.5. The fabrication leak rate test satisfies the requirements for the pre-shipment leak rate test, and there are no maintenance or periodic leak rate tests for the fuel. There are no leak rate requirements for the inner and outer packaging, since they are not relied on for containment.

The applicant included a reference to ANSI N14.5 for its definition of leaktight and included 10 CFR 71.51 for acceptable methods of leak testing. The applicant provided updates to the allowed deformation of the package shell, foam, and paper honeycomb.

4.6 Evaluation Findings

Based on the review of the statements and representations in the application, the staff concludes that the applicant adequately described and evaluated the Model No. RAJ-II package containment design and that the package design meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The purpose of the shielding review is to ensure that there is adequate protection to the public and workers against direct radiation from the contents of the Model No. RAJ-II transportation package and to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under NCT and HAC.

5.1 Description of the Shielding Design

5.1.1 Design Features

The general package description is provided in Sections 1.2 and 6.1.1.1 of the application. The Model No. RAJ-II transportation package is designed to transport unirradiated fuel which does not emit significant gamma or neutron radiation. The RAJ-II package allows positioning of the radiation sources within the inner container, which provides distance to the package surface. The applicant only modeled attenuation in select stainless steel components of the inner container to determine the gamma dose rates in addition to the radiation from uranium of the fuel.

5.1.2 Summary Table of the Maximum Radiation Levels

The applicant calculated dose rates based on conservative assumptions and simplified models to demonstrate compliance with the requirements of 10 CFR Part 71. Table 5-1 of the application shows that all dose rate requirements are met for non-exclusive use of transport.

The applicant stated that prior to shipping, direct measurements of the dose rates will be performed. The measurements are used to confirm compliance with the NCT dose rate limits established by 10 CFR 71.47(a). The packages are labeled with the Transport Index (TI) based on these measurements.

5.2 Source Specifications

Table 4-3 of the application shows the content maximum isotopic mass that the contents may contain uranium, neptunium, and plutonium isotopes, which decay by alpha emission. Contents may include technitium-99 which decays by beta emission. Alpha and beta particles have a short range and are not significant contributors to external dose rates.

5.2.1 Gamma Source

According to the applicant, the gamma source from gamma emitting isotopes other than actinides present in either the commercial-grade uranium or enriched reprocessed uranium has a maximum concentration of gamma emitters of 4.4×10^5 mega electron volts (MeV) per second per kilogram of uranium (MeV/s/kg U) (according to American Society for Testing and Materials (ASTM) C996-15). The maximum uranium payload is 484 kg (1,067 lbs.). A bounding gamma source term was used in the shielding analysis.

The applicant uses ORIGEN 2, version 2.1, to determine the spectrum and the source strength of the gammas emitted due to decay of the actinides. This includes gammas emitted from all decay modes, such as alpha or spontaneous fission, for all the actinides as well as the contribution from Bremsstrahlung due to slowing down of beta particles in the fuel, albeit a small effect on the gamma dose rate. The gamma source strength from actinides is 3.73×10^{10} gammas/s. This gamma spectrum is used for both NCT and HAC. Gamma source term is shown in Table 5-2 of the application.

5.2.2 Neutron Source

The neutron source strength from spontaneous fission and alpha-n events is determined using ORIGEN2, version 2.1, as 9.94×10^3 neutrons per second (n/s). According to the applicant, the main contributors to the spontaneous fission neutron source are ^{238}U, ^{236}U, and ^{235}U, totaling 5.74×10^3 n/s, and the main contributors to the alpha-n source are ^{234}U, ^{232}U, ^{236}U, ^{238}U, and ^{235}U, totaling 4.20×10^3 n/s.

5.3 Shielding Model Specifications

5.3.1 Configuration of Source and Shielding

The applicant stated that it only modeled the stainless steel sheets of the inner container, including the outer shell, the inner shell and the divider plate between the fuel compartments. The applicant modeled the alumina silicate thermal insulator and the polyethylene cushioning foam as void. The distance provided by the outer container was maintained but the stainless steel sheet of the outer container was modeled as void. Table 6-7 of the application includes the stainless steel sheet thicknesses accounting for dimensional tolerances. The dimensions of the outer container and inner container are shown in Figures 6-3, 6-4, and 6-5 for NCT, and Figures 6-6, 6-7, and 6-8 for HAC. The difference between the NCT and HAC models is the package dimensions. For both NCT and HAC, the source is modeled as two volumetric sources centered within the fuel compartments, with uranium density of 3.10 g/cm^3.

The applicant do not credit any materials in its neutron models, only the distances provided by the package. This approach accounts for any additional contribution to the total dose rate due to neutron induced gammas.

5.3.2 Material Properties

Table 6-8 of the application includes the material composition of the package. The applicant calculated the uranium density by assuming a maximum uranium payload of 484 kg smeared over the total compartment volume using the minimum active fuel length of 381 cm (7.80×10^4 cm^3). The resulting uranium density is 3.10 g/cm^3.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant stated that they used ORIGEN2, version 2.1, with the default DECAY, BWRUE, and GXUO2BRM libraries to determine the gamma and neutron source terms. The applicant used ORIGEN to obtain the source term due to decay of the actinides because fuel depletion is not necessary. The applicant used the Monte Carlo N-Particle code (MCNP5), version 1.30, with ENDF/B-VII.0 libraries (RSICC CCC-810) for the shielding analysis (GNF 2016b).

5.4.2 Input and Output Data

The applicant provided the MCNP5 input and output files. According to the applicant, the key RAJ-II package dimensions used in the shielding models were similar to those used in Chapter 6 of the application. Table 5-3 of the application includes the RAJ-II package dimensions. The outputs show that all tallies pass all of the MCNP5 statistical checks.

5.4.3 Flux-to-Dose Rate Conversion

The applicant uses ANSI/American Nuclear Society (ANS)-6.1.1-1977, “Neutron and Gamma Ray Fluence-to-Dose Factors,” flux-to-dose rate conversion factors in this evaluation. Tables 5-4 and 5-5 of the application show the gamma and neutron conversion factors.

5.4.4 External Radiation Levels

The model used in this amendment only credits the uranium mass and RAJ-II materials such as inner container stainless steel sheets. The approximations used in the source terms are acceptable, if ORIGEN2, version 2.1, calculations are verified by an undated depletion code. Due to approximations and other geometric modeling simplifications, the applicant applied an additional safety factor of 1.2 to the dose rate results provided in Table 5-1 of the application, which includes a 2 sigma statistical uncertainty. Table 5-1, "Summary Table of External Radiation Levels," of the application shows that the neutron contribution to the total dose rate is negligible. The results demonstrate that even though the shielding evaluation uses a conservative approach and simplified models, there is at least 93% margin to the regulatory limits in 10 CFR Part 71.

5.5 Conclusion

The staff reviewed the description of the package design features related to the shielding evaluation, the source terms, and the method and instructions for determining the contents. The staff reviewed also the shielding analyses, the assumptions and approximations used in the analyses as presented in the shielding safety analysis, and the results of the analysis, the maximum dose rates for NCT and HAC. The staff determined that the reported values were below the regulatory limit in 10 CFR 71.47 (a).

5.6 Evaluation Findings

Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding design of the Model RAJ-II package and the corresponding shielding evaluation is consistent with the appropriate codes and standards for shielding analyses, NRC's guidance, and requirements in 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The staff reviewed the applicant's proposed changes to the CoC for the Model No. RAJ-II package and the associated changes to the applicant's criticality analysis in Chapter 6 of the application. The specific changes that affect the criticality safety analysis are the following:

1. an additional configuration of the GNF 10 × 10 fuel assembly;
2. gadolinia (UO_2-Gd_2O_3) rod distribution requirements for all BWR fuel assemblies;
3. revised polyethylene equivalent mass limits for loose fuel rod shipment; and
4. increased maximum number of rods per compartment for CANDU-14 and CANDU-25 rods in five inch stainless steel pipes.

6.1 Description of Criticality Design

6.1.1 Design Features

The applicant did not request any changes to the design features for criticality safety associated with the RAJ-II packaging. However, the applicant did request several changes to the arrangement of fissile material, moderators, and neutron absorbers associated with the package contents.

The applicant revised footnote (b) of Table 6-1 of the application to describe gadolinia rod position requirements for fuel assembly contents. These gadolinia rod placement requirements reflect the configuration considered in the previously approved analysis for BWR fuel assemblies, described in Section 6.3.4.2 of the application, with the exception that some analyzed configurations include gadolinia rods in peripheral locations (see Figures 6-34 through 6-36 of the application). The applicant assumed that gadolinia rods in peripheral locations is conservative. However, gadolinia rods on the periphery of an assembly are less effective at absorbing neutrons than in interior assembly positions in the RAI-II package. Therefore, the staff finds the gadolinia rod placement requirements in footnote (b) of Table 6-1 to be conservative and acceptable.

The applicant revised Table 6-2 of the application to remove a limit on loose rod protective polyethylene sleeve thickness, and to include explicit limits on polyethylene packing material for loose rod shipment. The polyethylene mass considered in the loose rod analysis (unlimited amount for BWR rods and greater than 29 kg for CANDU and generic PWR rods) bounds the consideration of polyethylene sleeve thicknesses. Since these large polyethylene masses are called out explicitly in Table 4 of the CoC, the staff finds that a limit on sleeve thickness is redundant and unnecessary. Therefore, the staff determined that the using the restriction of total polyethylene mass in place of controlling the polyethylene sleeve thickness is acceptable.

The applicant also revised the application to replace the cladding thickness with water in the criticality analysis for loose rods. Criticality safety of the package with loose rods, therefore, does not depend on the presence of cladding. BWR assemblies are under-moderated, meaning that any moderator added to the assembly will increase reactivity. The staff finds that replacing the clad thickness with water in the criticality analysis models is conservative because of the additional moderation provided by this assumption.

The applicant revised Table 6-2 of the application to increase the number of CANDU-14 and CANDU-25 uranium carbide rods allowed to be transported in the RAJ-II package inside of a five-inch stainless steel pipe. The applicant previously evaluated CANDU rods as having a 450-centimeter (cm) active length, and determined the number of such rods that would be subcritical in the loose rod configuration. The applicant did not revise the analysis, but changed the number of allowable rods to reflect the actual active lengths of both types of CANDU rods (e.g, the 450-cm active length considered in the analysis, divided by the active length of the CANDU-14 rod, 47.752 cm, is roughly 9.4. Multiplying by the 74 rods considered in the criticality analysis yields 695 rods). Since the total mass and arrangement of fissile material considered in the analysis has not changed, the staff finds this increase in the allowable number of loose CANDU rods to be acceptable.

6.1.2 Summary Table of Criticality Evaluations

The applicant revised the summary table of criticality evaluations in Section 6.1.2 of the application to reflect the revised criticality analysis for the GNF 10 × 10. The applicant demonstrated that this new design, with a different water hole geometry and more partial length rods (16) has a higher k_{eff} than all of the other fuel types evaluated in the package. The applicant revised Table 6-3 of the application to reflect the revised upper subcritical limit (USL) associated with the applicant's revised analysis using SCALE 6.1 and continuous energy

ENDF/B-VII cross-section data. The maximum k_{eff} calculated for single packages and arrays of multiple packages, under both NCT and HAC, remains below the USL in all cases.

6.1.3 Criticality Safety Index

The applicant revised the Criticality Safety Index (CSI) for loose fuel rods to reflect the revised subcritical array size evaluated in the application. Using the same code and model as used in the previously approved analysis, the applicant demonstrated that an array of 8 × 8 × 1 packages (64) remained subcritical under HAC, which bounded the NCT array, resulting in a CSI of 1.6. The CSI for all other configurations remains at 1.0.

6.2 Fissile Material Contents

The applicant requested to add an additional configuration of the GNF 10 × 10 fuel assembly, with a 3 × 3 axially varying water hole and two additional partial length rods. The type and amount of fissile material allowed in the package is unchanged.

6.3 General Considerations

6.3.1 Model Configuration

The changes described in Section 6.1.1 of this SER did not require any changes to the previously approved criticality model for the RAJ-II package. All of the previously identified maximum reactivity configurations are applied to the model for the new optional GNF 10 × 10 fuel assembly contents, which the staff agrees is appropriate and conservative given the similarity of this fuel assembly design to the previously approved contents.

The applicant did not change the inner or outer container model from what was previously evaluated and approved. The only change to the model involved the addition of a different configuration for the GNF 10 × 10 fuel assembly. The new optional configuration for the GNF 10 × 10 fuel assembly has a single larger water rod, with an axially varying cross-section, and a maximum 3 × 3 rod displacement. The water rod is slightly off-center in the fuel assembly. This new assembly design includes up to 16 partial length rods – 2 more than the previously approved GNF 10 × 10 fuel assembly. Additionally, the applicant's model for the GNF 10 × 10 fuel assembly includes an increased pitch size, which bounds possible pitch sizes for all GNF 10 × 10 fuel assemblies.

The applicant determined the most reactive gadolinia rod configuration for the new GNF 10 × 10 fuel assembly design in the same way as for previously approved fuel assembly contents, as discussed in Section 6.3.4.2 of the application. Figure 6-25 of the application shows the resulting configurations. As with the configurations determined for previously approved contents, the gadolinia rod distribution for the new GNF 10 × 10 fuel assembly is hypothetical – gadolinia rods in actual fuel assembly designs are more evenly distributed, leading to much lower k_{eff} for these assembly designs. Therefore, the staff finds that the gadolinia rod configuration evaluated in the criticality model for the new GNF 10 × 10 fuel assembly design is acceptable and conservative.

6.3.2 Material Properties

The applicant did not request significant changes to the material properties of the package or its fissile material contents. Although the applicant proposes changes to the maximum allowable content of ^{232}U and gamma emitters in Table 2 of the CoC, these changes do not have any effect on the criticality safety of the package.

6.3.3 Computer Codes and Cross-Section Libraries

For the revised limiting GNF 10 × 10 fuel assembly analyses, the applicant modeled the system using SCALE 6.1, with the KENO-VI three dimensional Monte Carlo neutron transport code and the ENDF/B-VII continuous-energy cross-section library. The SCALE code system is a standard in the nuclear industry for performing Monte Carlo criticality safety calculations, and the ENDF/B-VII cross-section library is the most up-to-date nuclear data available with SCALE 6.1. Therefore, the staff finds the use of this code and cross-section data appropriate for analysis of the RAJ-II package and its contents.

For the revised loose rod package array analysis, to support a CSI of 1.6, the applicant used the GEMER Monte Carlo criticality code. The applicant previously validated this code for modeling loose rod package arrays, and the revised analysis is within the range of applicability determined in the previously approved validation analysis.

6.3.4 Demonstration of Maximum Reactivity

The applicant revised the package model previously demonstrated to be the most reactive, under NCT and HAC, to include the new GNF 10 × 10 fuel assembly contents. This model included:

(1) maximum fuel pellet diameter;

(2) maximum active fuel length;

(3) maximum fuel rod pitch;

(4) minimum cladding thickness;

(5) most reactive packaging material and fabrication tolerances;

(6) most reactive interstitial moderator density for packages in an array under NCT;

(7) most reactive combination of interstitial, outer container, and inner container moderator density under HAC; and

(8) most reactive mass and configuration of polyethylene packing materials in the packaging and within the fuel assembly contents.

6.4 Single Package Evaluation

6.4.1 Configuration

The applicant used the single package model determined to be the most reactive in the reactivity studies performed as described in Section 6.3.4 of the application. The applicant used the gadolinia rod configuration determined to be the most reactive from the HAC array calculation, as that configuration is more limiting than the single package. The applicant considered a full density 30-cm water reflector for both single package models. For the package under NCT, the applicant varied the outer container and inner container moderator density uniformly. For hypothetical accident conditions, the applicant modeled the outer container without moderator, and varied the moderator density in the inner container.

6.4.2 Results

The applicant determined the most reactive condition to be a single package under HAC, with the GNF 10 × 10 with a single varying water rod, 16 partial-length rods, fuel enriched to 5.0 weight percent ^{235}U, and twelve 2.0 wt.% gadolinia rods. However, there was little difference in reactivity between the NCT and HAC models, due to the similar moderation conditions. The k_{eff} for the single package under NCT and HAC are both below 0.71, which is significantly below the USL of 0.9340.

The staff reviewed the applicant's evaluation of the RAJ-II single package. The configurations evaluated by the applicant are consistent with the recommendations in the staff review guidance in NUREG-1609. The staff finds that the single package remains subcritical under NCT and HAC, meeting the requirements of 10 CFR 71.55.

6.5 Evaluation of Package Arrays

6.5.1 Configuration

The applicant evaluated package arrays under NCT and HAC using the most reactive model determined for the single package. The NCT package array consists of an array of 1,512 undamaged packages, with uniformly varying water density in the inner and outer containers. The applicant determined that the highest k_{eff} was obtained with no water present, as shown in Figure 6-46 of the application.

The HAC array consists of 100 damaged packages, where the water density is varied separately inside the inner container and outer container. The applicant modeled varying numbers of 2.0 wt.% gadolinia rods to determine the number required to maintain the array below the calculated USL. Table 6-1 of the application includes the resulting gadolinia rod requirements. The applicant determined that the most reactive fuel assembly configuration is with:

(1) 5.0 wt.% ^{235}U fuel,

(2) a single varying water rod,

(3) sixteen partial length rods, and

(4) twelve 2.0 wt.% gadolinia rods.

The applicant determined that the highest k_{eff} was obtained with no water in the outer container, and with the inner container fully flooded, as shown in Figures 6-47 and 6-48 of the application.

Additionally, the applicant evaluated the effect of the fuel channel thickness on the HAC array k_{eff}. The applicant determined that the variation of the fuel channel thickness, from largest tolerance to zero thickness, had a statistically insignificant effect on system reactivity. Therefore, the applicant demonstrated and the staff determined that the channel is not required to be present for the GNF 10 × 10 fuel assembly.

6.5.2 Results

The applicant determined the maximum k_{eff} for the NCT package array to be 0.8122, which is significantly below the calculated USL. For the HAC array, the applicant determined the maximum k_{eff} to be 0.9300, which is below the calculated USL.

The staff reviewed the applicant's evaluation of arrays of RAJ-II packages under NCT and HAC. The array configurations evaluated by the applicant are consistent with the recommendations in the staff review guidance in NUREG-1609. The staff finds that arrays of packages remain subcritical under NCT and HAC, meeting the requirements of 10 CFR 71.59.

6.6 Benchmark Evaluations

6.6.1 Applicability of Benchmark Experiments

The applicant validated package criticality safety calculations using SCALE 6.1 with the ENDF/B-VII cross-section library against 75 low enriched uranium heterogeneous compound critical experiments similar to the package configuration. Seventy of these experiments consisted of uranium compounds without neutron absorbers, and five experiments included gadolinia rods. These experiments span the range of parameters. As shown in Table 6-26 of the application, the applicant demonstrated the area of applicability for the code benchmarking analysis. The staff finds the critical experiments selected by the applicant adequate and acceptable.

6.6.2 Bias Determination

Due to the limited availability of critical experiments with gadolinia rods, the applicant performed the validation analysis by determining the following:

(1) the bias and bias uncertainty from the 70 configurations without gadolinia;

(2) the effect on bias and bias uncertainty from adding the five configurations with gadolinia rods; and

(3) an additional margin to subcriticality based on a similarity analysis comparing the RAJ-II package with the five gadolinia rod critical configurations selected for the validation.

The applicant calculated a bias and bias uncertainty using the single-sided lower tolerance limit method. The k_{eff} results for the critical configurations were normally distributed, and did not exhibit any significant trends. The applicant determined that there was no significant effect on the bias and bias uncertainty due to the addition of five configurations with gadolinia rods. However, due to the low number of gadolinia configurations available, the applicant determined a 0.003 additional margin of subcriticality and subtracted it from the calculated bias. The resulting USL, which includes the bias, bias uncertainty, 0.05 administrative margin, and additional margin of subcriticality, is 0.9340.

The applicant's benchmark analysis considered a few critical configurations that were representative of the RAJ-II package with gadolinia rods. Due to the low number of gadolinia configurations available, the applicant determined and applied an additional penalty to account for the lack of configurations, resulting in a calculated bias consistent with the SCALE 6.1 bias determined previously for similar fresh light water fuel systems. Therefore, the staff has reasonable assurance that the calculated USL is appropriate for the analysis of the RAJ-II package using SCALE 6.1 and the ENDF/B-VII continuous-energy cross-section library.

6.7 Conclusion

The staff finds that the applicant demonstrated that the RAJ-II package, when loaded with fuel assemblies meeting the characteristics described in Table 6-1 of the application, or loose fuel rods meeting the characteristics described in Table 6-2 of the application, will remain subcritical with adequate safety margin under NCT and HAC. On this basis, the staff finds that the RAJ-II package meets the fissile material requirements of 10 CFR 71.55 for single packages, and 10 CFR 71.59 for arrays of packages with a CSI of 1.0 for the BWR fuel assembly contents in Table 6-1 of the application, and 1.6 for the loose rod contents in Table 6-2 of the application.

6.8 Evaluation Findings

Based on the review of the statements and representations in the application, the staff has reasonable assurance that the applicant adequately described and evaluated the RAJ-II criticality safety evaluation and that the package design meets the requirements of 10 CFR Part 71.

7.0 PACKAGE OPERATIONS

The applicant provided mostly editorial changes to the package operations. The applicant added a note to ensure that the latest revision of the regulations are applicable to the package operations.

Under preparation for loading, the applicant provided a table with tightening instructions for the inner container, outer container, and hold down clamp bolts. The applicant added references to different sections of the application such as the polyethylene mass limits in Chapter 6. The applicant also removed the replacement parts list as the licensing drawings to ensure that those parts are replaced per their classification.

Therefore, the Model No. RAJ-II still meets the requirements of 10 CFR 71.87.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

The applicant provided mostly editorial changes to the acceptance tests and maintenance program. The applicant included discussion on material testing/certification as discussed in NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.” The applicant also referenced alumina silicate thermal properties to Section 3 of the application. Since the changes to Chapter 8 of the application were mainly editorial, the package still meets the requirements of 10 CFR Part 71 as it relates to acceptance tests and the maintenance program.

9.0 REFERENCES

(ASM 1990) ASM Handbook, Volume 2: "Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, "Zirconium and Hafnium," ASM International, Materials Park, Ohio, 1990, pp. 661-669.

(GNF 2014) Phillip D. Ollis, Global Nuclear Fuel, letter to Director, Division of Spent Fuel Management, U.S. Nuclear Regulatory Commission, July 11, 2014, Agencywide Documents Access and Management System (ADAMS) Accession Number (No.) ML14195A240.

(GNF 2016a) Murray, Scott P., Global Nuclear Fuel, letter to Director, Division of Spent Fuel Management, U.S. Nuclear Regulatory Commission, September 30, 2016, ADAMS Accession No. ML16274A097.

(GNF 2016b) Moore, Brian R., Global Nuclear Fuel, letter to Director, Division of Spent Fuel Management, U.S. Nuclear Regulatory Commission, November 28, 2016, ADAMS Accession No. ML16333A225.

(GNF 2017) Moore, Brian R., Global Nuclear Fuel, letter to Director, Division of Spent Fuel Management, U.S. Nuclear Regulatory Commission, April 7, 2017, ADAMS Accession No. ML17097A102.

(LANL 1987) Los Alamos National Laboratory. Monte Carlo Team 2003, "MCNP—A General Monte Carlo N-Particle Transport Code," Version 5, LA-UR-03-1987, Los Alamos, N.M.

(NRC 1999) U. S. Nuclear Regulatory Commission. NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," March 31, 1999.

(NRC 2010) U. S. Nuclear Regulatory Commission. "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report NEDC-33256P, NEDC-33257P, and NEDC-33258P, "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance," Global Nuclear Fuel – Americas, LLC, Project No. 712" January 22, 2010, ADAMS Accession No. ML100150653.

CONDITIONS

The following changes have been made to the CoC:

Condition No. 3.(b) has been updated to reflect the date of the latest application, as supplemented.

Condition No. 5.(a)(2), "Description," has been updated to remove the year for the ASTM C996 standard and the description of the containment boundary has been updated. Previously, the containment boundary was described as just the fuel rod clad and ceramic nature of the fuel pellets. The applicant proposed to update the description of its containment boundary as follows:

> "[t]he fuel rod cladding and welded end plugs provide the primary containment of the radioactive material. The radioactive material is bound in ceramic pellets with very limited solubility and minimal propensity to suspend in air."
>
> Condition No. 5.(a)(2) was updated as follows:
>
> "The fuel rod cladding and welded end plugs provide primary containment of the radioactive material. The radioactive material is bound in ceramic pellets with limited solubility and minimal propensity to suspend in air."

The condition has also been updated to include the "loose rods pipe nominal mass per component" and the "protective case nominal mass per component" values.

Condition No. 5.(a)(3), "Drawings," has been revised with the latest revisions of the drawings.

Condition No. 5.(b)(1), "Type and form of material," has been revised to:

- remove the year for the ASTM C996 standard, and
- include updated values for ^{232}U and the gamma emitters in Table 2, "Maximum Authorized Concentrations."
- Delete the following sentence from Condition Nos. 5.(b)(1)(iv) and (v):

  "When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness."

- update Table 3, "Fuel Assembly Parameters."
  - For the GNF 10 × 10, it has been updated to include "1-axially varying centered" water rods a longer fuel rod pitch, any channel thickness, and up to 16 partial length fuel rods.
  - For all fuel types, the fuel rod pitch is not listed as the "nominal fuel rod pitch."
  - For all fuel types, the lattice average enrichment is not listed as the "maximum lattice average enrichment."
  - For all fuel types, the number of partial length fuel rods allowed are no longer listed as a maximum but as a range.
  - For all fuel types, the minimum gadolina requirements have some changes and also represent greater lattice average enrichments.
  - The polyethylene equivalent mass per assembly is no longer listed as a maximum per assembly but as a range.

    - The table has been updated to include an equation for thermal performance criteria for all of the assemblies.

    - Revise the "units" column for consistency.

    - Table footnotes are also included for the gadolina rods and the thermal performance criteria equation.

- update Table 4, "Fuel Rod Parameters."

    - For 8 × 8, 9 × 9, and 10 × 10 fuel assemblies, the fuel density has changed from "less than" to "less than or equal to." The fuel density for the CANDU-14 and CANDU-25 fuel types has decreased from 98% to "≤97%". The fuel density for the generic PWR has increased to "≤100%."

    - "Polyethylene equivalent mass per compartment", "reference density for polyethylene equivalent mass calculation", and "thermal performance criteria" parameters have been added to the table.

    - The maximum number of loose rods for the CANDU-14 and CANDU-25 have increased from 74 to 695 and from 130 to 1,458, respectively.

    - Revise the "units" column for consistency.

    - The footnotes to the table have been changed and updated.

Condition No. 5.(c), "Criticality Safety Index for contents described in 5.(b)(1)(v) and limited in 5.(b)(2)(ii)," has been updated to 1.6 from 2.1.

Condition No. 8 has been updated to include the following:

> "Additionally, for GNF 10 × 10 fuel assembly designs, the water rod can occupy a space equivalent to a single 2 × 2 fuel rod equivalent at the bottom of the assembly and expanded at the top; this configuration is designated as 1-axially varying centered in the table."

Condition No. 11 has been revised to allow use of Revision 10 of the CoC for approximately one year.

The references section has been updated to include the application and responses to supplemental information and requests for additional information.

CONCLUSION

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

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