CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9330	7	71-9330	USA/9330/AF-96	1 OF	6

2.. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

U.S. Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

U.S Department of Energy consolidated application dated June 23, 2011, 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.: ATR FFSC

(2) Description

An insulated stainless steel package for the transport of unirradiated research reactor fuel, including intact fuel elements or fuel plates. The packaging consists of (1) a body, (2) a closure lid, and (3) inner packaging internals. The approximate dimensions and weights of the package are:

Overall package outer width and height	8 inches
Overall package length	73 inches
Cavity diameter	5-3/4 inches
Cavity length	68 inches
Packaging weight (without internals)	240 pounds
Maximum package weight (including internals and contents)	290 pounds

The body is composed of two thin-walled, stainless steel shells. The outer shell is a square tube with an 8-inch cross section, a 73-inch length, and a 3/16 inch wall thickness. The inner shell is a round tube with a 6-inch diameter and a 0.120-inch wall thickness. The inner tube is wrapped with ceramic fiber thermal insulation, overlaid with a stainless steel sheet. At the bottom end, the shells are welded to a 0.88-inch thick stainless steel base plate. At the top end (closure end), the shells are welded to a 1.5-inch thick stainless steel flange.

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

The closure is composed of circular stainless steel plates with ceramic fiber insulation. The closure engages the top end flange by way of four bayonets that are rotated and secured by two spring pins. The closure is equipped with a handle, which may be removed during transport. The closure does not have a gasket or seal.

The package internals consist of either a Fuel Handling Enclosure (FHE) for intact Advanced Test Reactor (ATR), Massachusetts Institute of Technology (MIT), University of Missouri Research Reactor (MURR), or Rhode Island Nuclear Science Center (RINSC) fuel elements and Small Quantity Payloads, or a Loose Fuel Plate Basket for ATR fuel plates. The RINSC, MIT, MURR, and Small Quantity Payload FHE use ball lock pins and end spacers to lock closed while the ATR FHE uses a spring plunger.

(3) Drawings

The packaging is constructed and assembled in accordance with the following Areva Federal Services LLC. or Packaging Technology, Inc., Drawing Nos.:

60501-10, Sheets 1-5, Rev. 3	ATR Fresh Fuel Shipping Container SAR Drawing
60501-20, Rev. 1	ATR Loose Fuel Plate Basket
60501-30, Rev. 1	ATR Fuel Handling Enclosure
60501-40, Rev. 0	MIT Fuel Handling Enclosure
60501-50, Rev. 0	MURR Fuel Handling Enclosure
60501-60, Rev. 0	RINSC Fuel Handling Enclosure
60501-70, Rev. 0	Small Quantity Payload Fuel Handling Enclosure

(b) Contents

(1) Type and form of material

Unirradiated Mark VII ATR fuel. The ATR fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum 94 weight percent U-235; the maximum U-234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Intact ATR fuel elements contain 19 curved fuel plates fitted within aluminum side plates, and the maximum channel thickness between fuel plates is 0.087 inch. The fuel meat thickness is a nominal 0.02 inch for all 19 plates, and the fuel meat width ranges from approximately 1.5 inches to 3.44 inches. The nominal active fuel length is approximately 48 inches. The maximum mass of U-235 per intact ATR fuel element is 1200 grams. The ATR fuel element must be contained within the ATR Fuel Handling Enclosure, as specified in 5.(a)(3).

Unirradiated ATR U-Mo fuel elements. The ATR U-Mo fuel element consists of a mixture of high-enriched uranium aluminide (UAl_x) fuel plates and low-enriched uranium and molybdenum alloy (U-Mo) fuel plates, with a maximum mass of U-235 per U-Mo fuel element of 1,240 grams. The ATR U-Mo fuel element contains 19 curved plates fitted within

5.(b)(1) Type and Form of Material (continued)

aluminum side plates; plates 1 through 4, and 16 through 18, contain high-enriched UAl_x fuel; plates 5 through 15 contain low-enriched U-Mo fuel; and plate 19 is an aluminum alloy plate. The maximum channel thickness between fuel plates is 0.087 inch. For the high-enriched UAl_x fuel plates, the uranium is enriched to a maximum 94 weight percent U-235; the maximum U-234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. For the low-enriched U-Mo fuel plates, the molybdenum content is a nominal 10 weight percent; the uranium is enriched to a maximum 20 weight percent U-235; the maximum U-234 content is 0.26 weight percent; and the maximum U-236 content is 0.46 weight percent. For the high-enriched UAl_x fuel plates, the fuel meat thickness is a nominal 0.02 inch; the fuel meat width ranges from approximately 1.5 inches to 3.44 inches; and the nominal active fuel length is approximately 48 inches. For the low-enriched U-Mo fuel plates, the fuel meat thickness is a nominal 0.013 inch, with a nominal 0.001 inch thick zirconium interlayer present between the fuel meat and the aluminum cladding layer; the fuel meat width ranges from approximately 2.25 inches to 3.28 inches; and the nominal active fuel length is approximately 48 inches. The ATR U-Mo fuel element must be contained within the ATR Fuel Handling Enclosure, as specified in 5.(a)(3).

Unirradiated MIT fuel element. The MIT fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum of 94 weight percent U-235; the maximum U- 234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Each MIT fuel element contains 15 flat fuel plates fitted within aluminum side plates and the maximum channel thickness between fuel plates is 0.090 inch. The fuel meat thickness is a nominal 0.03 inch for all 15 plates and the fuel meat width ranges from approximately 1.98 inches to 2.17 inches. The nominal active fuel length is 22.375 inches. The maximum mass of U-235 per intact MIT fuel element is 515 grams. The MIT fuel element must be contained within the MIT Fuel Handling Enclosure, as specified in 5.(a)(3).

Unirradiated MURR fuel element. The MURR fuel material is composed of uranium aluminide (UAl_x). The uranium is enriched to a maximum of 94 weight percent U-235; the maximum U- 234 content is 1.2 weight percent; and the maximum U-236 content is 0.7 weight percent. Each MURR fuel element contains 24 curved fuel plates fitted within aluminum side plates and the maximum channel thickness between fuel plates is 0.090 inch. The fuel meat thickness is a nominal 0.02 inch for all 24 plates and the fuel meat width ranges from approximately 1.71 inches to 5.72 inches. The nominal active fuel length is 24 inches. The maximum mass of U-235 per intact MURR fuel element is 785 grams. The MURR fuel element must be contained within the MURR Fuel Handling Enclosure, as specified in 5.(a)(3).

Small Quantity Payloads (RINSC fuel elements, ATR Full-size plate In Flux trap Position (AFIP) elements, U-Mo foils, Design Demonstration Elements (DDEs) and similar test elements, MIT loose fuel element plates, or MURR loose fuel element plates) where the maximum mass of U-235 is 400 grams and maximum U-235 enrichment is 94 weight percent. Aluminum plates, shapes, and sheets, and miscellaneous steel or aluminum fasteners may be used as dunnage to fill gaps between the small quantity payloads and the

5.(b)(1) Type and Form of Material (continued)

small quantity FHE. 1/8” neoprene strips may be used between the small quantity FHE and small quantity payloads and/or between the optional aluminum dunnage and the small quantity payload. The 1/8” neoprene strips shall not be stacked in more than two layers between the small quantity payload and any interior face of the small quantity FHE.

Unirradiated RINSC fuel element. The RINSC fuel material is composed of uranium silicide (U_3Si_2) dispersed in aluminum powder. The uranium is enriched to a maximum of 20 weight percent U-235; the maximum U- 234 content is 0.5 weight percent; and the maximum U-236 content is 1.0 weight percent. Each RINSC fuel element contains 22 flat fuel plates fitted within aluminum alloy side plates and the maximum channel thickness between fuel plates is 0.096 inch. The fuel meat thickness is a nominal 0.02 inch for all 22 plates. The maximum mass of U-235 per intact RINSC fuel element is 283 grams. The RINSC fuel element must be contained within the RINSC Fuel Handling Enclosure, as specified in 5.(a)(3).

AFIP fuel element. The AFIP fuel element is composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy coated with a thin zirconium interlayer. The uranium is enriched to approximately 20 weight percent U-235. Each AFIP element contains 4 curved fuel plates fitted within 6061 aluminum side plates. The maximum mass of U-235 AFIP element is 365 grams. Loose plates from an AFIP fuel element are also permitted. The AFIP fuel element must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

U-Mo Foils. The U-Mo foils are composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy and may contain a zirconium coating. The uranium is enriched to a maximum of 94 weight percent U-235. The maximum mass of U-235 is 160 grams. More than one U-Mo foil type may be transported at a time. The U-Mo foils must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

DDEs and similar test elements. The DDEs and similar test elements are composed of uranium molybdenum alloy in an aluminum-silicon matrix or uranium molybdenum alloy. The uranium is enriched to a maximum of 94 weight percent U-235. The maximum mass of U-235 is 365 grams. Loose plates from a DDE or similar test element are also permitted. The DDEs or similar test elements must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

MIT and MURR loose fuel element plates. MIT and MURR loose plates may either be flat or curved and may be banded or wire-tied in a bundle. The MIT and MURR loose plate payload is limited to 400 grams of U-235. The approximate mass of U-235 of each MIT fuel plate is 34.3 grams. The approximate mass of U-235 per each MURR fuel plate is 19 to 46 grams. A mixture of MIT and MURR fuel plates may be

5.(b)(1) Type and Form of Material (continued)

shipped together. The fuel plates must be contained within the Small Quantity Payload Fuel Handling Enclosure, as specified in 5.(a)(3).

ATR loose fuel plates: ATR loose plates may either be flat or curved and may be banded or wire-tied in a bundle. The ATR loose plate payload is limited to 600 grams of U-235. Additional aluminum plates may be used as dunnage to fill gaps between the fuel plates and the basket payload cavity. The fuel plates must be contained within the ATR Loose Fuel Plate Basket, as specified in 5.(a)(3).

(2) Maximum quantity of material per package

The maximum total weight of contents and internals, including dunnage and other secondary packaging, is 50 lbs. Radioactive contents are not to exceed a Type A quantity.

For intact ATR, ATR U-Mo, MURR, RINSC, and MIT fuel elements: One fuel element.

For ATR loose fuel plates: A maximum of 600 grams U-235.

For Small Quantity Payloads: A maximum of 400 grams U-235.

(c) Criticality Safety Index (CSI):

For ATR, ATR U-Mo, MURR, MIT fuel elements or ATR loose fuel plates:	4.0
For Small Quantity Payloads:	25

6. Fuel elements and fuel plates may be bagged or wrapped in polyethylene. The maximum weight of the polyethylene wrap shall not exceed 100 grams per package.

7. Types of small quantity payloads cannot be mixed in a single Fuel Handling Enclosure.

8. Air transport of fissile material is not authorized.

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

(a) The package must be loaded and prepared for shipment in accordance with the Package Operations in Section 7 of the application.

(b) The package must be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.

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

11. Revision No. 6 of this certificate may be used until May 30, 2014.

12. Expiration date: May 30, 2014.

REFERENCES

U.S. Department of Energy consolidated application dated June 23, 2011, as supplemented: August 18, 2011; January 10, and December 20, 2012; and March 20, July 23, and September 15, 2013.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Michele Sampson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: December 12, 2013

SAFETY EVALUATION REPORT
Docket No. 71-9330
Model No. ATR FFSC Package
Certificate of Compliance No. 9330
Revision No. 7

SUMMARY

By letter dated December 20, 2012, as supplemented March 20, and September 15, 2013, the U.S. Department of Energy (DOE) requested revision of Certificate of Compliance (CoC) No. 9330, for the Model ATR FFSC package to include U-Mo demonstration elements as authorized contents of the package. The staff performed its review of the application, as supplemented, using the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

Based on the statements and representations in the application, as supplemented, the U.S. Nuclear Regulatory Commission (NRC) staff agrees that these changes do not affect the ability of the package to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging Description

The Advanced Test Reactor Fresh Fuel Shipping Container (ATR FFSC) transportation package is a Type A fissile material package, used to transport a single unirradiated research reactor fuel element. The package is designed to transport fuel element plates that have either not yet been assembled into a fuel element or have been removed from an unirradiated fuel element.

The applicant has requested to include ATR U-Mo fuel demonstration elements as authorized contents of the package. The applicant has not requested any changes to the packaging design in connection with this request. The ATR U-Mo demonstration element is essentially identical in size and shape to the Mark VII YA ATR fuel element already authorized for shipment in the certificate of compliance. During shipment, the ATR U-Mo demonstration element will be placed inside the same ATR Fuel Handling Enclosure (FHE) used for Mark VII ATR fuel elements. The ATR FHE is a cover used to protect the fuel from handling damage during ATR FFSC loading and unloading operations. The applicant did not request any changes to the ATR FHE's design.

1.2 Package Drawings

The applicant did not request any changes to the packaging design drawings in connection with this revision request.

1.3 Contents

The applicant requested a revision to the package's certificate of compliance to include U-Mo demonstration elements as authorized contents of the package. The external geometry of the ATR U-Mo demonstration element is essentially identical to the ATR Mark VII YA fuel element. The U-Mo demonstration element contains 18 fueled plates, while plate 19 is an aluminum alloy plate. It contains a mixture of U-Al_x (high-enriched uranium [HEU]) and U-Mo (low-enriched uranium [LEU]) fuel plates, with a maximum U-235 mass of 1,240 g. Plates 1 through 4 and 16 through 18 are UAl_x plates identical in construction and composition to a standard HEU ATR fuel element. Boron is included in the UAl_x plates as a burnable poison. Plates 5 through 15 are fueled with an alloy of LEU uranium and molybdenum. The U-Mo fuel meat is nominally 10% molybdenum by weight, and the U-235 is enriched up to 20.0%. For the LEU fuel, the maximum weight percent for U-234 and U-236 are 0.26% and 0.46%, respectively.

The U-Mo fuel matrix is nominally 0.013-in thick, and a nominal 0.001-in thick zirconium layer is present between the fuel matrix and the aluminum cladding. The fuel element weighs not more than 32 lbs. and is enclosed within the ATR FHE, weighing 15 lbs.

2.0 STRUCTURAL EVALUATION

The applicant requested a revision to the certificate of compliance to include ATR U-Mo demonstration elements as authorized contents of the package. These elements are essentially the same in geometry as the previously authorized ATR element payload. The weight of the ATR U-Mo element plus the handling enclosure is approximately 47 lbs. The total loaded package containing ATR U-Mo elements is 287 lbs. These weights are both bounded by the previously authorized payload maximum of 50 lbs, and the maximum authorized loaded package weight of 290 lbs. Accordingly, the staff finds that the inclusion of ATR U-Mo demonstration element as authorized contents of the package assures that existing structural analyses in the application remain valid. The staff finds that these changes do not affect the ability of the package to meet the structural requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The staff reviewed the ATR FFSC application to verify that the thermal performance of the package has been adequately evaluated for the tests specified under normal conditions of transport (NCT) and hypothetical accident conditions (HAC) and that the package design satisfies the thermal requirements of 10 CFR Part 71. The staff's review considered the guidance provided in Section 3 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," as well as associated Interim Staff Guidance (ISG) documents.

3.1 Description of Thermal Design

The primary heat transfer mechanisms of the ATR FFSC are conduction and radiation inside the package and convection and radiation to the environment from exterior surfaces. The body and closure assemblies serve as the primary impact and thermal protection for the FHE or the Loose Fuel Plate Basket (LFPB) and their enclosed payloads of an ATR fuel element, ATR U-Mo demonstration element, or loose fuel plates. The FHE and LFPB provide additional thermal shielding of their enclosed payloads during the transient HAC event. The ATR FFSC body is a stainless steel weldment. It consists of two nested shells; the outer shell is fabricated of a square stainless steel tube, while the inner shell is fabricated from stainless steel tube. Three ribs are secured to the inner shell by fillet welds at four equally spaced intervals. The ribs are not mechanically attached to the outer shell. Instead, a nominal 0.06 inch air gap exists between the ribs and the outer shell, with a larger nominal gap existing at the comers of the ribs. These design features help to thermally isolate the inner shell from the outer shell during the HAC event. Further thermal isolation of the inner shell is provided by ceramic fiber thermal insulation which is wrapped around the inner shell between the ribs and by a stainless steel sheet used as a jacket material over the insulation. The stainless steel jacket maintains the insulation around the inner shell and provides a relatively low emissivity barrier to radiative heat exchange between the insulation and the outer sleeve. Thermal insulation is built into the bottom end closure plate of the packaging, while the ATR FFSC closure provides thermal insulation at the top end closure. The ATR FFSC closure incorporates ceramic fiber to provide thermal protection and is designed to permit gas to easily vent through the interface between the closure and the body.

3.2 Material Properties and Component Specifications

Material property tables for the ATR FFSC components are included in FSAR Section 3.2 and Tables 3.2-1 through 3.2-6. The materials present in the package include stainless steel, aluminum, ATR fuel plates, neoprene, ceramic fiber, and air. Thermal properties provided in the FSAR include thermal conductivity, density, heat capacity, viscosity, and emissivity. The temperature range for the material properties covers the range of temperatures encountered during the thermal analysis. The materials used in the ATR FFSC that are considered temperature sensitive are the aluminum used for the FHE, the LFPB, the ATR fuel, and the ATR U-Mo demonstration element, the neoprene rubber, and the polyethylene wrap used as a protective sleeve around the ATR fuel element and ATR U-Mo demonstration element. Only the aluminum used for the ATR fuel and ATR U-Mo demonstration element is considered critical to the safety of the package.

The staff finds the material properties used by the applicant in the thermal analyses of ATR FFSC package acceptable.

3.3 Description of ATR FFSC Thermal Model

To perform the thermal evaluation of the ATR FFSC package the applicant developed a thermal model with Thermal Desktop® and SINDA/FLUINT computer programs. The SINDA/FLUINT computer program is a general purpose code that handles problems defined in finite difference (i.e., lumped parameter) and/or finite element terms and can be used to compute the steady-state and transient behavior of the modeled system. Although the code can be used to solve any physical problem governed by diffusion-type equations, specialized functions used to address the physics of heat transfer and fluid flow make the code primarily a thermal code. Thermal Desktop® and SINDA/FLUINT codes provide the capability to simulate steady-state and transient thermal phenomena using temperature dependent material properties and heat transfer via conduction, convection, and radiation.

The applicant developed a three-dimensional (3-D), one-quarter symmetry thermal model of the ATR FFSC to perform the thermal evaluation during NCT. The model simulates one-quarter of the package, extending from the closure to the axial centerline of the package. Symmetry conditions are assumed about the package's vertical axis and at the axial centerline. This modeling choice captures the full height of the package components and allows the incorporation of the varying insolation loads that will occur at the top and sides of the package. Since the ATR FFSC package dissipates essentially no decay heat, the peak temperatures internal to the package are driven by the external heating occurring during NCT and HAC conditions. The potential for developing convective flows within the air filled cavity between the outer shell and the insulation jacket is addressed by increasing thermal conductivity associated with the air overpack nodes in the lower quadrant of the package by a factor of 2 from that for conduction as a means of simulating the type of enhanced heat transfer that convection would cause. The affected nodes are limited to those in the lower quadrant of the package since, in the assumed horizontal orientation of the package under both NCT and HAC conditions, the buoyancy forces associated with convection will tend to drive the flow in this portion of the package in a circular motion, but would only produce a stratified temperature layer in the upper quadrant. Boundary nodes are used to represent the ambient for convection radiation heat transfer purposes. Heating of the exterior surfaces due to solar insolation is assumed using a diurnal cycle. A sine wave model is used to simulate the variation in the applied insolation on the surfaces of the package over a 24-hour period, except that when the sine function is negative, the insolation level is set to zero. The applicant developed a detailed model of the ATR fuel element to simulate the heat transfer within the fuel element and between the fuel element and the FHE. The thermal model includes a separate representation of each composite fuel plate, the side plates (including the cutouts), and the upper end box casting. Heat transfer between the individual fuel plates is simulated via conduction and radiation across the air space separating the plates. The HAC thermal model is a modified version of the quarter symmetry NCT model with the principal model modifications consisting of simulating the expected package damage resulting from the drop events that are assumed to precede the HAC fire and changing the package surface emissivity to reflect the assumed presence of soot and/or surface oxidization.

3.4 Thermal Evaluation under Normal Conditions of Transport

The applicant's predicted maximum temperatures are provided in FSAR Table 3.3-1 for the case of diurnal cycle for insolation loading and an ambient air temperature of 100°F. The values provided in this table show that all components are within in their respective temperature limits during NCT. The applicant assumed that all package components reach a temperature of 100°F at steady-state conditions without insolation and ambient air temperature of 100°F. Since the package has essentially zero decay heat, the resulting 100°F package external surface temperature is below the maximum temperature of 122°F permitted by 10 CFR 71.43(g) for accessible surface temperature in a nonexclusive use shipment. All package components achieve the -40°F temperature under steady-state conditions. Per discussion provided in FSAR Section 3.2.2, Technical Specifications of Components, the -40°F temperature is within the allowable operating temperature range for all ATR FFSC package components. The maximum operating temperature of 190°F (based on the outer shell temperature, per FSAR Table 3.3-1, conservatively rounded up) is used to obtain the maximum pressure rise within the sealed volume. Using this temperature the applicant predicted than the maximum pressure will be less than 4 psi.

Based on the described thermal model and the analysis results, the staff finds the ATR FFSC transportation package thermal evaluation during NCT acceptable.

3.5 Thermal Evaluation under Hypothetical Accident Conditions

The applicant performed the analysis during HAC (fire) using a modified version of the quarter symmetry NCT model with the principal model modifications consisting of simulating the expected package damage resulting from the drop events that are assumed to precede the fire and changing the package surface emissivities to reflect the assumed presence of soot and/or surface oxidization. For performing the fire analysis the applicant modified the NCT model as follows. The worst-case damage arising from the postulated HAC free and puncture drops was simulated. An initial, uniform temperature distribution of 100°F based on a zero decay heat package at steady-state conditions with a 100°F ambient with no insolation was assumed. The emissivity of the external surfaces was increased to 0.8 to account for possible soot accumulation on the surfaces. The emissivity of the interior surfaces of the outer shell was increased to 0.45 to account for possible oxidization of the surfaces during the HAC event. The fire test conditions evaluated by the HAC analysis address 10 CFR 71.73(c) requirements.

Applicant's predicted temperatures are provided in FSAR Table 3.4-1. This table presents component temperatures seen prior to the fire, at the end of the 30-minute fire event, and the peak temperature achieved during the entire simulated fire event. As can be seen from this table all temperatures are within their allowable limit. The applicant calculated the maximum operating pressure by assuming a temperature of 70°F at the time of assembly and a maximum fire temperature of 1475°F based on the outer shell temperature provided in FSAR Table 3.4-1. The applicant's predicted a maximum pressure rise within the sealed volume due to ideal gas expansion to be less than 39 psig.

Based on the described model and the thermal evaluation results, the staff finds the ATR FFSC transportation package thermal evaluation during HAC fire conditions acceptable.

3.6 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the ATR FFSC transportation package thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The objective of the criticality evaluation is to verify that the amended ATR FFSC package design satisfies the criticality safety requirements of 10 CFR Part 71, including performance under the normal conditions of transport (NCT) specified in 10 CFR 71.71 and the hypothetical accident conditions (HAC) specified in 10 CFR 71.73. The staff's review considered the criticality safety requirements of the radioactive material transportation regulations in 10 CFR Part 71, and the review guidance presented in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

6.1 Description of the Criticality Design

The ATR fuel element was previously approved as authorized contents in previous amendments for the ATR FFSC package. An ATR fuel element consists of uranium aluminide with 94 wt% ^{235}U enrichment sandwiched in aluminum cladding to form fuel plates. The applicant has developed a demonstration element using low-enriched uranium (LEU) for several of the fuel plates. To achieve the necessary fissile mass in the LEU fuel plates, the fuel matrix for these plates is being changed from UAl_x to U-Mo, which allows a much higher uranium density. This amendment addresses the change from UAl_x to U-Mo. The applicant did not request any changes to the approved packaging design.

The applicant provided tables summarizing the results of the criticality evaluation for a single package and arrays of packages under NCT and HAC. SAR Table 6.12-1 contains a summary of the final analysis results of the criticality safety analyses. The package or package array is considered to be subcritical if k_{safe} (k_s) for each of the analysis cases is less than the Upper Subcritical Limit (USL). The computed k_{safe} is equated as $k_s = k_{eff} + 2\sigma < USL$. The applicant's USL, including the administrative margin and bias, is 0.9209. Staff reviewed this table and found that the applicant's calculated maximum k-effective values, including two standard deviations, are significantly less than the USL.

The applicant used a 9x9x1 array and a 5x5x1 array of packages for the NCT and HAC array calculations, respectively. For the purposes of determining a Criticality Safety Index (CSI), the HAC array is the most limiting and results in a higher CSI. The applicant calculated a CSI of 4.0. Based upon the applicant's analysis and staff's confirmatory calculations, the staff finds that the applicant correctly derived the package CSI and that a CSI of 4.0 is appropriate for the package.

6.2 Fissile Material Contents

The applicant described the proposed contents in Sections 1.2.2, 6.2 and 6.12.2 of the application. The proposed contents consist of one ATR U-Mo demonstration element. A schematic of the demonstration element is provided in Figure 6.12-1. The demonstration element contains a mixture of UAl_x (HEU) and U-Mo (LEU) fuel plates, with a maximum U-235 mass of 1,240 g.

The external geometry of the demonstration element is essentially identical to the external geometry of a standard ATR element shown on Figure 6.2-1. The width (or arc length) of the U-Mo fuel meat is also the same as a standard UAl_x element. However, the U-Mo fuel meat thickness is 0.013-in, and a 0.001-in zirconium interlayer is present between the fuel meat and the cladding. The cladding material is aluminum 6061 for all fuel plates.

Staff reviewed the dimensions provided by the applicant and finds them to be consistent with or bounded by those used in the applicant's analysis. Staff reviewed the fuel mixture mass and atom densities and finds them to be consistent. Staff reviewed the fuel element and plate descriptions and the U-235 enrichment and mass limits. Based on the applicant's analysis and staff's confirmatory calculations, the staff finds that the applicant has defined adequately the type, maximum quantity, and chemical and physical form of the fissile material in compliance with the requirements of 10 CFR 71.31(a)(1), 10 CFR 71.33(b)(1), 10 CFR 71.33(b)(2), and 10 CFR 71.33(b)(3).

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The staff reviewed the applicant's model description in Section 6.12.3.1 of the SAR. This model takes into account the effects of the NCT and HAC tests specified in 10 CFR 71.71 and 71.73 and is similar to the model used for ATR fuel in previous amendments. The applicant used the contents and packaging tolerances that maximize reactivity. The presence of moderator was assumed only in the package cavity since there is no damage to the packaging such that water can access the gap between the insulation and the outer steel wall of the package due to the NCT conditions tests. Additionally, analyses for the HAC array indicate that inclusion of moderator between the cavity and outer package steel tubes reduces system reactivity. The applicant's NCT array models also include rotation of and shifting of the contents to the center of the array. The staff reviewed the applicant's analysis models, and, based upon the information provided by the applicant as well as its own confirmatory calculations, the staff finds the model configurations and analysis to be acceptable.

6.3.2 Material Properties

The staff verified that the appropriate atom densities are provided for all materials used in the models of the packaging and contents by reviewing applicable tables in Chapter 6 of the SAR. The compositions and densities for the materials used in the computer models were reviewed by the staff and determined to be acceptable.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant performed the criticality evaluations for the ATR FFSC using the three-dimensional Monte Carlo code MCNP5 (version v1.30) with continuous energy cross-sections. The applicant used the most up-to-date cross section libraries for the model materials that are available in MCNP. These libraries were derived from ENDF/B-V, VI, and VII cross section data. In addition, the applicant used the appropriate MCNP5 option to properly account for the hydrogen bound to water.

The MCNP5 code is an industry standard for performing criticality analyses and is widely used in industry application for criticality calculations. As a result, the MCNP code and its associated cross-section sets have been extensively benchmarked against critical experiments. Thus, the staff agrees that the codes and cross-section sets used by the applicant are appropriate for this particular package design and contents.

The application contains sample input and output files, which staff reviewed to confirm that the model inputs and outputs were consistent with the descriptions in the application. The applicant included a sufficient number of particle histories in its calculations to achieve a statistical standard deviation of less than 0.001 in the calculated values of k_{eff}. The staff considers this to be sufficient for this application and finds the applicant's use of the code acceptable.

6.3.4 Demonstration of Maximum Reactivity

The applicant performed several calculations for the fuel element contents for a single package and for arrays of packages. Staff reviewed the applicant's analyses and finds reasonable assurance that the most reactive configuration of the package is considered. Optimum moderation conditions were identified, and appropriate consideration was given regarding preferential flooding. Further descriptions of these analyses and their results are provided in Sections 6.4 through 6.6 of this Safety Evaluation Report.

6.3.5 Confirmatory Analyses

Staff performed confirmatory analyses on the most reactive configurations described by the applicant. The SCALE 6 computer software package was used as an alternate independent code to the MCNP code used by the applicant for the analyses of the ATR FFSC. Staff calculations were performed with the CSAS26 criticality sequence of the SCALE 6 suite of codes. SCALE 6 was developed by Oak Ridge National Laboratory for use in criticality and shielding analyses. The CSAS26 sequence uses KENO-VI geometry. Staff used the 238-group cross section libraries derived from ENDF/B-V data. Significant parameters were varied to ensure maximum reactivity peaks were adequately captured and in all instances staff calculations were bounded by or in close agreement with the applicant's results.

6.4 Single Package Evaluation

The staff reviewed the applicant's evaluation of a single package. The single package was modeled with full water reflection on all sides. The fissile material was modeled in the most reactive credible configuration consistent with the condition of the package and contents. The staff verified that the applicant's evaluation demonstrates that a single package is subcritical under both normal conditions of transport and hypothetical accident conditions. The applicant's results show that a single package is significantly subcritical.

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

The applicant performed calculations using a 9x9x1 array surrounded by a full water reflector. The most reactive single package case is used as the basis for the NCT array model. The most reactive array configuration for the fuel element contents was full density water between the fuel element plates, 0.3 g/cc water density in the package cavity with neoprene (without chlorine) and insulation present, and void between the steel tubes forming the package cavity and outer wall. Due to the similarity of the NCT and HAC models, the NCT array reactivity exceeds the HAC array reactivity. The maximum k-effective is significantly less than the applicant's USL.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The staff reviewed the applicant's analysis of an array of damaged packages. The package array (a 5x5x1 array) was surrounded by a full water reflector. The contents were oriented and shifted toward the center of the array. The applicant included the outer tube of the packaging in its analysis. Tolerances were used in the analysis to maximize reactivity. Based upon the structural evaluation (see Section 2 of this safety evaluation report), the HAC tests resulted in only localized deformation of the package; i.e., there is no overall deformation of the package that would increase system reactivity. Therefore, the staff finds the analytical model to be acceptable. The analysis also includes the impact of the insulation. For the fuel element contents, the most reactive conditions were full density water between fuel plates, 0.7 g/cc water density in the package cavity, void between the package cavity and outer steel wall, and neoprene (without chlorine) and insulation present.

6.7 Fissile Material Packages for Air Transport

Air transport for the ATR FFSC was not sought by the applicant and is not authorized.

6.8 Benchmark Evaluations

The applicant examined critical experiment cases from the International Handbook of Evaluated Criticality Safety Benchmark Experiments based upon their similarity to the ATR FFSC and contents. The important selection parameters were HEU and LEU uranium plate-type fuels with a thermal spectrum. Thirty-five benchmarks are available for HEU plate fuel, while only one is available for LEU plate fuel. Therefore, the plate-type benchmarks were supplemented with 54 LEU rod benchmarks, for a total of 90 benchmarks.

Ideally, benchmarks would be limited to those with a fuel matrix of HEU UAl_x and LEU U-Mo, aluminum cladding, and no absorbers, consistent with the ATR demonstration element criticality models. However, no experiment set is available that meets all of these criteria since U-Mo fuel is in research and development stage, and benchmarks for U-Mo fuel designs are not available. Therefore, the selected experiments are subdivided into two general subsets, plate-type benchmarks and LEU rod benchmarks. Trending is performed for both subsets of benchmarks and the entire benchmark set. The USL selected is the minimum of all sets.

The primary difference between the U-Mo demonstration element and a standard ATR element is the presence of molybdenum rather than aluminum in the fuel matrix. It is demonstrated in SAR Section 6.12.5, “Evaluation of Package Arrays under Normal Conditions of Transport,” that deletion of molybdenum from the MCNP model has very little effect on the reactivity. Therefore, because molybdenum has little effect on the reactivity and margins to the USL are very large, the lack of U-Mo benchmarks has little effect on the USL and is acceptable.

6.8.1 Bias Determination

The USL and bias were determined using USLSTATS for several trending parameters. The USLSTATS tool in SCALE provides trending analysis for bias assessment. Separate USL ranges were calculated for six parameters: (1) energy of the average neutron lethargy causing fission (EALF), (2) U-235 number density, (3) channel spacing, (4) hydrogen to U-235 atom ratio (H/U-235), (5) plate pitch and (6) U-235 enrichment. All parameters were evaluated for trends and to determine the minimum USL for each parameter using the full benchmark set of 90 experiments and the subsets of plate-type benchmarks and LEU rod benchmarks. The applicant then selected the minimum value among these six parameters, which resulted in an overall USL of 0.9224. However, the benchmarking analysis documented in SAR Section 6.8, Benchmark Evaluations, for the standard HEU element resulted in a USL of 0.9209. Both for consistency and added conservatism, a USL of 0.9209 was selected for this analysis. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations. The staff reviewed the benchmark comparisons in the SAR and agrees that the Monte Carlo computer program MCNP5 v1.30 used for the analysis was adequately benchmarked to representative critical experiments relevant to the package design and contents specified.

6.9 Burnup Credit

The applicant did not request credit for burnup. All analyses are modeled using fresh, un-irradiated fuel.

6.10 Evaluation Findings

Based on the review of the statements and representations in the application, supplemental information supplied by the applicant, and staff confirmatory analyses, the staff has reasonable assurance that the nuclear criticality safety design has been adequately described and evaluated by the applicant and that the package meets the criticality safety requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES EVALUATION

The applicant did not request any changes to the operating procedures in its application. The ATR U-Mo demonstration elements are almost identical in size and shape to the ATR Mark VII fuel element already authorized in the certificate of compliance. Accordingly, the staff finds that the existing operating procedures for the ATR Mark VII fuel remain valid and acceptable for use with the ATR U-Mo demonstration elements.

CONDITIONS

The following changes have been made to the certificate of compliance:

Condition No. 5.(b)(1) has been revised to include U-Mo demonstration elements as authorized contents of the package.

Condition No. 5.(b)(2) has been revised to specify the amount of U-Mo fuel elements that can be shipped in a single package.

Condition No. 5.(c) has been revised to include the criticality safety index for U-Mo fuel elements.

The previous Conditions No. 11 was deleted. A new Condition No. 11 has been included to allow use of Revision No. 6 of the certificate until the expiration date of the certificate.

CONCLUSION

Based on the statements contained in the application, and the conditions listed above, the staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9330, Revision No. 7,
on December 12, 2013.