NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9228	28	71-9228	USA/9228/B(U)F-96	1	OF 4

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

GE Hitachi Nuclear Energy consolidated application dated April 28, 2016, as supplemented.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: 2000

(2) Description

The cask body is constructed of two concentric 1-inch thick 304 stainless steel cylindrical shells (ASTM A240) joined at the bottom end to a 6-inch thick 304 stainless steel forging (ASTM A182). The overall packaging dimensions are approximately 131.5 inches in height and 72 inches in diameter, and its gross weight is approximately 33,550 lbs. The cavity of the packaging is approximately 26.5 inches in diameter and 54.0 inches deep.

The cask lid is fully recessed into the cask top flange and secured to the cask body by 15, 1.25-inch diameter socket head screws. The packaging is equipped with a seal test port on the side of the body, a vent port in the lid, and a drain port near the bottom of the packaging. The cask lid utilizes four O-rings in a metal retainer.

The overpack is constructed from two 0.5-inch thick concentric 304 stainless steel cylindrical shells (ASTM A240), separated radially by eight equally spaced tubes and horizontally by two tube sections. A 304 stainless steel toroidal shell impact limiter is attached to each end of the overpack. The overpack opens just above the lower impact limiter for access to the packaging. The top of the overpack is joined to the base by 15, 1-3/8-inch diameter shoulder screws. Gussets on the top and bottom impact limiters provide tie-down points for the package. The lifting devices are detached during transport.

5.(a) (2) Description (Continued)

A high performance insert (HPI) is fabricated from two concentric stainless steel cylindrical shells. The annulus between the cylindrical shells is filled with depleted uranium. The HPI is positioned within the cask cavity by support disks arranged axially to provide uniform support. Vertical lifting arms connect the support disks and also serve as the primary lifting fixtures. The HPI is shielded using encapsulated depleted uranium within both a top and a bottom plug. The top plug has a stepped design and an optional spacer may be added to provide additional shoring.

A material basket is also used for the shipment of contents described in 5(b)(1)(ii).

(3) Drawings

(i) With the exception of packaging Serial No. 2001, the packaging is constructed and assembled in accordance with the following General Electric Company Drawings:

Drawing No.	Drawing Title	Revision
129D4946	Model 2000 Transport Container	12
105E9520	Model 2000 Shipping Cask all S/N's Except S/N 2001	9 (Sheet 1 of 2) 9 (Sheet 2 of 2)
105E9521	Model 2000 Cask Overpack All S/N's Except S/N 2001	7

(ii) Packaging Serial No. 2001 is constructed and assembled in accordance with the following General Electric Company Drawings:

Drawing No.	Drawing Title	Revision
129D4946	Model 2000 Transport Container	12
101E8718	Model 2000 Shipping Cask S/N 2001	17 (Sheet 1 of 2) 17 (Sheet 2 of 2)
101E8719	Model 2000 Shipping Cask Overpack S/N 2001	14

(ii) The HPI and HPI material basket are constructed and assembled in accordance with the following General Electric Company Drawings:

Drawing No.	Drawing Title	Revision
001N8422	GE 2000 HPI and Material Basket Licensing Drawing	3
001N8423	GE 2000 HPI Licensing Drawing	2
001N8424	GE 2000 HPI Material Basket Assembly Licensing Drawing	2
001N8425	GE 2000 HPI Body Licensing Drawing	2
001N8427	GE 2000 HPI Top Plug Assembly Licensing Drawing	2
001N8428	GE 2000 HPI Bottom Plug Assembly Licensing Drawing	2

5.(b) (1) Contents

(i) Irradiated hardware components composed of metallic alloys (e.g., stainless steels, carbon steels, FeCrAl, nickel alloys and zirconium alloys). Irradiated byproducts such as control rods and/or control blades containing either hafnium or boron carbide. The minimum cooling time for either irradiated hardware or irradiated byproducts shall be at least 30 days prior to shipment.

(ii) ^{60}Co as either normal form rods, normal form encapsulated pellets or special form.

(iii) GE BWR 10x10 irradiated fuel rods with the following characteristics:
1. a minimum active fuel height of 5.3 inches,
2. a minimum pellet diameter of 0.784 cm,
3. a minimum cooling time of 120 days prior to shipment
4. a maximum U-235 mass of 1750 grams,
5. a maximum burnup of 72 GWd/MTU, and
6. an initial U-235 enrichment between 1.5 wt% and 5 wt%.

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

(i) For the contents described in 5(b)(1)(i), the maximum quantity of material shall not exceed the limits specified in Section 7.5.1 of the safety analysis report.

(ii) For the contents described in 5(b)(1)(ii), the maximum quantity of material shall not exceed the limits specified in Section 7.5.2 of the safety analysis report, and the total activity in any axial 1-inch increment shall be less than or equal to 17,000 Curies.

(iii) For the contents described in 5(b)(1)(iii), the maximum quantity of material shall not exceed the limits specified in Section 7.5.3 of the safety analysis report.

(iv) For a combination of contents described in 5(b)(1)(i), 5(b)(1)(ii) and 5 (b)(1)(iii), the maximum quantity of material shall not exceed the limits specified in Section 7.5.4 of the safety analysis report.

(v) The contents described in 5(b)(1)(i) and 5(b)(ii) may contain fissile material provided the quantity of material does not exceed the exempt quantity under 10 CFR 71.15.

(vi) The thermal heat load of the package shall not exceed 1500 W.

(vii) The combined weight of the HPI, HPI basket, radioactive material, shoring, and secondary containers shall not exceed 5,450 lbs.

5.(c) Criticality Safety Index: 50.0

6. The HPI shall be used to transport contents 5(b)(1)(i), 5(b)(1)(ii) and 5 (b)(1)(iii).

7. The HPI and the HPI material basket shall be used to transport content 5(b)(1)(ii) and 5 (b)(1)(iii).

8. Appropriate shoring must be provided as necessary to minimize content movement during accident conditions of transport.

9. The package shall be shipped in a vertical orientation.

10. Air transport is not authorized.

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

    (a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7.0 of the application, as supplemented.

    (b) The package must meet the Acceptance Tests and Maintenance Program in Section 8.0 of the application, as supplemented.

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

13. Revision No. 27 of this certificate may be used until April 30, 2021.

14. Expiration date: March 31, 2023.

<u>REFERENCES</u>

GE Hitachi Nuclear Energy Company application dated April 28, 2016.

Supplements dated: May 4, 2016; June 13, and September 29, 2017; January 9, and February 27, 2018; July 31, 2019; January 31, and April 2, 2020.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John B. McKirgan Digitally signed by John B. McKirgan
Date: 2020.04.23 12:46:45 -04'00'

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

Date: April 23, 2020

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

SAFETY EVALUATION REPORT

Docket No. 71-9228
Model No. 2000
Certificate of Compliance No. 9228
Revision No. 28

SUMMARY

By letter dated July 31, 2019, as supplemented on January 31, and April 2, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML19212A592, ML20031C697 and ML20093B448) respectively), GE-Hitachi Nuclear Energy Americas, LLC submitted an amendment request to revise the certificate of compliance for the Model No. 2000 package. The applicant submitted a revised shielding analysis and a criticality analysis to reintroduce GE Boiling Water Reactor (BWR) 10x10 irradiated fuel as an approved content. Nuclear Regulatory Commission staff (the staff) reviewed the application using the guidance in NUREG -1609, "Standard Review Plan for Transportation Packages for Radioactive Material" and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." Based on the statements and representations in the application, as supplemented, the staff finds that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

In addition to various editorial and administrative changes, the applicant requested that GE BWR 10x10 irradiated fuel be reintroduced as an approved content. The applicant provided the following characteristics of the irradiated fuel: minimum fuel length, initial enrichment, maximum burnup, minimum cooling time and maximum uranium 235 (^{235}U) mass. Because the applicant did not specify the cladding type, staff asked the applicant during a teleconference to clarify if the purpose of the amendment was to transport accident tolerant fuel. The applicant confirmed that the purpose of the amendment was to authorize transport of accident tolerant fuel (ADAMS Accession No. ML19269E524).

The applicant chose to treat irradiated fuel cladding as irradiated hardware, but the applicant left the definition of irradiated hardware unchanged. Staff questioned the applicant if the irradiated hardware definition encompassed FeCrAl which was one of the cladding types discussed in pre-application meetings (ADAMS Accession No. ML20105A279). Based upon the conversation, the applicant chose to revise the irradiated hardware description to encompass the FeCrAl accident tolerant fuel cladding. After reviewing the information, the staff has reasonable assurance that the irradiated fuel content has been adequately described.

The applicant also requested to change the quality assurance plan references in the safety analysis report (SAR). The SAR referenced "Quality Assurance Program for Shipping Packages for Radioactive Material (Docket 71-0170)," QAP-1, which is issued to the Vallecitos Nuclear Center in Sunol, California. The applicant wanted to revise the SAR to reference "GE Hitachi Nuclear Energy Quality Assurance Program Description (Docket 71-0254)," NEDO-11209-A which is issued to GE-Hitachi Nuclear Energy Americas, LLC in Wilmington, N.C. The applicant discussed this requested change with the staff on March 27, 2020 (ADAMS Accession No. ML20094L153). Based upon the information presented during the phone call, and because NEDO-11209-A is currently approved by the NRC under 10 CFR Part 71, the staff has reasonable assurance that NEDO-11209-A satisfies the requirements of 10 CFR 71, Subpart H.

Based on a review of the statements and representations in the application, the staff concludes that the contents have been adequately described to meet the requirements of 10 CFR Part 71.

2.0 STRUCTURAL

In this amendment, the applicant reintroduced irradiated GE BWR 10x10 fuels rods as radioactive contents and modified the Irradiated Hardware and Byproducts content to encompass the metallic alloy FeCrAl being developed for use with accident tolerant fuels.

2.1 General

2.1.1 Source Specification

Irradiated Fuel:

The applicant defined the irradiated fuel content as GE BWR 10x10 fuel with the following characteristics:

- Cooling time: minimum of at least 120 days,
- Length: minimum active fuel length of at least 5.3 inches for each segment,
- Arrangement: placed into the High Performance Insert material basket in an upright position with or without additional shoring that ensures the fuel remains upright,
- Initial ^{235}U enrichment: minimum of 1.5 wt% and a maximum of 5 wt%,
- Burnup: maximum of 72 GWd/MTU, and
- cladding composed of an approved irradiated hardware material below.

Irradiated Hardware and Byproducts:

The applicant revised the definition of Irradiated Hardware and Byproducts to more generically identify the irradiated hardware and byproduct materials as metallic alloys. The applicant also added FeCrAl as a metallic alloy example.

The staff has reviewed the content description and finds that it meets the requirements of 10 CFR 71.31.

2.1.2 Material Properties

The applicant corrected the Type 316 stainless steel yield strength used in the structural analysis from 16,900 psi to 17,700 psi. The staff determined that this change is acceptable based on a review of ASME BPVC Section II Part D and Metals Handbook. The staff previously reviewed the codes and standards for the package design and find that they are acceptable.

The applicant provided no data for the cladding properties. However, staff determined that this was acceptable because the criticality and shielding safety analyses did not rely on cladding integrity for maintaining fuel pellet configuration.

2.1.3 Chemical and Galvanic Reactions

To the maximum credible extent, the applicant followed the vacuum drying process in compliance with the ASTM Standard Guide for Drying Behavior of Spent Nuclear Fuel (ASTM C1553-08). Based upon the materials used to fabricate the Model No. 2000 and the fact that the applicant followed ASTM C1553-08 to the maximum credible extent, staff finds there is no significant chemical, galvanic or other reactions for each packaging component, among package contents, or between the packaging components and the contents in dry or wet environment conditions.

2.1.4 Radiation Effects

The applicant previously evaluated the radiation effects on seals and determined the maximum exposure occurred with the Model No. 2000 fully loaded with Cobalt-60 rods at the maximum activity. Regarding the potential embrittlement by radiation, the package containment design utilized materials that meet the requirement of Regulatory Guides 7.11 and 7.12 for embrittlement. Based upon a review of the application, the staff finds that gamma radiation has no significant effect on seals and metals.

2.2 Evaluation Findings

Based on a review of the statements and representations in the application, the NRC 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.

3.0 THERMAL EVALUATION

Staff reviewed the proposed change and determined that it did not impact the staff's previous SER findings regarding the package thermal design. Therefore, the staff finds that a new evaluation is not needed.

4.0 CONTAINMENT EVALUATION

Staff reviewed the proposed change and determined that it did not impact the staff's previous SER findings regarding the package containment design. Therefore, the staff finds that a new evaluation is not needed.

5.0 SHIELDING EVALUATION

The applicant submitted a request for amendment of CoC 71-9228 for the Model No. 2000 transportation package with modified contents and packaging design. The applicant requested to reintroduce irradiated BWR fuel, limited to the GNF BWR 10x10 design, as authorized contents. The other contents and the packaging remained unchanged from the previously approved CoC. The staff reviewed the application to verify the new proposed contents will meet the external radiation limit requirements of 10 CFR 71. Staff performed its shielding review using NUREG-1609 "Standard Review Plan for Transportation Packages for Radioactive Material" and NUREG-1617 "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

5.1 Shielding Design Description

5.1.1 Design Features

The Model No. 2000 packaging design consisted of an overpack, a cask, and a high-performance insert (HPI). These features remained unchanged from CoC 71-9228, Revision 27 (ADAMS Accession No. ML18102B443).

5.1.2 Summary Table of Maximum Radiation Levels

The applicant presented a summary of maximum calculated normal conditions of transport (NCT) and hypothetical accident conditions (HAC) dose rates in SAR Tables 5.1-2 and 5.1-3 respectively. The applicant calculated the dose rates for the requested contents at the locations prescribed by regulation. For NCT, those locations are the package surface, two meters from the package side and an occupied position (i.e., cab). For HAC, the applicant showed the dose rates one meter from the package surface. Staff reviewed the tables and confirmed the calculated dose rates meet the regulatory limits in 10 CFR 71.47 and 71.51(a)(2).

5.2 Radiation Source

5.2.1 Irradiated BWR Fuel

The applicant determined the gamma and neutron source term for this content type based on burnup and initial enrichment. The applicant used ORIGEN-ARP in the SCALE 6.1 code suite with the GE BWR 10x10 reactor library included in the software distribution. ORIGEN-ARP uses ORIGEN-S to interpolate from pre-calculated libraries at an array of burnup, enrichment and moderator density combinations. Because the ORIGEN-ARP code has been evaluated by comparing measured and computed spent fuel isotopic compositions for LWR systems ("ARP: Automatic Rapid Process for the Generation of Problem-Dependent SAS2H/ORIGEN-S Cross-Section Libraries," 1998, ORNL/TM-13584), staff finds its use here acceptable. Staff reviewed the assembly design characteristics for the GE BWR 10x10 library and finds its use acceptable since the design is very close to the GNF BWR 10x10. Staff reviewed the required parameters for irradiated fuel in Section 5.2 of the application and verified it falls within the burnup and enrichment bounds of the library. The applicant considered cladding and non-fuel hardware as irradiated hardware which is discussed in the next section.

The applicant split the burnup and enrichment range into bands. Within each band, the applicant used the minimum enrichment and maximum burnup. The applicant also selected secondary parameters (e.g., moderator density and specific power) that result in a bounding calculated source term for each band. Because this follows the guidance in NUREG-1617, staff finds it acceptable since it will result in the maximum calculated source term. The applicant normalized its calculated source term on a per-gram basis of ^{235}U. The applicant did this in order to determine the total source by multiplying the mass by the normalized source term. Since the applicant determined the spent fuel dose contribution from either a minimum-length line (NCT) or single point (HAC), staff finds this to be conservative as the calculated, bounding dose contribution from additional source mass will not decrease as the payload volume increases.

5.2.2 Irradiated Hardware and Byproducts

The applicant defined irradiated hardware as irradiated metals that are part of the fuel assembly or reactor internals (e.g., fuel cladding, spacers, tie plates). Irradiated byproducts consisted of irradiated control blades with either hafnium or boron carbide neutron poison. Except for the fuel cladding, the irradiated hardware and byproduct source term calculations remained unchanged from the CoC 71-9228, Revision 27 (ADAMS Accession No. ML18102B443). In their prior analysis, the applicant evaluated several materials with ORIGEN-S to determine isotopes that contribute to external dose. The loading procedure described in SAR Chapter 7 required the activity of the isotopes of concern to be known before loading. The applicant evaluated the dose contribution from a series of isotopes. Provided the irradiated hardware and byproduct activation composition can be found in Sections 5.4 and 5.5 of the SAR, the applicant's dose calculation will be applicable. Staff finds reasonable assurance the loading instructions will determine whether irradiated hardware, including cladding, and byproducts fall under existing evaluation.

5.2.3 Combined Contents

Since the applicant defined the cladding to be irradiated hardware and byproduct material, the applicant considered irradiated BWR fuel with cladding to be combined contents. The applicant provided an example of a loading table for combined contents in SAR Table 5.5-42. NRC staff previously reviewed the applicant's procedure to determine if combined contents will meet the 1500 W thermal limit and the dose rate limits in 10 CFR 71.47 or 10 CFR 71.51(a)(2) and found it acceptable.

5.3 Shielding Model Specification

5.3.1 Source and Shielding Configuration

Staff confirmed the applicant used dimensions in their shielding model consistent with those in the Model No. 2000 packaging drawings. The applicant used minimum dimensions for the HPI model except for the HPI cavity. Staff finds this acceptable since minimizing the shielding material will maximize calculated dose rates, and small changes in source location within the HPI cavity inside of the shielding will have negligible effect on external dose rates. The applicant generally used nominal dimensions to model the cask body and overpack. However, if there was a significant variation in thickness over a component, the applicant used the minimum thickness throughout that component. Staff finds the use of minimum thickness acceptable since it will underestimate shielding and maximize calculated dose rates. The applicant's use of nominal dimensions for other cask body and overpack components is not a conservative modeling assumption. However, staff evaluated the applicant's other conservative assumptions (e.g., use of point and line source configurations) and finds the use of nominal dimensions acceptable due to the margin provided from those assumptions which is discussed in more detail in SER. Section 5.5

For NCT, the applicant determined the gamma and neutron dose rates separately. The applicant used the same geometry for both models. The applicant conservatively omitted the basket and rod holders from the NCT model. For the gamma dose rate, the applicant used the material properties for the HPI, cask body, and overpack given in Tables 5.3-2 through 5.3-5. Staff reviewed the material properties and finds the isotopic composition and density appropriate for the materials used. For the neutron dose rate, the applicant assumed the geometry spacing is maintained, but modeled all materials as void (i.e., source is unshielded). Staff finds this assumption acceptable since it removes all shielding material from the model and results in maximum calculated dose rates.

For HAC, the applicant considered the HPI and Model No. 2000 cask body. However, the applicant omitted the overpack. Staff finds this acceptable since it maximizes calculated dose rates by reducing distance to the source and removing shielding material. The applicant also assumed 4 mm lead slump which is greater than the maximum 3.56 mm lead column deformation determined in SAR Section 2.12.2. In SAR Section 2.12.2, the applicant also determined that the overpack sufficiently protects the cask body and HPI with no gross deformations. As a result, staff finds the applicant's use of NCT dimensions for the cask body and HPI under HAC acceptable. The applicant used the same material properties for the HAC gamma dose model and omitted all material for the HAC neutron dose model. Staff finds these assumptions acceptable for the same reasons identified for NCT.

5.3.2 Source Location

The applicant modeled the spent fuel source as a single line source of minimum rod segment length permitted in the CoC under NCT. For HAC, the applicant modeled the spent fuel source term as a single point. For all other contents, the applicant modeled the source as a single point source. For both the point and line sources, the applicant performed evaluations with the source in different locations for each surface dose rate location: top, side, or bottom. For the line sources, the applicant placed the source in the HPI cavity position that results in the highest dose rate for the respective regulatory dose rate location. Staff reviewed the applicant's locations and finds the applicant considered enough locations to determine the source position that yields the maximum dose rate.

5.3.3 Dose Rate Location Evaluations

The applicant provided the locations for calculating the dose rates under NCT in SAR Figure 5.3-4. The staff confirmed the locations used for evaluating dose rates under NCT are consistent with the locations prescribed in 10 CFR 71.47(b).

The applicant provided the locations for calculating dose rates under HAC in SAR Figure 5.3-5. Since the applicant took no credit for the presence of the impact limiter, the applicant located the tally locations under HAC relative to the cask body. Staff confirmed the locations used for evaluating dose rates under HAC are consistent with the distances prescribed in 10 CFR 71.51(a)(2).

The applicant encompassed the entire exterior of the cask with one-centimeter thick tally cells. Prior staff review of CoC 71-9228, Revision 27 (ADAMS Accession No. ML18102B443) found the applicant's use of these relatively small tallies appropriate because they prevent a maximum calculated dose rate from being reduced due to averaging over a volume with lower dose rates.

5.3.4 Material Properties

The applicant modeled stainless steel, lead, and depleted uranium with the compositions and densities listed in SAR Section 5.3.2. The applicant modeled the HPI materials with densities having the minimum values specified in the SAR. Staff finds the material properties acceptable since they are consistent with the guidance provided in both NUREG-1609 and NUREG-1617.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant calculated photon and neutron dose rates in separate calculations. The applicant assumed no shielding material was present for the neutron dose rates and calculated these dose rates analytically. Since the applicant's neutron dose rate analysis relies simply on distance from a line or point source, the analysis proved straight-forward. Therefore, staff finds it acceptable. The applicant performed the gamma shielding calculations using MCNP6 which is a general-purpose Monte Carlo transport code. The applicant calculated photons and neutrons in separate calculations. The applicant used the MCPLIB84 photon transport data library which is based on ENDF/B-VI.8 nuclear data. Given the code's capabilities, its extensive and well-vetted use within the nuclear industry, the use of standard nuclear data libraries, and the code's acceptance by the staff for these types of applications, staff finds their use acceptable here.

The applicant used the MCNP model to effectively determine its own spent-fuel gamma response functions by tallying the dose contribution from reference sources (ADAMS Accession No. ML19134A257). Staff reviewed the applicant's reference sources and finds reasonable assurance the applicant achieved proper convergence across the spent-fuel source spectrum. The applicant then applied the scaled payload source term to its response function to determine external dose rate. NRC staff has approved this method in other spent-fuel storage casks and transportation packages. Therefore, staff finds the applicant's response function acceptable for the Model No. 2000 provided the following conditions are met:

- the bounding source location and length within the HPI cavity for the corresponding dose point remains unchanged,
- there are no changes to the material composition of the HPI, cask, or impact limiter, and
- there are no changes to HPI, cask, or impact limiter geometry.

5.4.2 Input and Output Data

Aside from the source, the applicant's gamma shielding analysis remained unchanged from the review of CoC 71-9228, Revision 27 (ADAMS Accession No. ML18102B443). Prior staff review found the applicant's MCNP inputs acceptable. Given the simple changes to the source configuration for the new content type, staff finds reasonable assurance that the applicant has acceptably performed its shielding analysis. The applicant provided the equations used for its neutron shielding analysis hand calculations. Since there is no shielding material present and the applicant relies entirely on geometry, staff finds the applicant's neutron dose rate calculations acceptable.

5.4.3 Flux-to-Dose-Rate Conversion Factors

The applicant used standard flux-to-dose rate conversion factors derived from the ANSI/ANS 6.1.1-1977 standard which is consistent with the recommendations in both NUREG-1609 and NUREG-1617. The applicant showed the factors used for the gamma and neutron doses in SAR Tables 5.4-1 and 5.4-2 respectively. Staff reviewed these factors and confirmed that they are appropriate.

5.4.4 Combined Contents

The applicant considered irradiated cladding as Irradiated Hardware & Byproduct and not as part of the irradiated fuel material. As a result, irradiated fuel rod segments with cladding will be considered combined contents. The applicant provided an example of a spent fuel loading table in SAR Section 5.5.5. After reviewing the example spent fuel loading table and SAR Sections 7.5.1, 7.5.3, and 7.5.4, staff determined that the applicant's procedures properly treat clad fuel. Therefore, staff finds the applicant's procedures appropriately account for the dose contributions from each of the contents.

5.4.5 External Radiation Levels

The applicant calculated external dose rates for the contents to ensure they meet both NCT and HAC requirements of 10 CFR 71.47(b) and 10 CFR 71.51(a)(2) respectively. The applicant applied a 2σ uncertainty to the calculated values to account for uncertainty with the MCNP code. The applicant determined σ by multiplying the relative uncertainty calculated by MCNP by the dose response function. Staff finds this acceptable because the applicant's conservative modeling assumptions yield calculated dose rates below the regulatory limits.

5.5 Staff Confirmatory Analysis

Staff modeled the steel, lead, and depleted uranium components for the cask and HPI. However, staff ignored all components internal to the HPI and modeled the HPI cavity as void to maximize dose rates by minimizing self-shielding. Staff used the MONCACO/MAVRIC code in the SCALE 6.2 code suite with 27 neutron and 19 gamma group cross section based on ENDF/B-VII nuclear data. Staff also allowed the code to sample source particles throughout the HPI cavity versus confining the source to a point or a line with a specific length in a given location. Staff results showed calculated dose rates increased, at a minimum, by a factor of about 2.5 when using a 5.3-inch line source versus sampling source particles throughout the HPI cavity. The dose rate increase proved to be even higher for a point source which the applicant used for the evaluation of Irradiated Hardware and Byproducts under NCT as well as all sources, including irradiated fuel, under HAC. Staff used this minimum margin to evaluate any non-conservative modeling assumptions made by the applicant and found sufficient margin existed between the calculated dose rates and the regulatory limits.

5.6 Findings

The staff performed its review following the guidance provided in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," and NUREG-1617 "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." Based on its review of the statements and representations in the application, the staff finds that applicant has adequately described and evaluated the shielding design of the package. Staff finds reasonable assurance that the package will meet the external radiation requirements of 10 CFR Part 71.

6.0 CRITICALITY

The applicant requested to modify the Model No. 2000 Certificate of Compliance to reintroduce irradiated GE BWR 10x10 fuel as allowable contents. The applicant limited the requested fuel mass to 1,750 grams ^{235}U enriched to a maximum of 5.0 weight percent. The applicant planned to ship fuel rods with a minimum pellet diameter of 0.784 cm.

The applicant modeled the BWR 10x10 fuel rods with varying height, pitch, and water density inside the HPI in the Model No. 2000 package internal cavity in order to find the most reactive configuration. The applicant also varied pellet diameter above the 0.784 cm minimum. In addition, the applicant conservatively ignored the fuel rod cladding and any other structural material which may be present inside the HPI cavity.

The applicant used the same single package model for both NCT and HAC evaluations. The applicant assumed that the package is optimally moderated by water in the HPI, and that full density water filled the void spaces in the rest of the packaging. This single model satisfies the requirement in 10 CFR 71.55(b) that the applicant consider water moderation to the most reactive credible extent.

For the NCT array model, the applicant assumed optimum water moderation in the HPI cavity, and void in all the other empty regions of the packaging. Staff finds these assumptions acceptable because structural evaluations show that the package leakage under NCT is not credible. The applicant modeled seven packages in a hexagonal array with optimum water moderation between the packages.

For the HAC array model, the applicant assumed optimum moderation by water in the HPI cavity, and varied the moderation in the void spaces of the packaging to find the most reactive condition. The applicant modeled two packages in an array, with optimum moderation by water between the packages. Based on the array evaluations under NCT and HAC, the applicant determined the resulting criticality safety index for the package with irradiated GE 10x10 BWR fuel contents to be 50.0.

The staff reviewed the applicant's resulting most reactive configurations under NCT and HAC, for both the single package and package arrays. The staff finds that the applicant has identified the most reactive package and content configurations under all conditions.

The applicant used Versions 1.0 and 2.0 of MCNP6, a general purpose Monte Carlo radiation transport code, with the continuous-energy ENDF/B-VII.1 neutron cross section library for the calculations to find the most reactive condition. Because the MCNP code system is a standard tool in the nuclear industry for performing Monte Carlo criticality safety and radiation shielding calculations, staff finds its use for this application acceptable.

The applicant summarized the criticality calculation results in SAR Table 6.1.2-1. Staff reviewed the k_{eff} values reported in SAR Table 6.1.2-1 and found that the highest reported k_{eff} value is 0.9256. Staff finds this is acceptable because it is below the upper subcritical limit (USL) of 0.9370 which the applicant calculated from their benchmarking analysis as discussed below.

The applicant performed a benchmarking analysis for the package modeled with MCNP6, Version 1.0, and the continuous-energy ENDF/B-VII.1 cross section library. The applicant benchmarked the Model No. 2000 against 36 critical experiments containing low enriched UO_2 moderated by water. The experiments ranged in enrichment from 2.35 to 4.306 weight percent ^{235}U. The applicant determined a USL using the guidance in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages." Following this guidance, the applicant determined the USL to be 0.9370. The applicant also demonstrated that the benchmarking analysis k_{eff} results were normally distributed, and evaluated the results for trends against various experimental parameters. The hydrogen to ^{235}U ratio proved to be the most significant trend, and the applicant used this trend to define a function to determine the USL for the analysis of the Model No. 2000 package.

The staff determined that the applicant's benchmarking analysis followed available benchmarking guidance in NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," with two exceptions: 1) the applicant performed a benchmarking analysis on a code version different from the one used to determine the most reactive configuration; and 2) the enrichment of the most reactive configuration is outside the range of applicability of the benchmarking analysis, with no adjustment to the USL for increased bias uncertainty. However, the staff determined that the applicant's criticality results generated using the MCNP6, Version 2.0 code with the continuous-energy ENDF/B-VII.1 cross section library are acceptable for the following reasons:

1. MCNP6 Version 2.0, used for the applicant's most reactive configuration analysis, is very similar to MCNP6 Version 1.0, used in the benchmarking analysis. The applicant also referenced a report by the MCNP developer demonstrating that the two code systems provide statistically similar k_{eff} results when using the same cross section library (ADAMS Accession No. ML20031C704 and ML20093B468).

2. Based on staff experience with similar low-enriched UO_2 systems, the applicant's calculated USL is conservative (i.e., low).

3. There is significant margin (greater than 1% in k_{eff}) between the applicant's maximum calculated k_{eff} and the USL.

4. Staff confirmatory analyses using a different code system provides k_{eff} results that are either statistically the same or bounded by the applicant's results.

The staff performed a confirmatory analysis of the Model No. 2000 package with GE BWR 10x10 fuel contents using the SCALE 6.2.3 Monte Carlo radiation transport code. The staff used the CSAS6 criticality sequence and the ENDF/B-VII.1 continuous-energy neutron cross section library in their analysis. In modeling the package, the staff used assumptions similar to those of the applicant. The staff's independent evaluation resulted in k_{eff} values that were similar to, or bounded by, the applicant's results.

The staff reviewed the applicant's requested changes to the Certificate of Compliance, initial assumptions, model configurations, analyses, and results. The staff finds that the applicant has identified the most reactive configuration of the Model No. 2000 with the requested contents, and that the criticality results are conservative. Therefore, the staff finds with reasonable assurance that the package, with the requested contents, will meet the criticality safety requirements of 10 CFR Part 71.

7.0 PACKAGE OPERATIONS

The applicant added a new SAR section 7.5.3 for loading irradiated fuel. The applicant provided instructions for completing a loading table which would document the fuel rod length, initial enrichment, burnup and ^{235}U mass. The instructions directed how the package user would calculate the thermal load associated with irradiated fuel. The instructions also directed how the package user would calculate the dose rate at various locations including the top, side and bottom of the package, two meters from the package surface as well as occupied spaces. Because the amendment added irradiated fuel as authorized content, the applicant modified references in SAR sections 7.5.1 and 7.5.2. In addition, the applicant revised SAR section 7.5.4 to address the irradiated fuel content. Based on a review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

Staff reviewed the proposed change and determined that it did not impact the staff's previous SER findings regarding the package acceptance tests and maintenance program. Therefore, the staff finds that a new evaluation is not needed.

CONDITIONS

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

The package identification number was changed to B(U)F to designate the package as authorized to transport fissile material.

Condition 5(b)(1)(i) was modified to identify irradiated hardware to metallic alloys and to include the metallic alloy FeCrAl.

Condition 5(b)(1)(iii) was added.

New Condition 5(b)(2)(iii) was added and subsequent Conditions were renumbered 5(b)(2)(iv) thru 5(b)(2)(vii).

New Condition 5(b)(2)(iv) was modified to address new content 5(b)(1)(iii) and to recognize the new SAR section for the combined content loading table.

New Condition 5(c) was added and subsequent Conditions were renumbered.

New Condition 6 and new Condition 7 were modified to address new content 5(b)(1)(iii).

The references section has been updated to include this request.

Minor editorial corrections were made.

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. 2000 package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9228, Revision No. 28
on April 23, 2020.