February 28, 2013

Mr. Anthony Patko
Director, Licensing
Engineering
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, GA 30092

SUBJECT: REVISION 58 OF CERTIFICATE OF COMPLIANCE NO. 9225 FOR THE MODEL NO. NAC-LWT PACKAGE (TAC NO. L24697)

Dear Mr. Patko:

As requested by your application dated October 26, 2012, as supplemented on December 5, 2012, January 14, and February 14 2013, enclosed is Certificate of Compliance No. 9225, Revision No. 58, for the Model No. NAC-LWT 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.

Sincerely,

/RA/

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

Docket No. 71-9225
TAC No. L24697

Enclosures: 1. Certificate of Compliance No. 9225, Rev. No. 58
2. Safety Evaluation Report
3. Registered Users

Upon removal of Enclosure 3, this document is uncontrolled

cc w/encls. 1& 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy

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SAFETY EVALUATION REPORT

Docket No. 71-9225
Model No. NAC-LWT
Certificate of Compliance No. 9225

SUMMARY

By application dated October 26, 2012, as supplemented on December 5, 2012, January 14, and February 14, 2013, NAC International (NAC or the applicant) requested a revision to Certificate of Compliance (CoC) No. 9225 for the Model No. NAC-LWT (NAC-LWT) transportation package. NAC requested the addition of National Research Universal Reactor (NRU) and National Experimental Reactor (NRX) spent fuel as authorized contents to be transported in a basket designed specifically for these fuel types. The staff performed its review of the application as supplemented using the guidance in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

This shipment is necessary to support shipments to the Savannah River Site in the U.S. The package loading operations and the shipment schedule will be established by the U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) Foreign Research Reactor Return (FRR) program.

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 NAC-LWT package is a Type B(U)F-96 radioactive material transportation package. It is authorized to transport shipment of light water reactor and research reactor spent fuel. The NAC-LWT package is shipped by truck, within an international shipping organization (ISO) container, or by railcar, as a Type B(U)F-96 package, as defined in 10 CFR 71.4.

1.2 Packaging Drawings

The applicant submitted five new engineering drawings that show the design of the basket and caddy to transport the NRU/NRX fuel, and its configuration within the NAC-LWT package.

The new drawings include:

LWT 315-40-170, Rev. 1	LWT Transport Cask Assy,. AECL NRU/NRX Components
LWT 315-40-172, Rev. 0 (sheets 1 - 2)	Lid Assembly, NRU/NRX
LWT 315-40-173, Rev. 0 (sheets 1 - 2)	Basket Weldment, NRU/NRX

LWT 315-40-174, Rev. 0	Basket Spacer, NRU/NRX
LWT 315-40-175, Rev. 1	Caddy Assembly, NRU/NRX

1.3 Contents

NAC requested a revision to authorize shipment of NRU/NRX spent fuel in the NAC-LWT package specified as follows:

- Up to 18 undamaged NRU or NRX fuel assemblies (or the equivalent number of loose rods) may be loaded per NRU/NRX fuel basket in accordance with NAC Drawing Nos. 315-40-172, 315-40-173, 315-40-174 and 315-40-175.
- NRX fuel shall be placed into the fuel caddy.
- Placement of NRU fuel into the fuel caddy is optional.
- NRU and NRX fuel may not be comingled within a single package.

2.0 STRUCTURAL EVALUATION

2.1 Description of the Structural Design

To support the transportation of NRU/NRX fuel, in the NAC-LWT package the applicant requested addition of a new basket design – the NRU/NRX fuel basket. The outer structural components of the package remain unchanged. A detailed description of the basket is provided in Section 2.6.12.13 of the safety analysis report (SAR).

2.2 Material Properties

2.2.1 Materials and Material Specifications

The NAC-LWT package description is discussed in Section 1.2 of the application. The principal structural members of the NAC-LWT package per this amendment request include the NRU/NRX fuel assemblies, basket weldment, and the fuel caddy assembly.

NRU/NRX FUEL ASSEMBLY:

The NRU and NRX fuel assemblies contain fuel types that are typically solid cylindrical rods clad in aluminum (Al). Uranium (U) metal fuels may be susceptible to corrosion and can form pyrophoric uranium hydride when it corrodes in an oxygen scarce, but hydrogen generous atmosphere. NRU and NRX are uranium-aluminum alloy fuels of high-enriched uranium (HEU) and low-enriched uranium (LEU), which are less vulnerable to corrosion in comparison. The uranium silicide (Si) fuels (U-Al-Si) are the current NRU driver fuels, used in the production of medical radioisotopes and neutrons for condensed matter and material research.

BASKET WELDMENT:

The NRU/NRX fuel basket assembly consists of a top basket weldment, lid assembly, and a basket spacer assembly. Each fuel tube is capable of holding a NRU/NRX fuel assembly or the equivalent number of individual fuel rods positioned inside an Al caddy. The basket assembly, lid assembly, and basket spacer assembly are fabricated primarily from ASTM SA 240, Type 304 stainless steel (SS).

CADDY ASSEMBLY:

The caddy consists of a cylinder with an end cap or bottom plate as discussed in Section 2 of the application. The cylinder is fabricated from ASTM B 210 Alloy 6061-T6 Al, covered with a commercial Al mesh, welded and a bottom plate constructed of ASTM B 209, Alloy 6061-T6 Al. Al alloy 6061 is a precipitation-hardened alloy, containing magnesium and silicon as its major alloying elements. NRX assemblies or rods must be placed into the fuel rod caddy assembly for handling and geometry constraint. NRU fuel rods are not required, however may be placed in a caddy assembly.

The staff reviewed the materials selected and determined that they are acceptable and provide reasonable assurance for safety of the package. These types of materials are selected based on properties such as strength, ductility, and resistance to corrosion. Specifications and temperature dependent mechanical properties, including yield strength, tensile strength, allowable strength, modulus of elasticity, and coefficient of thermal expansion conform to ASME Code, Section II, Part D.

2.2.2 Chemical or Galvanic Reactions

The staff concludes that, during normal conditions of transportation, the NAC-LWT package internals will not be subject to exposure to moisture and water intrusion is not likely to occur. The NAC-LWT package internals are dried and backfilled with helium, as a result all conditions necessary for galvanic potential between the different metals used in fabrication do not exist. In addition, as a transportation package, time between contact of internals and contents is limited. Further, visual inspections to be performed of the important-to-safety payload cavity in the NAC-LWT package at various timed intervals provide reasonable assurance against any significant corrosion occurring unnoticed.

2.2.3 Brittle Fracture

The staff finds that by avoiding the use of ferritic steels brittle fracture concerns are precluded. Specifically, most primary structural packaging components are fabricated of Type 304 SS. In austenitic SS metal the force required to move dislocations is not strongly temperature dependent and dislocation movement remains high (i.e., will deform more readily under load before breaking) even at low temperatures and the material remains relatively ductile.

2.3 Structural Review

Structural acceptance criterion for the new basket design, generally, follows the criterion identified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, paragraph NG-3200 for normal conditions of transport and hypothetical accident conditions (Appendix F). Staff determined this approach acceptable, as it is consistent with Table 1.1 of NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," for criticality components. However, staff has noted that the analyses pertaining to localized buckling does not follow ASME code allowables, but contains adequate margin of safety and follows a well-established academic methodology in "Practical Stress Analysis in Engineering Design," by Alexander Blake for those analyses of the fuel basket.

The NRU/NRX fuel basket was evaluated for a lateral (side) and longitudinal (end) drop load of 25 g, which is representative of a 1-foot normal condition free drop, and a lateral (side) and longitudinal (end) drop load of 61 g, which is representative of a 30-foot hypothetical accident free drop. Staff finds this approach acceptable given the other packaging components remain as previously approved.

The staff did question the ability of the NRU/NRX fuel (which was not analyzed in this application) to meet regulation 10 CFR 71.55(d)(2), in that:

> "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests in 10 CFR 71.71 ("Normal conditions of transport") – the geometric form of the package contents would not be substantially altered."

Per the information provided by NAC in response to NRC's request for additional information 2-1, dated January 14, 2013 (see Agencywide Documents Access and Management System (ADAMS) Accession No. ML13022A387), the NRU/NRX fuel basket has a "common basket lid" which prevents any loaded fuel rods from leaving the tube/location as loaded. Also, the post-normal conditions of transportation fuel configuration criticality analyses performed is bounding, as it considers broken fuel rods. Based on response to this question and Section 7.2.4 of the application, for unloading operations, NRC staff is satisfied that the intent of 10 CFR 71.55(d)(2) is met, in that there is no substantially altered configuration that challenges the criticality analyses, or that poses an undue risk to health and/or safety during operations.

2.4 Evaluation Findings

The staff finds that the NAC-LWT package meets the regulatory requirements for mitigating galvanic or chemical reactions, is unaffected by cold temperatures and is constructed with materials and processes in accordance with acceptable industry codes and standards.

Based on review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated and that the new NRU/NRX fuel basket has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

3.1 Description of Thermal Design

The maximum heat load for 18 NRU or NRX fuel rod assemblies is 0.64 kW per package, which is bounded by the heat load, 2.3 kW per package (143 W per PWR MOX rod), for 16 pressurized water reactor, mixed oxide fuel rods. Up to 18 NRU or NRX fuel assemblies, or the equivalent number of fuel rods of either type, will be loaded into each basket in the NAC-LWT package. During shipment, the package will be enclosed in an ISO container.

3.2 Thermal Evaluation for Normal Conditions of Transport

3.2.1 Thermal Evaluation of NRU or NRX Fuels

Thermal analysis of the NAC-LWT package containing NRU or NRX fuel assemblies is performed using the general-purpose ANSYS computer code and a three-dimensional finite element model representing a periodic section of the contents of the NAC-LWT package. The model represents the contents inside the package's inner shell. The NAC-LWT package is supported in an ISO container with solar insolation applied on the surface of the ISO container and is considered to be insulated from the environment (only for the normal conditions of transport, steady state condition heat test).

The gas inside the ISO container is air. The cavity of the NAC-LWT package is backfilled with helium as required by the operating procedures. The ambient temperature is 100°F with solar insolation. Half of the basket content for a periodic portion is modeled due to the symmetry of the geometry and the heat load. The three-dimensional model of NRU or NRX fuels loaded in the NAC-LWT package, as shown in Figure 3.4-21 in the application, consists of: 1) a center tube assembly that is modeled as a hollow cylinder; 2) fuels inside the tube assemblies; 3) helium inside the package cavity; 4) support disk with a half thickness; 5) a helium gap with a uniform thickness of 0.0525 inches between the basket and the package inner shell; and 6) stainless steel tubes. The fuel inside the tubes is conservatively modeled as uranium oxide (UO_2). Using UO_2 thermal conductivities is conservative since the fuel meat of the NRU and NRX fuel assemblies is composed of uranium and aluminum alloy, and is significantly more conductive than UO_2. Thermal conductivities of UO_2 are listed below in Table I. There is no contact between the tubes, as well as the support disk and the package inner shell due to the modeling of the helium gap. The radiation between the tube assemblies is conservatively neglected.

Table I. Thermal Conductivities for UO_2

Temperature, °F	**k, BTU/hr-in-°F for UO_2**
100	0.380
257	0.347
482	0.277

A constant temperature of 202°F is applied to the outer surface of the model, which corresponds to the maximum inner shell temperature. This maximum package inner shell temperature is obtained using the two-dimensional ANSYS model for normal condition (Condition 1) from Section 3.4.1.3 in the application after deleting all elements inside the package inner shell. A heat flux, computed based on a heat load of 0.64 kW/package, is applied to the inner surface of the package's inner shell of this two-dimensional model to obtain the maximum inner shell temperature of 202°F. For the three-dimensional model, the top and the bottom surfaces of the three-dimensional model are adiabatic due to the symmetry. The total heat load of 0.64 kW for the whole package is used and is distributed over the active fuel length of 108 inches. The heat generation rate for the fuel in the three-dimensional model is computed below.

$$H_{gen} = \frac{640\,W * 3.413\frac{\frac{Btu}{hr}}{W}}{18 * \pi * (1.185\,in.)^2 * 108\,in.} = 0.2547\frac{Btu}{hr * in.^3}$$

Where 640 Watts is the heat; 18 is the number of the fuel assemblies; 1.185 inches is the inner radius of the tube; and 108 inches is the active fuel length.

The maximum temperature in the model is computed to be 245°F, which is much lower than the allowable temperature of 400°F for aluminum fuel cladding as defined in Section 3.4.1.3.3 in the application. The maximum temperature for stainless steel tubes and support disks is bounded by the maximum fuel temperature of 245°F and is lower than the allowable temperature of 800°F for stainless steel (Section 3.4.10 in the application). The heat load of the MTR fuel (1.26 kW, Section 3.1) bound the heat load of the NRU/NRX material (0.64 kW). Therefore, the maximum temperatures of the package components for the MTR contents (Condition 1, Table 3.4-6 in the application) bound the maximum temperatures for the package components for the NRU/NRX material contents. The staff finds the values that were calculated acceptable.

3.2.2 Maximum Temperatures

Within this amendment, the maximum temperature inside the package for NRU/NRX fuels is 245°F. The maximum temperatures of the package components for the MTR contents (Condition 1, Table 3.4-6) bound the maximum temperatures for the package components for the NRU/NRX material contents. The staff finds the values that were calculated acceptable.

3.3 Maximum Internal Pressure for Aluminum-Based Fuels

Section 3.4.4.8 of the SAR determines the bounding NAC-LWT package internal pressure for the package during normal conditions of transport for aluminum-based research reactor fuel contents (i.e., ANSTO, DIDO, MTR, and NRX/NRU fuels). NRU/NRX payloads are not evaluated for system pressure as inputs into the analysis outlined below all indicate a conservative system pressure being obtained from the MTR payload:

- Total heat load and temperature are below that of the MTR payload. Furthermore, total fuel and fissile material mass (U-Al, or U-Al-Si) in the 18 NRU or NRX assemblies is less than MTR fuel mass (2 elements maximum).

- MTR elements were evaluated at higher burnup levels than NRU/NRX and therefore, the NRX/NRU fuel will contain less fission gas.

- The void space in the NRU/NRX package cavity is higher than that of the fully loaded MTR system as the NRU/NRX bottom basket spacer occupies very little volume versus a loaded MTR basket and the NRU/NRX fuel assembly and basket cross section contains significant void areas.

The staff finds the values that were calculated acceptable.

3.4 Thermal Evaluation for Hypothetical Accident Conditions

The heat load used for the NRU/NRX material (0.64 kW) is lower than the heat load of 1.26 kW for the MTR fuel (mentioned in Section 3.1 of the SAR). The same NAC-LWT package is used for loading the NRU/NRX material and for loading the MTR fuels. Therefore, it is conservative to use the temperature increase of the package inner shell of the MTR configuration for the fire accident condition of the NRU/NRX fuels. The maximum inner shell temperature is 337°F for fire accident (Table 3.5-2, Condition 2 located in the SAR) and the minimum inner shell temperature for the normal condition is 180°F (Table 2.4-6, Condition 2 located in the SAR). For the MTR configuration, the bounding temperature increase of inner shell due to fire accident is 157°F (337°F - 180°F) during the fire and cool down stages, which corresponds to the Condition 2 (package transported via truck trailer and cavity gas is air).

To get a bounding fuel temperature of the NRU/NRX material for the fire test in the hypothetical accident conditions, the applicant added this temperature increase of 157°F to the maximum fuel temperature of 245°F for the normal condition (Section 3.4.1.18 of the SAR) since it neglects the thermal inertia of the entire contents inside the NRU/NRX package. Therefore, the maximum fuel temperature for the NRU/NRX material is calculated to be 402°F (245°F + 157°F) for the fire accident condition, which is lower than the allowable temperature of 500°F defined in Section 3.5.3.2 in the application. This value can be viewed as conservative due to the modeling of the tubes containing UO_2. In addition, using UO_2 thermal conductivities is conservative since the fuel meat of the NRU/NRX is composed of uranium and aluminum alloy, which are significantly more conductive than UO_2. The maximum temperature of the tube and support disk is bounded by the fuel temperature of 402°F, which is lower than the normal allowable temperature of 800°F for stainless steel, therefore, the tube and support disk are determined to be acceptable for the fire accident. Therefore, the staff finds the thermal evaluation for hypothetical accident conditions is acceptable.

Staff performed analysis by running the input files in ANSYS as provided by the applicant and performing calculations as listed in the SAR. The values viewed in ANSYS are similar to those listed in the application, thus the values did not exceed any of the temperature barriers for normal conditions of transport and hypothetical accident conditions.

3.5 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the 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.

4.0 CONTAINMENT EVALUATION

The staff reviewed the application to revise the NAC-LWT package to verify that the package containment design has been described and evaluated under normal conditions of transport and hypothetical accident conditions as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617.

This section presents the findings of the containment review for a request for authorization to approve shipment of the NRU/NRX fuel for the NAC-LWT package.

4.1 Containment Evaluation

The NAC-LWT package remains leaktight as previously evaluated. Since the package remains leaktight, no containment source terms were provided or required. There are no changes in the containment boundary.

4.2 Conclusion

The staff evaluated the containment safety analysis for the packages that are loaded with the NRU/NRX fuel. Based upon the information provided by the applicant, the staff has reasonable assurance that the package design continues to meet the containment safety requirements in 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The objective of the shielding review is to verify that the incorporation of the NRU/NRX fuel as authorized content to the NAC-LWT package satisfies the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.1 Description of the Shielding Design

The applicant chose lead and steel as effective gamma radiation shields, and a water tank on the outside of the package is provided to efficiently moderate and absorb the neutron radiation. The shield materials were selected and arranged to minimize package weight while maintaining overall shield effectiveness. Also, these proposed fuel contents will be transported in an ISO container.

The applicant requested approval for transporting up to 18 NRU or up to 18 NRX assemblies. NRU fuel assemblies contain 12 fuel pins arranged in an annular configuration (nine outer pins and three inner pins). The NRU fuel consists of either U-Al containing HEU or U_3-Si-Al containing LEU with aluminum clad. The HEU NRU fuel was analyzed for a loading of 43.7 g U-235 per pin at a minimum enrichment of 91.0 weight percent U-235, with a maximum burnup of 364 MWd per assembly and a 19-year cool time. The LEU NRU fuel was analyzed for a loading of 43.7 g U-235 per pin at a minimum enrichment of 19.0 weight percent U-235, with a maximum burnup of 363 MWd per assembly and minimum cooling time of 3 years.

NRX fuel assemblies contain seven fuel pins arranged in an annular configuration (six outer pins, 1 central pin). The NRX fuel consists of U-Al alloy with aluminum clad and was analyzed for a loading of 79.2 g U-235 per pin at a minimum enrichment of 91.0 weight percent U-235, with a maximum burnup of 375 MWd per fuel assembly and a minimum cooling time of 18 years. The design basis fuel parameters were provided in Table 5.1.1-1 of the application.

Fuel in undamaged and damaged configuration was considered for the shielding evaluation. The undamaged fuel configuration is assumed to be structurally sound rods or assemblies.

5.2 Source Specifications

The applicant calculated the source terms for the NRU and NRX assemblies using TRITON code in SCALE 6.1. The TRITON models use the 238-group ENDF/B-VII cross section library. According to the applicant, single unit cells were used for the TRITON source term calculation. The single unit cell (assembly reflected) was compared against a model using supercells (assembly plus surrounding in-core material) to define surrounding fuel assemblies. The single unit cell was determined to be more conservative for neutron source terms and is not significantly different for gamma source terms, which dominate dose rate contributions for the material. NRU source terms are calculated using detailed operating histories for HEU and LEU fuel provided by Atomic Energy of Canada Limited (AECL). NRX source terms are calculated using the maximum reactor power and U-235 core loading.

5.3 Model Specification

The shielding analysis of the NAC-LWT package for the NRU and NRX contents was performed using MCNP5 Version 1.60 (v1.60) computer code. The MCNP shielding model was utilized with the source terms to estimate the dose rate profiles at various distances from the side, top, and bottom of the package for both normal conditions of transport and hypothetical accident conditions. Dose rates were computed for the three fuel sources (NRU HEU, NRU LEU, and NRX HEU) for both undamaged and collapsed configurations. Both undamaged and damaged fuel configurations were analyzed under normal conditions of transport and hypothetical accident conditions. The applicant stated that the undamaged fuel configuration includes NRU and NRX pins modeled as cropped for loading in the NAC-LWT package. The damaged fuel configuration collapses the fuel in the basket tubes fully. Collapsed fuel was modeled at the nominal fuel density. The fuel meat alloy will not compact as a result of any transport condition. Also, collapsed models do not include clad or end plug material. The fuel and basket was shifted towards the top of the NAC-LWT package cavity. The radial lead gamma shield extends from the bottom of the package cavity to approximately 3 inches (7.62 cm) below the top of the cavity. In that way, the model locates the fissile material closest to the point of minimum gamma shielding and is conservative.

5.4 Evaluation

The applicant utilized a large number of conservative assumptions throughout their shielding calculations to provide assurance that the actual dose rates will always be below the calculated dose rates, as well as below regulatory limits. Minimum dimensions were used where applicable. To provide conservative source terms, NRU source terms were calculated using detailed operating histories for HEU and LEU fuel provided by AECL while NRX source terms were calculated using the maximum reactor power and U-235 core loading. Also, all sources were calculated for a U-235 depletion of greater than 80%. NRU LEU is composed of U_3-Si-Al. All Si is modeled as aluminum to produce bounding neutron source terms due to neutron production from the α,n reaction.

The applicant used MCNP5 v1.60 to calculate dose rates at the various desired locations. Since MCNP5 calculates neutron or photon fluxes, these values are converted to dose rates and use the response functions from ANSI/ANS 6.1.1-1977, “American National Standard for Neutron and Gamma-Ray Flux to Dose Factors.” External dose rates were calculated on the surface of the package and 2 meters from the edge of the transport vehicle during normal conditions of transport. For hypothetical accident conditions, the maximum dose rates were calculated 1 meter from the package surface.

The NRU/NRX fuel, which is composed of a metal alloy, is not expected to fail during transport, and will not produce rubble. However, the applicant looked at a damaged fuel configuration that fully collapses the fuel in the basket tubes. Collapsed fuel was modeled at the nominal fuel density. The fuel meat alloy will not compact as a result of any transport condition. Also, collapsed models do not include fuel clad or end plug material. The applicant’s analyses demonstrated that the expected dose rates would still be below regulatory limits. Tables 5.3.20-23 and 5.3.20.24 in the application (as reproduced below) summarize the maximum dose rates for undamaged fuel and collapsed fuel.

Table II. Summarized Maximum Dose Rates for Undamaged Fuel

Transport Condition	Dose Rate Location	NRU HEU	NRU LEU	NRX HEU	Limit [mrem/hr]
Normal	Side Surface of Package	2.28	41.5	1.98	1000
	2m from Truck - Radial	0.064	4.02	0.065	10
	Dose at Cab of Truck	0.001	0.057	0.001	2
Accident	Side 1m	0.463	25.1	0.481	1000

Table III. Summarized Maximum Dose Rates for Collapsed Fuel

Transport Condition	Dose Rate Location	NRU HEU	NRU LEU	NRX HEU	Limit [mrem/hr]
Normal	Side Surface of package	30.3	313.3	39.9	1000
	2m from Truck - Radial	0.450	6.59	0.595	10
	Dose at Cab of Truck	0.010	0.153	0.011	2
Accident	Side 1m	3.75	41.6	4.68	1000

5.5 Evaluation Findings

The staff reviewed the description of the package design features related to shielding and the source terms for the design basis fuel and found them acceptable. The methods used are consistent with accepted industry practices and standards. Confirmatory analyses were performed by the staff using ORIGEN-ARP, depletion code, as part of the SCALE 6.1 code package. Comparison of the source terms were about 5% in agreement with the applicant. In a conference call with the applicant, staff asked the applicant to clarify whether these contents are shipped in an enclosed vehicle. Table 5.3.20-20 in the application shows a maximum dose rate on the side surface of the transportation package to be 313.3 mrem/hour, which is above the limit of 200 mrem/hr to an open vehicle. The applicant responded that this fuel would be transported in an ISO container, which allows the dose rate to be 1000 mrem/hr on the surface of the package.

The staff reviewed the maximum dose rates for normal conditions of transport and hypothetical accident conditions and determined that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51.

Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance, and that the package design and contents satisfy the shielding and dose rate limits in 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

NAC performed a criticality evaluation of NAC-LWT package containing NRU and NRX fuel assemblies and loose rods. The staff reviewed this evaluation as well as the pertinent information from the NAC-LWT SAR Revision 41 (see ADAMS Accession No. ML101750226). The staff performed its review using the guidance in NUREG-1617, and NUREG/CR-5661, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages.”

6.1 Description of Criticality Design

The staff reviewed the General Information section in Chapter 1 of the NAC-LWT SAR as well as any additional information in the Criticality Section, Chapter 6, of the NAC-LWT SAR. The staff verified that the information is consistent as well as all descriptions, drawings, figures, and tables are sufficiently detailed to support an in-depth staff evaluation.

6.1.1 Packaging Design Features

The applicant provided drawings of the package in Section 1.4 of the NAC-LWT SAR. The staff reviewed these drawings and found that they sufficiently describe the locations, dimensions, and tolerances of the containment system and basket. Therefore the staff found that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5) with respect to the criticality evaluation.

For criticality control, the NAC-LWT design with AECL NRU/NRX fuel relies upon limiting the amount of fissile material as well as a fixed geometry of the fuel assemblies. The neutron multiplication factor (or k_{eff}) will be less than 0.95, including all biases and uncertainties, during all normal conditions of transport and hypothetical accident conditions.

6.1.2 Codes and Standards

The staff found that the applicant appropriately identified the regulations in 10 CFR Part 71 that are applicable to the criticality design of the package. In Section 6.5.4.1 of the NAC-LWT SAR the applicant also identifies standards applicable to the validation and application of the criticality computer codes used. The applicant has described and justified the basis and rationale used to formulate the package quality assurance program in Section 1.3 of the NAC-LWT SAR. The quality assurance program provides control over all activities designated important to safety that are applicable to the design, fabrication, assembly, testing, maintenance, repair, modification and use of the packaging for transportation of radioactive materials. The staff found this acceptable because the program met 10 CFR Part 71 Subpart H, requirements and requirements in 10 CFR 71.31(c), with respect to criticality safety.

6.1.3 Summary Table of Criticality Evaluations

The applicant provided a summary table of the criticality evaluations in Table 6.7.2-6 of the NAC-LWT application. The applicant performed an evaluation for a single package considering the effects of the tests for normal conditions of transport and hypothetical accident conditions and an infinite array of packages after the tests for normal conditions of transport.

The applicant demonstrated that the limiting conditions for the NAC-LWT with NRU/NRX fuel is the single package after the tests for hypothetical accident conditions.

This analysis gives a maximum k_{eff} of 0.92560 and includes two times the standard deviation (2σ). The applicant demonstrated that the maximum k_{eff} for the limiting configuration is less than the upper subcriticality limit (USL) of 0.9270. The staff found that this meets the requirements of 10 CFR 71.55(b), (d) and (e), 71.59(a)(1) and (2).

6.1.4 Criticality Safety Index

Section 6.7.2.5 of the NAC-LWT application states that the criticality safety index (CSI) is 100. Per 10 CFR 71.59(b) the value of “N” is 0.5. The applicant calculated an infinite array for normal conditions of transport (5N) and a single package for hypothetical accident conditions (2N). The staff found that these array sizes are acceptable and meet the requirements of 10 CFR 71.59(a)(1) and 10 CFR 71.59(a)(2). In addition, the staff found that the licensee met 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Spent Nuclear Fuel Contents

The applicant proposed to add new fuel types in this amendment request. These are NRU and NRX fuel assemblies and loose rods. NRU fuel rods are built with HEU or LEU, NRX fuel is built with only HEU. Section 6.7.2.1 of the SAR states that NRU fuel is analyzed at 94% and

21% enrichment and NRX fuel is analyzed at 94% enrichment. The MCNP input file for NRU HEU under hypothetical accident conditions in Figure 6.7.2-19 in the application shows that this was analyzed using 93.6% enrichment. The staff finds this acceptable because it bounds the limits in the proposed CoC, which is 90% for NRU HEU, 19% for NRU LEU, and 91% for NRX HEU. The applicant requested approval to ship up to 18 NRU or NRX assemblies in the NAC-LWT package. The NRU/NRX fuel in the NAC-LWT cannot be mixed. Each shipment will consist of a single fuel type.

As stated in Section 1.3.2.12 of the NAC-LWT application, NRU and NRX HEU fuel rods are made of uranium and aluminum with aluminum cladding. NRU LEU fuel rods are made of uranium, aluminum, and silicone with aluminum cladding. The fuel rods are arranged in a circular geometry for both fuel types. NRU assemblies have 12 rods per assembly; NRX assemblies have 7 rods per assembly.

The applicant used nominal dimensions for modeling the fuel. The staff found this acceptable based on other conservative analytical assumptions such as the fresh fuel assumption.

All NRU/NRX fuel will be undamaged. The fuel will either be loose rods or intact fuel assemblies. All NRX fuel will be loaded inside the caddy (Drawing LWT 315-40-175, Rev. 1). NRU fuel may be loaded in the caddy.

The staff found that this meets 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) because the package and contents are adequately defined.

The applicant does not take credit for burn-up. All assemblies are assumed to be fresh fuel. The staff found this conservative because the fuel will be burned and fresh fuel has a higher reactivity. Therefore, the staff found this assumption acceptable.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The staff verified that NAC's criticality analyses determined and used appropriate fuel and package dimensions. The applicant uses nominal design dimensions for the package components. The staff found this acceptable because the package components do not significantly contribute to k_{eff}. NAC considered manufacturing tolerances of the fuel tubes for the NRU HEU fuel and found no statistically significant effect on k_{eff}. The applicant did not consider manufacturing tolerances on the fuel tubes for the NRX fuel or the caddy. The staff found this acceptable because the NRU HEU is the most reactive configuration.

The staff reviewed the structural and thermal evaluation sections of the SAR and determined the effects of normal conditions of transport and hypothetical accident conditions on the packaging and its contents. Although the package body, basket, and caddy are all intact as a result of normal conditions of transport, the fuel condition is unknown especially given that rods can be shipped as loose rods. It is not expected that the U-Al alloy would break apart, however since the integrity of the fuel is not evaluated, the applicant performed a bounding analysis where the fuel was assumed to break apart and achieve an optimal pitch such that reactivity was maximized. The staff found this acceptable because if the fuel were to break up, it would probably be in a random fashion and achieving an optimal pitch that increases reactivity is unlikely. Therefore, the applicant's evaluation is conservative. The hypothetical accident conditions models do not contain the liquid neutron shield, neutron shield shell, or impact limiters.

The applicant considered flooding of the package and preferential flooding within the package. The applicant varied moderator density in and outside of the package tubes as well as the package exterior. These studies are documented in the application in Figures 6.7.2-10, 6.7.2-11, 6.7.2-16 and 6.7.2-17.

6.3.2 Material Properties

The staff verified that the appropriate mass fractions and densities are provided for all materials used in the models of the packaging and contents. The applicant provided this information in Table 6.7.2-4 of the application. The staff found this acceptable because the values used are typical values for the commonly used materials and are reasonable for use in the criticality analysis. The staff found that the material properties are consistent with the package after the tests for normal conditions of transport and hypothetical accident conditions.

6.3.3 Computer Codes and Cross Section Libraries

Section 6.7.2.2 of the NAC-LWT application states that the applicant performed the criticality evaluations using the MCNP5 v1.60; a three-dimensional Monte Carlo code using ENDF/B-VI continuous energy cross sections. The MCNP5 code is widely used in these types of applications and the staff found it is appropriate for this application because it is listed as an acceptable code in Section 6.5.3.3, “Computer Codes and Cross Section Libraries,” of NUREG-1617 and meets the requirements described in Section 4.1, “Computer Code System,” of NUREG/CR-5661.

In Section 6.7.2.2 of the application, NAC states that it is using cross section data from the ENDF/B-VI library. Section 6.5.5.1 of NAC-LWT application states that these are the same cross sections used to perform the validation of the code. The ENDF data is considered acceptable per the guidance in Section 4.2, “Cross Sections and Cross Section Processing,” NUREG/CR-5661. Therefore, the staff found the cross sections used are appropriate for use with the NAC-LWT NRU/NRX application.

The staff verified that the applicant provided representative input and output files. The staff also verified that the information regarding the model configuration, material properties and cross sections were properly represented in the input files. The staff reviewed the key input data for the criticality calculations specified in the input files and found them acceptable. The staff viewed the output files provided and determined that they have proper convergence and that the calculated k_{eff} values from the output files agree with those reported in the text.

6.3.4 Demonstration of Maximum Reactivity

The applicant determined that the NRU HEU fuel is the most reactive in the NAC-LWT package. The applicant performed an evaluation comparing the NRU HEU to the NRU LEU fuel. The applicant modeled both fuel enrichments with the most reactive configuration and found the LEU fuel to be less reactive (Section 6.7.2.6 of the NAC-LWT application). The applicant did not model other moderator or fuel configurations of the LEU fuel. The staff found this acceptable since the NAC-LWT is a thermal system and both LEU and HEU fuel have the same U-235 mass, therefore increasing the U-238 mass causes the fuel to be less reactive. The presence of

U-238 would increase absorption and displace moderator. Figure 6.7.2-16 of the NAC-LWT SAR shows that the system is more reactive with more moderator inside the fuel tubes.

The applicant compared the NRU HEU fuel to the NRX fuel and found the NRU HEU fuel to be more reactive. The NRX fuel has a higher fissile mass and would be inherently more reactive, however, the applicant requires the use of the fuel caddy (Drawing LWT 315-40-175, Rev. 1) inside the fuel tubes when loading this fuel type. Since the caddy restricts the geometry of the fuel, this brings down the reactivity of the NRX fuel below that of the NRU HEU fuel.

The applicant proposed to load undamaged fuel in the form of loose fuel rods or intact fuel assemblies. NAC performs an analysis for nominal pitch (in-core configuration), the most reactive pitch and loose rod segments. The model for the loose rod segments has the maximum number of rod segments that would fit inside the tube or caddy. Since this is more rods than an actual fuel assembly, the assembly is truncated to preserve the fuel mass. The applicant found the most reactive pitch. The results of the applicant's calculations are presented in Figures 6.7.2-12 and 6.7.2-13 of the NAC-LWT application.

The applicant performed moderator density studies to determine the most reactive configuration. The results of the applicant's calculations are presented in Figures 6.7.2-10 and 6.7.2-11 of the NAC-LWT application. NAC demonstrated that the most reactive moderator configuration is full moderator density in the fuel tubes and package cavity, with a flooded package exterior and loss of the neutron shield

The staff found that the applicant's analysis demonstrated that they have found the maximum reactivity per the requirements of 10 CFR 71.55(b)and (e).

6.3.5 Confirmatory Analysis

The staff performed independent calculations of the NAC-LWT package. The staff used the SCALE 6.1 code package. The staff used KENO VI with ENDF/B-VII continuous energy cross sections. The staff modeled a single package under hypothetical accident conditions with the NRU HEU fuel assuming a broken rod geometry and full in-leakage of water inside the tubes and package cavity.

The staff confirmed the k_{eff} results from the applicant. The staff made several simplifying assumptions to its model and given these changes, the staff's k_{eff} verifies that of the applicant's. The staff found that this verification helps demonstrate that the package and contents are subcritical and were adequately described in the application and associated drawings.

6.4 Single Package Evaluation

6.4.1 Configuration

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 modeled the most reactive fuel in a conservative broken rod configuration consistent with the condition of the package and the chemical and physical form of the contents.

The applicant modeled full water moderation inside the tubes and package cavity and this is in the most reactive extent and satisfies the requirements of 10 CFR 71.55(b). The NAC-LWT single package analyses included reflection of 20 cm water on all sides. "Full reflection" is

30 cm per the recommendations in NUREG/CR-5661; however, the staff performed a sensitivity study that showed that the difference between 20 and 30cm reflection for a fully flooded single package NAC-LWT model with NRU HEU fuel is within the uncertainty of the calculation and is therefore insignificant. The staff found that the 20-cm reflector meets the requirement in 10 CFR 71.55(b)(3).

6.4.2 Results

6.4.2.1 Normal Conditions of Transport

The staff confirmed that the results of the applicant's criticality calculations are consistent with the information presented in Table 6.7.2-6 of the application. The maximum k_{eff} for a single package under normal conditions of transport is 0.92525. Since k_{eff} is less than the USL of 0.9270 after the tests specified in 10 CFR 71.71, the staff verified that this meets the requirements of 10 CFR 71.55(d)(1) which requires that the contents be subcritical.

Since the applicant performs criticality evaluations using the broken rod model, the staff found that this is a reasonably bounding geometry of the fuel, and therefore the staff found that the geometric form of the package contents could not be altered in such a way that would affect the conclusions from the criticality safety analyses. The staff found that the applicant meets the requirements in 10 CFR 71.55(d)(2).

The applicant performed calculations where moderation is present to such an extent to cause maximum reactivity consistent with the chemical and physical form of the material. The staff found that this meets 10 CFR 71.55(d)(3).

Under the tests specified in 10 CFR 71.71, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention including (1) the total volume of the packaging will not be reduced on which the criticality safety is assessed, (2) the effective spacing between the fissile contents and the outer surface of the packaging is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm cube. The staff found that this meets the requirements in 10 CFR 71.55(d)(4).

6.4.2.2 Hypothetical Accident Conditions

The staff confirmed that the results of the applicant's criticality calculations are consistent with the information presented in Table 6.7.2-6 of the application. The maximum k_{eff} for a single package after the tests for hypothetical accident conditions is 0.92560. Since the k_{eff} is less than the USL of 0.9270 under the tests specified in 10 CFR 71.73, the staff verified that this meets the requirements of 10 CFR 71.55(e), which requires that after the tests for hypothetical accident conditions the contents be subcritical.

The staff verified that (1) the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents, (2) water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents, and (3) there is full reflection by water on all sides, as close as is consistent with the damaged condition of the package. This meets the requirements of 10 CFR 71.55(e)(1) through (3).

6.5 Evaluation of Package Arrays

6.5.1 Configuration

The applicant specified a CSI of 100; therefore, the array calculations are the same as a single package for hypothetical accident conditions. For normal conditions of transport, the applicant modeled an infinite array by applying a reflective boundary condition at the surface. This configuration assumes no inleakage of water and nothing between the packages. The applicant used the broken rod fuel model for this evaluation. This model was determined to be most reactive under wet conditions. Even though this determination was not made independently for dry conditions, the staff found the fuel model acceptable because the k_{eff} is so low (0.07690) without moderator, the system will remain subcritical even with a different fuel pitch.

6.5.2 Results

6.5.2.1 Normal Conditions of Transport

The maximum k_{eff} for the normal conditions of transport array analyses is 0.07690. Since k_{eff} for an infinite array is less than the USL of 0.9270 under the tests specified in 10 CFR 71.71, the staff verified that this meets the requirements of 10 CFR 71.59(a)(2) which requires that an array size 5N of undamaged packages be subcritical.

6.5.2.2 Hypothetical Accident Conditions

The applicant did not perform calculations for an array size greater than one. The staff found this acceptable because the CSI is 100. The staff verified that this meets the requirements of 10 CFR 71.59(a)(2) which requires that an array size 2N of packages under hypothetical accident conditions be subcritical.

6.6 Benchmark Evaluations

The applicant performs the criticality evaluations using the MCNP5 v1.60 three-dimensional Monte Carlo code and continuous energy cross sections. The applicant performed benchmarks with the same computer code and cross section set.

6.6.1 Experiments and Applicability

The applicant performed benchmark comparisons and determined a USL based on the guidance published in NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages.” The staff found the use of this guidance acceptable.

The applicant uses experiments from the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The experiments are listed in Table 6.5.5-1 of the NAC-LWT SAR. The staff found that these are appropriately referenced.

The staff verified that the following important design parameters for the NRU/NRX fuel in the NAC-LWT system were within the benchmark experiments cited by the applicant.

- Enrichment
- Type of fissile material

- Fuel rod pitch and diameter
- Energy of the Average Lethargy Causing Fission (EALCF)

The applicant analyzed 94% enriched fuel, while the maximum enrichment in the experiments was 93.2%. Per Table 6.5.5-5 of the application the USL was reduced to extend the applicability to lower enrichments. For the enrichment range from 17 to 93.2% the minimum USL is 0.928. The USL is slightly negatively correlated with enrichment and if extrapolated to 100% enrichment is 0.92781 which is bounded by the USL of 0.9270. Therefore, the staff has reasonable assurance that the USL of 0.9270 bounds the USL at this slightly increased enrichment of 94%. The staff verified that the selected critical experiments include uranium-aluminum fuel with aluminum clad. The fuel rod pitch and diameter are within the range of selected benchmarking experiments.

The range for EALCF for the experiments used in determining the bias is 0.05 eV to 0.4 eV with most experiments falling between 0.05 eV to 0.15 eV. The EALCF of the NRU/NRX criticality calculations is 0.123 eV.

6.6.2 Bias Determination

The applicant calculated a USL of 0.9270 using the USLSTATS code. The staff found this acceptable because this includes the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95.

The NAC-LWT SAR's previous benchmarking evaluation for high-enriched research reactor fuel was based on MCNP5 v1.30 using the same experiment base gave a USL of 0.9171. The change in USL comes from the applicant restricting the data used in determining the USL to EALCF energies up to 0.4 eV rather than 1.2 eV used in the benchmarking for MCNP v1.30. The staff found this acceptable for the NAC-LWT NRU/NRX application as the EALCF for this application is within this range.

6.7 Burnup Credit

The applicant does not request credit for burnup.

6.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the subcriticality requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES EVALUATION

The staff reviewed Chapter 7 of the application to verify that it meets the requirements of 10 CFR Part 71 and is adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

The chapter includes the procedures for package loading, unloading, and preparation of the empty package for transport. To support this revision request, Sections 7.1.13 of the application was added to include the procedures for dry loading of the NRU/NRX fuel. Section 7.2.4 was revised to include procedures for unloading the NRU/NRX fuel.

The staff reviewed and evaluated the proposed loading and unloading procedures of the NRU/NRX fuel. Based on the statements and representations in the application, the staff concluded that the package operations meet the requirements of 10 CFR Part 71, and that they are adequate to assure the package will be operated in a manner consistent with its evaluation for approval. Further, the certificate is conditioned to specify that the package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the SAR, as amended.

8.0 ACCEPTANCE TESTS AND MAINTENANCE REVIEW

To support this revision request, no changes were made to Chapter 8 of the SAR. The staff concludes that the package continues to meet the requirements of 10 CFR Part 71.

CONDITIONS

In addition to the new and revised drawings in Condition 5.(a)(3)(ii), the following changes have been made to the Certificate:

Condition 5.(b)(1)(xix) was added to specify the type and form of the NRU and NRX fuel assemblies.

Condition 5.(b)(2)(xx) was added to specify the quantity of material per package for the NRU and NRX fuel assemblies.

Condition 5.(c) was modified to specify the CSI for the NRU and NRX fuel.

Condition 19 is being retained to support planned international shipment under Revision 55 of the certificate.

Condition 20 was renumbered to condition 21 and a provision was added to allow the use of Revision 57 of the certificate for approximately 1 year.

CONCLUSION

CoC No. 9225 has been revised to authorize shipment of NRU/NRX fuel as specified above in the Model No. NAC-LWT package. Based on the statements and representations in the application, and with the conditions listed above, the staff agrees that this authorization does not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued on 2/28/13.