September 10, 2009

Mr. Patrick L. Paquin
General Manager – Engineering
and Licensing
Energy*Solutions*
140 Stoneridge Drive
Columbia, SC 29210

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9204 FOR THE MODEL NO. 10-160B

Dear Mr. Paquin:

As requested by your application dated December 20, 2007, supplemented on October 3, 2008, May 1, June 11, and August 10, 2009, enclosed is Certificate of Compliance No. 9204, Revision No. 13, for the Model No. 10-160B package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

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. 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 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me or Michele Sampson of my staff at (301) 492-3300.

Sincerely,

/RA/

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9204
TAC No. L24162

Enclosures:
1. Certificate of Compliance No. 9204, Rev. No. 13
2. Safety Evaluation Report
3. Registered Users

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

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OFC:	SFST	E	SFST	E	SFST		SFST		SFST		SFST	
NAME:	M. Sampson		J. Chang		M. Waters		V. Wilson		L. Campbell		M. DeBose	
DATE:	08/24/09		09/01/09		09/04/09		09/01/09		09/05/09		09/09/09	
OFC:	SFST		SFST		SFST		SFST		SFST			
NAME:	E. Benner											
DATE:	09/10/09											

SAFETY EVALUATION REPORT
Docket No. 71-9204
Model No. 10-160B
Certificate of Compliance No. 9204
Revision No. 13

SUMMARY

By application dated December 20, 2007, as supplemented on October 3, 2008, and May 1, June 11, and August 10 and 25, 2009, Energy*Solutions* requested an amendment to Certificate of Compliance (CoC) No. 9204, for the Model No. 10-160B package. Energy*Solutions* requested a “-96” designation to the package identification number, the addition of fissile material to the authorized contents, and increase of the maximum decay heat for contents. In support of the request, the applicant provided a consolidated application as specified in 10 CFR 71.38(c). The staff reviewed the consolidated application and concluded that the application incorporated the changes that were previously referenced in the CoC.

NRC staff reviewed the application 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 staff finds that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1. Package Description

The applicant has proposed no changes in the Model No. 10-160B package design. To reflect changes in the company name, the title block on Drawing No. C-110-D-29003-010, sheets 1 through 5, has been updated from Duratek to Energy*Solutions*. Additionally, the model number has been revised from Model No. CNS 10-160B, to Model No. 10-160B. The revised model number has been reflected in the Certificate.

1.2.1 Contents

Energy*Solutions* has evaluated the addition of a plutonium-239 (Pu-239) beryllium (Be) neutron source, in special form, and the addition of up to the fissile gram equivalent (FGE) of 325 grams of Pu-239, as determined using the conversion factors in Table 9.1.3, in Chapter 4, Appendix 4.10.2, of the application. The fissile contents are requested to expand the quantities of transuranic-containing waste authorized for transport in the package. The Certificate has been revised to reflect the package type as B(U)F.

Changes to the regulatory requirements for plutonium in 10 CFR 71.63 have been reflected in the authorized contents for the Model No. 10-160B. Plutonium content exceeding 0.74 terabecquerels (TBq) (20 curies (Ci)) must be in solid form.

The maximum decay heat for the package has also been increased from 100 watts to 200 watts.

1.3 Evaluation for the -96 Designation

The applicant requested an amendment to Certificate of Compliance No. 9253 to revise the package identification number from USA/9204/B(U)-85 to USA/9204/B(U)F-96, as specified in 10 CFR 71.19(e). To support the request for the "-96" designation, the applicant provided a table addressing the changes in 10 CFR Part 71 as a result of the rulemaking process that resulted in the revised rule, which was published on January 26, 2004 (69 FR 3698). The staff evaluated the applicant's request, and summarized the impact of the 19 issues proposed in the rulemaking, as described below.

- Issue 1, Changing Part 71 to the International Systems of Units (SI) Only. This proposal was not adopted in the final rule, and therefore no changes were needed in the package application or the Certificate of Compliance to conform to the new rule.

- Issue 2, Radionuclide Exemption Values. The final rule adopted radionuclide activity concentration values and consignment activity limits in TS-R-1 for the exemption from regulatory requirements for the shipment or carriage of certain radioactive low-level materials. In addition, the final rule adopted an exemption from regulatory requirements for certain natural material and ores containing naturally occurring radionuclides. The applicant identified no changes to the Model No. 10-160B package as a result of this revision. The staff agrees, based on the design purpose of the Model No. 10-160B package and the allowed contents specified in the certificate. Thus, no changes were needed to conform to the new rule.

- Issue 3, Revision of A_1 and A_2. The final rule adopted changes in the A_1 and A_2 values from TS-R-1, with the exception of two radionuclides. The A_1 and A_2 values were modified in TS-R-1 based on refined modeling of possible doses from radionuclides, and the NRC agreed that incorporating the latest in dosimetric modeling would improve transportation regulations. The applicant stated that this change was not applicable to the Model No. 10-160B. Thus, no changes were needed to conform to the new rule.

- Issue 4, Uranium Hexafluoride (UF_6) Package Requirements. The Model No. 10-160B is not authorized for the transport of uranium hexafluoride. Therefore, no changes were needed to conform to the new rule.

- Issue 5, Criticality Safety Index (CSI). The final rule adopted the new term Criticality Safety Index from TS-R-1. The applicant added fissile contents in this application, and included the appropriate CSI. The Certificate has been revised to reflect the CSI and to identify the package type as B(U)F.

- Issue 6, Type C Packages and Low Dispersible Material. This proposal was not adopted for the final rule. Thus, no changes were needed.

- Issue 7, Deep Immersion Test. The final rule adopted an extension of the previous version of 10 CFR 71.61 from packages for irradiated fuel to any Type B package containing activity greater than 10^5 A_2. The contents for the Model No. 10-160B are limited to 3000 Type A quantity. Thus, no changes were needed to conform to the new rule.

- Issue 8, Grandfathering Previously Approved Packages. The final rule adopted a process for allowing continued use, for specific periods of time, of previously approved package designs without demonstrating compliance to the final rule. The applicant has decided in accordance with 10 CFR 71.19(e) to submit information demonstrating compliance with the final rule. Thus, grandfathering the design of the Model No. 10-160B package is not necessary.

- Issue 9, Changes to Various Definitions. The final rule adopted several revised and new definitions. These changes were adopted to provide clarity to Part 71. Thus, no changes were needed to conform to the new rule.

- Issue 10, Crush Test for Fissile Material Packages. The revised 10 CFR 71.73 expanded the applicability of the crush test to fissile material packages. The crush test is required for packages with a mass not greater than 500 kilograms (1100 pounds). Since the Model No. 10-160B package has a mass greater than this, the crush test is not applicable. Therefore no changes were needed to conform to the new rule.

- Issue 11, Fissile Material Package Design for Transport by Aircraft. The final rule adopted a new section, Section 71.55(f), which addresses design requirements for packages transporting fissile material by air. The Model No. 10-160B has not been tested to the requirements of 71.55(f); therefore, the Certificate of Compliance has been conditioned to specify that air transport is not authorized for fissile material.

- Issue 12, Special Package Authorization. The final rule adopted provisions for special package authorization that will apply only in limited circumstances and only to one-time shipments of large components. This provision is not applicable to the Model No. 10-160B package. Thus, no changes were needed to conform to the new rule.

- Issue 13, Expansion of Part 71 Quality Assurance (QA) Requirements to Certificate Holders. Energy*Solutions* is the holder of Certificate of Compliance No. 9204, and has a NRC approved quality assurance program. No changes are needed to conform to the new rule.

- Issue 14, Adoption of the American Society of Mechanical Engineers (ASME) code. This proposal was not adopted in the final rule. Thus, no changes were needed to conform to the new rule.

- Issue 15, Change Authority for Dual-Purpose Package Certificate Holders. This proposal was not adopted for the final rule. Thus, no changes were needed to conform to the new rule.

- Issue 16, Fissile Material Exemptions and General License Provisions. The final rule adopted various revisions to the fissile material exemptions and the general license provisions in Part 71 to facilitate effective and efficient regulation of the transport of small quantities of fissile material. The criticality safety of the Model No. 10-160B package does not rely on limiting fissile materials to exempt or generally licensed quantities. Therefore, no changes were needed to conform to the new rule.

- Issue 17, Double Containment of Plutonium. The final rule removed the requirement that packages with plutonium in excess of 0.74 TBq (20 Ci) have a second, separate inner container. The application does not refer to the double containment of plutonium. The Certificate of Compliance specifies that plutonium in excess of 0.74 TBq must be in solid form.

- Issue 18, Contamination Limits as Applied to Spent Fuel and High Level Waste Packages. This proposal was not adopted for the final rule. Thus, no changes were needed to conform to the new rule.

- Issue 19, Modification of Events Reporting Requirements. The final rule adopted modified reporting requirements. While the final rule is applicable to the package, no changes were needed to conform to the new rule.

The staff concluded that the design has been adequately described and meets the requirements of the revised regulations in 10 CFR Part 71. The Certificate of Compliance has been revised to include the "-96" designation in the package identification number. To allow time to modify the packaging markings to include the "-96" designation, the certificate has been conditioned to allow use of packages marked with the "-85" designation for a period of approximately one year. After October 31, 2010, the packaging must be marked with the package identification number including the "-96" designation.

2.0 STRUCTURAL

The applicant has proposed no structural changes to the Model No. 10-160B package design.

3.0 THERMAL

3.1 Review Objective

The applicant, Energy*Solutions*, requested the addition of fissile contents and the increase of heat load limit for the Model No. 10-160B from 100 to 200 watts. The objective of this review is to verify that the package design and performance, with these requested changes, still satisfy the thermal requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

3.2 Thermal Design Evaluation

The applicant did not change the thermal design of the Model No. 10-160B transportation package with the addition of fissile material to the authorized contents. The Model No. 10-160B container uses the impact limiter and the fire shield for thermal protection of the cask body. The impact limiters provide the protection to the top and bottom of the cask and the fire shield protects the side walls between the impact limiters. The impact limiters are sheet metal enclosures filled with polyurethane foam which acts as a thermal insulation to heat flow. The fire shield is a 0.104-inch thick steel plate with a 0.156-inch thick air gap between it and the outer shell of the cask. During HAC fire accident, both fire shield and air gap provide a thermal barrier which impedes the transfer of heat from the fire shield to the cask. There are no special devices and coolants used for the transfer or dissipation of heat.

The heat transfer from the fire source takes place by a combination of radiation and forced convection. The total heat flow rate is a function of thermal resistance provided by the air gap and the equivalent thermal resistance of radiation heat transfer between the fire shield and the cask outer shell. The applicant selected the appropriate nodes at suitable locations in the computer model to represent the locations of both primary and secondary O-ring seals, and then predict the temperatures of the corresponding O-ring seals at these locations. The staff reviewed Chapter 1, “General Information”, and Chapter 3, “Thermal Evaluation”, of the application, and agreed that the application provided a significant description of design features and operating characteristics of the package and a sufficient basis for thermal evaluation of the package is provided to satisfy 10 CFR 71.31, 71.33, and 71.35.

3.2.1 Material Properties

The applicant listed all temperature-dependent thermal properties of materials used in the construction of the Model No. 10-160B cask, such as stainless steel, carbon steel, and lead; and the thermal properties of air in Chapter 3, Tables 3.3(a) and 3.3(b) of the application. The elastomer seals chosen for use can be butyl rubber, ethylene propylene rubber, or silicone rubber which all have thermal properties such that the usable temperature range meets or exceeds the range required for NCT (minimum -40^{o}F and maximum 250^{o}F) and for HAC (+400^{o}F for 1 hour). The staff reviewed and verified the material properties listed in the tables, and confirmed that the material properties used in the thermal model are appropriate.

3.2.2 Analysis Model

The applicant performed a 2-D analysis of the Model No. 10-160B during NCT and HAC by using the ANSYS finite thermal model of the cask. The ANSYS axisymmetric model was solved using temperature-dependent material properties, bounding cask geometry, and associated heat transfer equations. ANSYS has been demonstrated as one of the standard codes to be appropriate for cask thermal evaluation in accordance with the requirements of 10 CFR 71.31(c).

The applicant modeled the design heat load of 200 watts within the cask as a cavity surface heat load and simulated the air gap within the fire shield to be exactly equal to the nominal wire diameter (5/32-inch) during NCT and HAC, that is consistent with the existing wires wrapped around the cask outer shell. The staff accepted this assumption and approach used for thermal simulation of the gap after reviewing the related description of the welding configuration and the license drawings, C-110-D-29003-010.

The staff reviewed calculation package CSG 01.1000, Rev. 2, to evaluate the thermal performance of the Model No. 10-160B package during HAC fire accident with new emissivity factors that satisfy 10 CFR 71.73 requirements. The applicant used the surface emissivity of 0.8, an ambient temperature of 100^{o}F, and natural convection heat transfer for NCT analysis. The applicant used a conservative overall emissivity of 0.9 between fire shield and the environment, a fire temperature of 1475^{o}F, forced convection heat transfer and zero insolation during the HAC 30-minute fire transient. The applicant then applied a surface emissivity of 0.8, a hot temperature of 100^{o}F, and the natural convection during 48-hour cool-down period after HAC. The solar heat of 320 gcal/cm^2, used in NCT and cooldown of HAC is less conservative than the average insolation for a 12-hour period, but still acceptable for thermal evaluation because of its limited effect in heat transfer. The staff reviewed the methodology and agreed the load steps of both NCT and HAC analyses (30-min transient and 48-hour cool-down) are described in sufficient detail to meet the thermal regulatory requirements of 10 CFR 71.35, 71.71, and 71.73.

The staff notes that for any future amendment requests, the applicant should update the application's Chapter 3, "Thermal Evaluation", and Chapter 2.7.3, "Thermal of Structure Evaluation", to be consistent with the revised calculation package CSG-01.1000, Rev. 2, which uses surface emissivity of 0.9 and fire temperature of 1475^{o}F for HAC fire analysis.

3.2.3 Internal Pressures of NCT and HAC

The applicant assumed the gas within the cask behaves as an ideal gas and the inside surface of the cask is dry, and calculated a maximum internal pressures of 12.22 psig under NCT and 15.42 psig during HAC fire accident. The staff reviewed the stress effects of combined thermal and pressure loading in Attachment 5 to Chapter 2, Structure Evaluation, of the application, and verified that both maximum pressures under NCT and HAC are below the design limit of 31.2 psig. The staff confirmed that the package pressure performance meets the requirements of 10 CFR 71.35, 71.71, and 71.73.

3.2.4 Thermal Stress

The applicant predicted a maximum average cask wall temperature of 289^{o}F with the maximum temperature differences of 39.8^{o}F across the cask body (between the outside surface of the outer shell and the inside surface of the inner shell), 20.2^{o}F across the outer shell, and 2.3^{o}F across the inner shell. The temperature gradients, estimated by dividing the temperature difference with the maximum average cask wall temperature, are very small. Section 2.7.3 of the application summarizes the data of temperatures, pressures, and thermal expansions, used for thermal stress calculation. The staff confirmed that the thermally-induced stresses during HAC fire accident are below the allowable limits and satisfy the performance requirements of 10 CFR 71.35 and 71.73.

3.2.5 Maximum Temperatures of NCT and HAC

The applicant provided the maximum component temperatures in Table 2 and the temperature distributions in Figures 8-10 of package CSG-01.1000, Rev. 2, for NCT. The maximum temperatures of O-ring seal and the lead are 174^{o}F and 173^{o}F, which are well below the allowable limits of 250^{o}F and 620^{o}F, respectively. There is no lead melting and the cavity air pressure is maintained within the design limit during NCT. The maximum accessible surface temperatures of the lid and the fire shield are 175^{o}F and 171^{o}F, respectively, which are in compliance with 10 CFR 71.43(g).

The applicant displayed the maximum calculated component temperatures in Table 3 of package CSG-01.1000, Rev. 2, and commented that the material temperatures fall within acceptable limits of HAC fire case. The maximum temperatures of lid, inner shell, outer shell, and fire shield, all made of carbon steel, are far below the maximum allowable limit of 2600^{o}F (melting point). The maximum O-ring seal temperature of 164.5^{o}F is far below the required limit of 400^{o}F and no lead melts with its maximum temperature of 274.4^{o}F far below the melting point of 620^{o}F. The temperature 273.8^{o}F of cavity air results in a gas pressure within the design limit.

The staff validated the data in Tables 2 and 3 and Figures 8-17 of CSG-01.1000, Rev. 2, and concluded that the cask body is well below its service limits for both NCT and HAC cases and meets the requirements of 10 CFR 71.71 and 71.73.

3.3 Evaluation Findings

Based on the review of the statements and representations in the application, the staff confirmed that 1) the thermal design features are adequately described and evaluated, 2) the thermal analyses are discussed and described in sufficient detail for validation of NCT and HAC, and 3) the corresponding thermal features (such as temperatures, pressures, and thermal stresses) fall within the safety margins. The staff concluded that with maximum decay heat of 200 watts, the thermal performance of Model No. 10-160B package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT

4.1 Review Objective

The applicant, Energy*Solutions*, requested the addition of fissile contents and the increase of heat load limit for the Model No. 10-160B from 100 to 200 watts. The objective of this review is to verify that the package design and performance, with these requested changes, still satisfy the containment requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

4.2 Package Description Evaluation

The containment design was not changed in this request and was previously approved by NRC. The containment vessel is defined as the inner shell, the primary and secondary lids, together with the associated O-ring seals and lid closure bolts. The shell is fabricated of an outer shell of steel plate, a layer of lead, and an inner shell of steel. The containment vessel is fabricated from steel with full penetration welds. The cylindrical shell is attached at the base to a circular plate with full penetration welds. A stainless steel liner is welded to the cask cavity surface and the lid surface to protect all accessible areas from contamination. A steel thermal shield is welded to the exterior barrel of the cask as protection during the fire accident. The primary lid is fastened to the cask body by bolts and the secondary lid is attached to the primary lid with bolts. The primary lid is sealed with two high-temperature elastomeric O-rings in the machine grooves, and the secondary lid is also sealed with two high-temperature elastomeric O-rings in the machined grooves. There are two penetrations, the drain port and the vent port, of the containment vessel. The drain port is located at the cask base and the vent port penetrates the secondary lid into the main cask cavity. Both the vent port and the drain port are sealed with a seal and a cap screw.

The staff reviewed the containment design features in Chapter 1, General Information, and Chapter 4, Containment, of the application and concluded that the containment system, including containment vessel, welds, seals, bolts, lids, and cover plates, are consistently defined in General Information and Containment chapters; and the vent and drain boundary penetrations and their method of closure are adequately described. The staff ensured that all components of the containment system are displayed in the drawings C-110-D-29003-010, sheets 1 - 5, Rev. 14, and agreed the Model No. 10-160B package description is in compliance of 10 CFR 71.33.

4.2.1 Source Term Analysis

The applicant stated that the methodology of source term analysis was not changed and has conservatively assumed the revised contents in the package are at the maximum allowed in the containment analysis. The staff agreed that the requested content change does not affect the source term assumption in the release analysis. Therefore the content change satisfies 10 CFR 71.31 and 71.71.

4.2.2 Thermal Evaluation of Containment Components

The applicant performed a thermal analysis with a heat load of 200 watts and predicted the temperature profiles of NCT with maximum ambient temperature conditions and HAC fire accident in Chapter 3 (Calc. Pkg. No. CSG-01.1000) of the application. The staff reviewed the applicant's Thermal Evaluation and concluded that the maximum temperatures of seals, bolts, and other key containment components are below the allowable temperature limits and are remained within the safety margins under NCT and HAC, without relying on any mechanical cooling system. The staff verified that the thermal performance of the Model No. 10-160B package with a heat load of 200 watts satisfies 10 CFR 71.71 for NCT and 10 CFR 71.73 for HAC.

4.2.3 Leak Test Evaluation

The applicant revised the calculation for allowable leakage rate based on the change in contents. The applicant specified, in Chapter 4, Section 4.2.1, "Leak Test Requirements", that the calculated maximum allowable leakage rate is 2.57×10^{-6} cm^3/sec according to the release analysis under NCT pressures and temperature. The staff verified that the calculated maximum allowable leakage rate of 2.57×10^{-6} cm^3/sec during NCT, is equivalent to the reference air leakage rate of 3.25×10^{-6} ref-cm^3/s during NCT. The staff also verified that the calculated allowable leakage rate 1.53×10^{-2} cm^3/s during HAC is larger than the reference air leakage rate 3.25×10^{-6} ref-cm^3/s during NCT. Therefore, the staff agreed that the leak rate of NCT is the limiting criteria to determine the maximum allowable leak rate of leak tests for both NCT and HAC and the leak test must be able to detect leaks of 3.25×10^{-6} ref-cm^3/s (based on dry air at 25^{o}C with a pressure difference of 1 atm) to assure there would be no loss or dispersal of radioactive contents to a sensitivity of 10^{-6} A_2 per hour, in compliance with 10 CFR 71.51(a)(1) and (a)(2).

4.2.4 Helium Leak Test for Elastomeric Silicone Rubber O-Ring Seals

The applicant revised Chapter 4, Section 4.9, to indicate that the helium leak test is only applicable when butyl rubber O-rings and ethylene propylene seals are used in the Model No. 10-160B. The staff determined that using the helium leak test for the elastomeric silicone rubber O-rings seals could generate measurement errors from the combination of leakage and permeation through the seals. The staff reviewed Section 4.9, and confirmed that the described helium leak test method meets the requirements of ANSI N14.5 and 10 CFR 71.51. The Certificate has been conditioned to preclude use of the helium leak test when elastomeric silicone rubber O-ring seals are in use on the package.

4.2.5 Inerting Method and Payload Control for TRU Waste

The applicant revised in Chapter 4, Section 4.8(ii), to state "The secondary container and the cask cavity must be inerted with a diluent to assure the oxygen, including that radiolytically generated, shall be limited to 5% by volume in those portions of the package which could have hydrogen greater than 5%. The criterion does not apply to the TRU wastes, which shall be governed by the requirements of Appendix 4.10.2" to eliminate the unintended inconsistency. The staff accepted this revision to Section 4.8(ii) for TRU wastes and confirmed that the intent of the gas generation methodology, described in Appendix 4.10.2, is to directly limit the hydrogen generation under 5% in TRU wastes. The staff also confirmed the gas generation methodology, described in Appendix 4.10.2, satisfies 10 CFR 71.43(d).

The applicant initially requested a change to give the package user responsibility for determining the acceptability of TRU waste from a particular site. The site specific evaluation would have been performed against the payload parameter restrictions specified in Appendix 4.10.2 of the application, and documented in a TRU Payload Assessment by the 10-160B Payload Engineer. NRC staff review and approval would not have been required prior to transport of the TRU waste. However, the staff determined that Appendix 4.10.2, did not contain sufficient information to provide stand-alone guidance in its current form. The applicant withdrew the requested change, and may submit revisions to Appendix 4.10.2 at a later date.

4.2.6 Gas Distribution and Pressure Buildup for TRU waste

The applicant identified that the primary mechanism for gas generation during TRU waste payload transportation in the Model No. 10-160B package is by radiolysis of the waste materials, and the gas generation from other mechanisms such as chemical, thermal, or biological activity is not significant for TRU waste payload, given the transportation time, the nature of the waste (solid or solidified), and the environment of the payload (lack of water contents). The staff reviewed Chapter 4, the main tier Appendix and sub tiers Appendices, and concluded that the gas pressure, generated in the payload and released into the inner vessel (IV) cavity, will be controlled below the design limit of 31.2 psig within the IV cavity to comply with 10 CFR 71.35, 71.71 and 71.73.

4.2.7 Hydrogen Gas Generation for TRU Waste

The applicant provided the requirement and methodology of hydrogen gas generation for TRU wastes in Appendix 4.10.2, and the site specific compliance methodology of Appendices 4.10.2.1 to 4.10.2.5 to ensure that the hydrogen gas concentration shall not exceed 5% by volume in all void volumes within the Model No. 10-160B cask during its shipping period. The staff reviewed the application and verified that the methodology to evaluate the hydrogen gas generation is not changed due to the addition of up to 325 FGE of fissile materials, including solid plutonium in excess of 20 Ci. The staff also identified that the criteria and methods used to comply with the limit of hydrogen gas concentration for TRU waste are consistent between main tier and sub tier appendices:

- The decay heat limit compliance method is used when the G-values are available and the flammable volatile organic compounds (VOCs) are less than 500 particles per million (ppm);

- The headspace sampling method is used if the process knowledge can not show that the concentration of flammable VOCs is less than 500 ppm; and

- The headspace sampling method is used if the decay heat is greater than the decay heat limit.

The staff verified that the allowable hydrogen generation rate and its calculation for a given content code and for each site-specific content code have been reported in the site-specific payload compliance appendix (sub tier to main tier Appendix 4.10.2), in compliance with the main tier Appendix 4.10.2, and 10 CFR 71.33, 71.35, and 71.43(d).

4.3 Evaluation Findings

Based on the review of the statements and representations (Chapters 1, 4, 7, 8, Appendix 4.10.2, and Appendices 4.10.2.1-5) in the application, the staff concluded that the containment design of the Model No. 10-160B package has been adequately described, and evaluated per the change of contents and the package design still meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING

Energy*Solutions* documented the shielding evaluation in Section 5 of the application. The applicant is changing the approved contents to include up to 325 FGE of fissile material. The shielding analysis was revised to include a neutron source. In addition the applicant updated the analysis using the SCALE programs. The staff's review is limited to the changes in approved contents. Therefore the staff has not reviewed the revised gamma shielding analyses.

The Model No. 10-160B (formerly called the CNS 10-160B) shielding analysis was previously approved by the staff during the original review of the design and was documented in a staff safety evaluation report dated November 2, 1990 (Reference 5-1). Since then, it had undergone many revisions including an increase in radioactive material contents and the shielding analysis was again reviewed and was approved by the staff and documented in the safety evaluation report dated August 10, 2001 (Reference 5-4).

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The shielding design features of the Model No. 10-160B are 1-7/8-inches of lead surrounded by 1-1/8-inches of carbon steel on the inside and 2 inches of carbon steel shield on the outside and 5 ½ inches of carbon steel at the ends. There are no materials specifically used for neutron shielding. The tests and analyses performed under Chapters 2.0 and 3.0 of the application demonstrated the ability of the containment vessel to maintain its shielding integrity under normal conditions of transport. The applicant does recognize that along the cask wall lead slump may occur and accounts for this effect in its HAC analyses.

5.1.2 Codes and Standards

The applicant identifies the appropriate regulations in 10 CFR Part 71, in Chapter 5 of the application. The staff also verified that the applicant appropriately identified the ANSI/ANS 6.1.1 1977 version for their flux to dose rate conversion factors.

5.1.3 Summary Table of Maximum Radiation Levels

The staff examined the summary table in Table 5.1 of Chapter 5 of the application. This summary table differs from the previously approved version because the applicant used a different evaluation method and also added a neutron source to the table. The staff reviewed this table to ensure that the applicant meets the requirements in 10 CFR 71.47 and 71.51. Since the applicant states that the Model No. 10-160B will be operated under "exclusive use," the radiation levels must not exceed those specified in 10 CFR 71.47(b).

The staff verified that the summary table states that the limit of 200 mrem/h will not be exceeded on the external surface of the package. The applicant evaluated the maximum neutron source and the maximum gamma sources separately. The package user must perform dose rate measurements to ensure that regulatory limits in 10 CFR 71.47(b)(1) are met if a combination of neutron and gamma sources are shipped together.

Although the applicant did not show dose rates at the outer vehicle surface in the summary table, the table does show that there will be less than 200 mrem/h dose rate at the surface of the package, therefore 10 CFR 71.47(b)(2) is also met. This regulation requires that the dose rate be limited to 200 mrem/h at the vehicle surface.

The staff verified that the summary table states that the limit of 10 mrem/h will not be exceeded at any point 2 meters from the outer lateral surface of the vehicle. The staff finds that this meets the requirement in 10 CFR 71.47(b)(3).

The staff verified that the summary table states that the external radiation dose during HAC does not exceed 1 rem/h at 1 meter from the external surface of the package. The staff finds that this meets, in part, a requirement of 10 CFR 71.51(a)(2).

5.2 Source Specification

The staff compared the source specification for the contents with those listed in the General Information section of the application. The Model No. 10-160B has a wide range of contents including gamma and neutron sources as well as TRU waste. The staff finds that the contents are not well defined. Therefore the staff has limited the shipping contents based on the analysis.

5.2.1 Gamma Source

The applicant states that the gamma source is modeled in SCALE with a Co-60 source. Although the applicant has provided a new analysis method for the Gamma shielding, the staff did not review the changes in relation to the gamma source and shielding analysis because the applicant did not request that there be any additional material that exceeds the currently approved activity limits for gamma emitters.

5.2.2 Neutron Source

The applicant included a neutron source in this revision to be added to the allowable contents. The applicant specified a Pu-239 PuBe source with an emission rate of 1.1×10^8 n/sec.

The applicant provided the energy spectrum of the PuBe source. The applicant referenced the 1987 edition of Cember's "Introduction to Health Physics." The staff viewed the information referenced by the applicant (in Table 9.9 of Reference 5-3). Cember's table has a much more detailed neutron spectrum than that used by the applicant in Section 5.2.3 of the application. The applicant collapsed the spectra to match the cross section set used in the SCALE (SAS4) calculation.

The staff performed an MCNP calculation comparing Cember's more detailed energy spectra with the applicant's and found only a trivial difference in the calculated dose results and therefore finds the collapsing of the neutron energy spectra acceptable for use in the Model No. 10-160B shielding analyses.

5.3 Model Specification

The staff viewed Chapter 2, "Structural Evaluation", and Chapter 3, "Thermal Evaluation", of the application to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents. The staff viewed the information in Section 2.6 of the application and the applicant's conclusion that there will be no damage to the package as a result of NCT. This information was previously reviewed and approved by the staff and documented in Reference 5-1.

The staff also viewed the information in Section 2.7 of the application to determine the effects of HAC on the package. From the tests required in 10 CFR 71.73, any damage is limited to the impact limiter. This is summarized in Section 2.7.5 and 3.5.6. As stated in Section 5.3.1 on page 5-3, the applicant does not include the impact limiter in the shielding analyses and assumes nominal design dimensions.

The applicant has not accounted for the tolerances of the cask dimensions in their shielding analysis. However they state that the cask has a gamma scan as part of the acceptance criteria for the lead shield and allows no more than 10% reduction in shielding from nominal thickness. This is not considered in the shielding analysis however the staff finds that the dose rate measurements (which are slightly stricter than regulations) will ensure that the regulations are met. Future reviews of this package that reduce safety margins should consider this design tolerance.

The applicant does account for the lead slump resulting from the 30-foot drop. This is represented as a void 0.05-cm high at the top of the lead shell. This is consistent with the information presented in Section 2.7.1.1.3 (and previously reviewed and approved by the staff) of the application, states that the lead slump is predicted to be less than 0.02 inches (0.05cm). The staff finds that these models are consistent with the effects from the accident analyses described in Chapters 2 and 3.

5.3.1 Configuration of Source and Shielding

The staff examined the figures (5.1 and 5.2) in the application, the description of the modeling, and the SCALE (SAS4) input deck to determine how the shielding is modeled. The staff verified that the cask dimensions are consistent with the license drawings, C-110-D-29003-010, Rev. 14. The staff notes that the stainless steel liner is neglected and that this is conservative.

The cask is modeled as concentric cylinders with an inner carbon steel shield, surrounded in the radial direction by a lead shield and then surrounded by an outer carbon steel shield. The applicant models the neutron source as a point source.

The staff does not find current source modeling applicable for TRU waste, however the staff finds that the TRU waste is generally much lower in activity than sources and limited by the current approved contents for gamma emitters. In addition, dose rate measurements will be performed and are limited to more restrictive values than 10 CFR 71.47, ensuring that regulations will be met.

For NCT the source is in the middle of the cask surrounded by air. For HAC the source is placed at the corner of the top lid next to the area where the lead slump occurs.

This is an exclusive use package; therefore the applicant defines the transport vehicle as being an 8' wide trailer.

The staff verified that the applicant has a dose point for the following locations:

- External surface of the package (axial and radial);
- 2m from the surface of the transport vehicle.

The applicant did not have a dose point on the outer surface of the vehicle. Since this limit was met at the external surface of the packaging, the staff finds that it would meet the same limit at the outer surface of the vehicle as well. The staff finds that the applicant has evaluated the appropriate dose points per 10 CFR 71.47(b)(1),(2) and (3).

5.3.2 Material Properties

The staff verified that the applicant identified the materials and mass densities of the shield. The applicant identifies the inner and outer shield as "steel" in Table 5.2 of the application with a density of 7.82 g/cm^3, and the lead shield as "lead" with a density of 11.34 g/cm^3. Staff review of the SAS4 input decks in Section 5.7 confirm that the inner and outer shield is input as the SCALE default of carbon steel with the default density, 7.82 g/cm^3. Lead is represented as elemental lead using the default isotopic concentration and the default density, 11.34 g/cm^3. The staff finds the material specification of the shield acceptable.

The staff does not find that any of the shielding materials would degrade during the service life of the packaging and that there are no temperature sensitive materials present.
Since the applicant assumes a point source they do not make any assumptions regarding self-shielding, the staff finds that this is conservative.

5.4 Evaluation

5.4.1 Methods

The applicant uses the SAS4 code in the SCALE package. The applicant's references (5.6.1) indicate that they are using SCALE version 4.4. The SAS sequence codes in the SCALE package are widely used in packaging applications and are generally acceptable to the staff for shielding evaluations (Reference 5-2). The SAS4 code performs three-dimensional Monte Carlo shielding analyses of a nuclear fuel transport or storage container using an automated biasing procedure (Reference 5-3).

The applicant is using the 27n-18g coupled 27 neutron group, 18 gamma-ray group library for the shielding calculations. The neutron data were taken from the 27-group ENDF/B-IV library, and the gamma-ray data were taken from the standard SCALE gamma library (Reference 5-3). The staff finds this is an acceptable cross section library for use in the Model No. 10-160B as it is a standard library commonly used for these types of shielding analyses.

5.4.2 Key Input and Output Data

The staff performed a review of the SAS4 input deck and ensured that the geometry and materials were appropriately specified. In addition the staff reviewed the options selected and detector specification. The staff viewed the output files provided and determined that the results were properly represented in the application.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant performed the flux-to-dose-rate conversion using the SAS4 code. The staff confirmed that the applicant used the ANSI/ANS 6.1.1-1977 standard and finds this acceptable.

5.4.4 Radiation Levels

The staff confirmed that the applicant's calculated radiation levels under both NCT and HAC are in agreement with the summary tables and that they satisfy the limits in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2). The staff verified that the analysis showed that the locations selected are those of maximum radiation levels and include the radiation streaming path due to lead slump.

The staff also verified that the applicant's evaluation addresses damage to the shielding under NCT and HAC, as discussed in Section 5.3 of this SER.

5.4.5 Confirmatory Analysis

The staff performed an independent analysis of the Model No. 10-160B to confirm the applicant's results for the neutron dose rates. The staff did not perform calculations to confirm the gamma dose rates as the requested levels were previously found acceptable by the staff (Reference 5-4). The staff used the MCNP5 code with cross sections from the ENDF-VI library with the exception of lead. The staff used cross sections from the ENDF-V library for this element. The staff modeled NCT and HAC.

For NCT, the staff modeled a neutron point source at the center of the cask. The staff calculated the dose along the outside of the side and the top/bottom of the cask using a surface flux (F2) tally in MCNP. The staff used the same neutron energy spectrum and source rate as cited in Section 5.2.3 of the application. The source rate is 1.1×10^8 n/sec. The staff then used the flux-to-dose-rate conversion factors from the ANSI/ANS 6.1.1-1977 standard.

For HAC, the staff used the same model as the NCT but instead the neutron point source is at the corner of the cask next to the point where the cask experiences lead slump. The staff's HAC model assumes a gap of 0.05-cm to account for lead slump. The staff then used the flux-to-dose-rate conversion factors from the ANSI/ANS 6.1.1-1977 standard.

Since the source specification for the HAC model is asymmetric in relation to the package, the staff performed calculations using both the surface flux tally (F2) and the point detector tally (F5) in MCNP. The point detector tally gives much higher results as its dose point is located adjacent to the source and streaming path. In either case, the staff's calculations show that the applicant meets the allowable limits for HAC. The results of the staff's evaluation are summarized in Table 5-1.

Table 5-1: Results of Staff Neutron Dose Calculations

Case	Staff Dose Calculation (mrem/h)	Applicant Dose Calculation (mrem/h)
NCT, Package Surface – Side	107	114
NCT, Package Surface – Top/Bottom	83.1	86.3
HAC, 1 m from Surface – Side (surface flux tally)	16.1	82.7
HAC, 1 m from Surface – Top/Bottom (surface flux tally)	33.2	39.5
HAC, 1 m from Surface – Side (point detector tally)	133	82.7
HAC, 1 m from Surface – Top/Bottom (point detector tally)	111	39.5

Table 5-1 shows that the staff calculations also predict that the Model No. 10-160B is within regulatory limits of 200 mrem/hr at the package surface during NCT and 1 rem/hr at 1 m from the package surface during HAC.

5.5 Evaluation Findings

The staff's evaluation of the Model No. 10-160B results in the following evaluation findings for the addition of a Pu-Be neutron source specified with an emission rate of at 1.1×10^8 n/sec:

- As documented in Section 5.1 of this SER, the staff finds that the package description and evaluation satisfies the shielding requirements of 10 CFR Part 71;

- As documented in Section 5.2 of this SER, the staff finds that the source specification used in the shielding evaluation is sufficient to provide a basis for evaluation of the package against the shielding requirements of 10 CFR Part 71;

- As documented in Section 5.3 of this SER, the staff finds that the models used in the shielding evaluation are described in sufficient detail to permit an independent review and independent calculations of the package shielding design;

- As documented in Section 5.4 of this SER, the staff finds that the external radiation levels satisfy the requirements of 10 CFR 71.47 for packages transported by an exclusive-use vehicle;

- As documented in Section 5.3 of this SER, the staff finds that the radiation levels will not significantly increase during NCT consistent with the tests specified in 10 CFR 71.71;

- As documented in Section 5.3 of this SER, the staff finds that the maximum external radiation level at one meter from the external surface of the package will not exceed 1 rem/hr during HAC consistent with the tests specified in 10 CFR 71.73.

5.6 References

5-1 U.S. Nuclear Regulatory Commission, *Safety Evaluation Report, Model No. CNS 10-160B Package, Certificate of Compliance No. 9204,* Revision No. 0, November 2, 1990, ADAMS Accession No. ML030690362

5-2 J. S. Tang, M. B. Emmett, ORNL/TM-2005/39, Version 5.1, Vol. 1 Book 3, Sect. S4, Nuclear Science and Technology Division, *SAS4: A Monte Carlo Cask Shielding Analysis Module Using an Automated Biasing Procedure,* November 2006

5-3 H. Cember , *Introduction to Health Physics,* 3rd Edition, Copyright 1996 by McGraw-Hill Companies

5-4 U.S. Nuclear Regulatory Commission, *Safety Evaluation Report, Model No. CNS 10-160B Package, Certificate of Compliance No. 9204, Revision 7,* August 10, 2001, ADAMS Accession No. ML012410466

6.0 CRITICALITY

Energy*Solutions* submitted an application requesting approval to include fissile material as part of the allowable contents in the Model No. 10-160B package. The applicant documented the criticality evaluation in Chapter 6 of the application.

6.1 Description of Criticality Design

The applicant provided a criticality evaluation of the Model No. 10-160B package. The application supports shipment of up to ten TRU waste drums per cask. The criticality safety evaluation defines generic but bounding contents to cover all possible constituents. The neutron multiplication factor (k-effective, or k-eff.) will be less than 0.95 during all normal and accident conditions. The applicant does not take credit for burn-up. Details of the staff's evaluation of the package design features and summary follows.

6.1.1 Packaging Design Features

The staff reviewed the General Information provided in Chapter 1 and other applicable portions of the application to determine if the applicant specified the information important for criticality safety.

The staff verified that the information presented in the text was consistent with the license drawings, C-110-D-29003-010, Revision 14.

Although the Model No. 10-160B will contain ten separate drums, the criticality analysis does not take credit for any geometry features of the drum packaging. In addition there are no fixed neutron absorbers being credited in the analysis.

The applicant does not take credit for the presence or geometry of the drums. Some analysis assumptions (total amount of moderator (CH_2) and reflector (Be)) are based on the volume of the drums. The applicant identified that the drums are nominally 55 gallons and that design tolerance were not accounted for when calculating the amount of moderator and reflector to include in the analyses. Since the analyses are conservative, considerations of design tolerances would not negatively affect any safety conclusions resulting from the criticality analyses. The staff agrees with this statement and finds that it is acceptable to neglect design tolerances of the drums. However the staff acknowledges that this may need to be considered in the future should there be revisions to the contents and modeling of this package that may reduce safety margins.

With respect to the criticality evaluation, the staff finds that the description of the packaging is in sufficient detail to provide an adequate basis for its evaluation, and that the description includes types and dimensions of materials of construction and materials specifically used as non-fissile neutron absorbers or moderators. Therefore the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5).

Sections 6.5 (Page 6-27) and 6.6 (Page 6-29) of the application state that analyses were performed using an infinite number of packages. Therefore there is no limit (with respect to criticality) to the number of packages that can be transported in a single shipment. The staff finds that by specifying the allowable number of packages that may be transported in a single shipment that the applicant meets the requirements of 10 CFR 71.35(b).

6.1.2 Codes and Standards

The applicant identified the regulations in 10 CFR Part 71 that are applicable to the criticality design of the package. The staff finds that these regulations are appropriately identified. In addition the applicant states that any user of the package must have a quality assurance program that meets 10 CFR Part 71, Subpart H requirements. The staff finds that this meets the requirements of 10 CFR 71.31(c), with respect to criticality design, and finds it acceptable.

6.1.3 Summary Table of Criticality Evaluations

The applicant provided a summary table of the criticality evaluations in Table 1 on page 6-2 of the application. The staff reviewed the summary table of criticality evaluations and finds that the applicant performed criticality evaluations for a single package that is in the damaged condition, optimally moderated with CH_2 and water and the most reactive form of the fissile material and had close full reflection[1] of the containment system on all sides using 30-cm of water. The applicant's table does not show the results for a single package during NCT in the summary table. Even though the applicant does not show the results for NCT in the summary table, the staff notes that the applicant performs this analysis. This is discussed in Section 6.4.2 of this SER. The staff also notes that the HAC results bound that of NCT.

In the summary table, the applicant shows that the maximum k-eff for the limiting configuration is less than the upper subcriticality limit (USL) of 0.94. The staff finds that this meets the requirements of 10 CFR 71.55(b), (d) and (e).

The staff reviewed the summary table of criticality evaluations and finds that the applicant performed criticality evaluations for an infinite array of packages in the undamaged and damaged conditions. The applicant shows that under normal and accident conditions that the infinite array of packages has a k-eff less than the USL of 0.94. The staff finds that this meets the requirements of 10 CFR 71.59(a)(1) and (2).

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[1] "Close reflection by water" is defined in 10 CFR 71.4 as immediate contact by water of sufficient thickness for maximum reflection of neutrons.

The staff verified that the table includes the maximum value of k-eff. The applicant included the standard deviation and states that the USL is 0.94, to account for bias and uncertainties. The summary table shows that the package meets the subcriticality criterion.

The applicant shows that the limiting conditions for the Model No. 10-160B is the infinite array under hypothetical accident conditions with no interstitial moderation and full internal moderation. This analysis gives a maximum k-eff of 0.93873 with a standard deviation of 0.00020.

6.1.4 Criticality Safety Index

In Section 6.1.3 of the application, Energy*Solutions* calculated the Criticality Safety Index (CSI) to have a value of 0. This is based on the analysis performed that uses a value of N to be infinity (i.e., an infinite array of casks). The staff finds that the CSI was appropriately determined per 10 CFR 71.59(b). The staff finds that the application meets 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Fissile Material Contents

The Model No. 10-160B will contain remotely handled transuranic-containing waste (RH-TRU) as described in Appendix 4.10.2 of the application. The TRU waste will be limited to 325 grams of Pu-239, or the FGE, and must not be machine compacted and may have no more than 1% by weight of special reflectors and no more than 25% by volume hydrogenous material. The criticality safety analysis is based on 325 g of pure Pu-239.

The applicant states in Section 6.2 (page 6-2) that "the quantities of all fissile isotopes other than Pu-239 present in the RH-TRU waste matrix may be converted to a FGE as determined using the conversion factors outlined in the *Remote-Handling Transuranic Waste Authorization Methods for Payload Control."* The conversion factors are in Table 5.1-1 of Reference 6-4. Section 3.1.2 explains that these factors came from a ratio of subcritical mass limits established by ANSI/ANS-8.1 (Reference 6-76-7) and ANSI/ANS-8.15 (Reference 6-8). The concept of FGE was originally approved by the staff for the TRUPACT-II (Reference 6-1). The FGE factors used for the Model No. 10-160B were previously approved for use in the RH-TRU 72B (Reference 6-5). The applicant summarized the applicable conversion factors in Table 9.1.3, in Chapter 4, Appendix 4.10.2 of the application.

The staff notes that the contents of the Model No. 10-160B are similar to Case A of the RH-TRU 72B and therefore the staff finds the FGE conversion factors applicable to the Model No. 10-160B.

The staff finds that the applicant has described the contents in sufficient detail to provide an adequate basis for its evaluation. The staff finds that the applicant has defined adequately the type, maximum quantity, and chemical and physical form of the fissile material. The staff finds 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).

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The staff reviewed Chapter 2, “Structural Evaluation”, and Chapter 3, “Thermal Evaluation”, of the application to determine the effects of the NCT and HAC on the packaging and its contents. From the tests required in 10 CFR 71.73, any damage is limited to the impact limiter. This is summarized in Sections 2.7.5 and 3.5.6 of the application. As stated in Section 6.3.1 on page 6-4 of the application, the applicant does not include the impact limiter in the criticality analyses and assumes nominal design dimensions. The staff finds this acceptable.

The staff examined the sketches of the model used for the criticality calculations and verified that the dimensions and materials are consistent with those in the drawings of the actual package. The applicant discusses the differences in Section 6.3.1 of the application (page 6-4). The criticality model does not include drain ports, lifting holes, leak ports, the thermal barrier and impact limiter. The staff finds this acceptable and does not find the differences will impact the criticality safety evaluation.

The staff verified that the application considers deviations from nominal design configurations by analyzing a conservative condition.

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 5 in Chapter 5 of the application. There are no materials in the casks that need to be adjusted to be consistent with accident conditions, i.e., there are no materials used in the model that change form, such as a neutron shield or neutron absorbers that could melt, that are important to maintaining criticality calculations.

The applicant does not use any materials as neutron absorbers and the applicant does not request credit for burnable poisons in the fuel. The staff did not identify any criticality properties that could degrade during the service life of the package.

6.3.3 Computer Codes and Cross Section Libraries

The applicant uses the Monte Carlo N-Particle Code, version 5, Release 1.40 (MCNP5). Per the guidance in the standard review plan, NUREG 1609, the staff finds the MCNP code appropriate for the criticality evaluations.

The applicant uses a nuclear library based on the ENDF/B-VI nuclear data set. The staff recognizes this as a standard in criticality safety and finds its use acceptable for the criticality analyses for the Model No. 10-160B package.

The staff verified that the applicant provided other key input data for the criticality calculations. These are summarized below in Table 6-1 below.

Table 6-1: Criticality Key Input Data

Parameter	Value
Number of Neutrons per Generation	4000
Number of Generations	3950 (active)
Number of Cycles Skipped Before Accumulating Data	50

The staff verified that the applicant provided representative input files. The staff also verified that the information regarding the model configuration, material properties and cross sections is properly represented in the input files. The applicant provided a representative output file for both the limiting single package case and HAC array case. The staff viewed this file and determined that the multiplication factors from the output files agree with those reported in the evaluation. In addition the staff found that the calculation passed important statistical checks and has appropriate convergence behavior.

6.3.4 Demonstration of Maximum Reactivity

The staff reviewed the application to determine if each type of fissile material in the allowable contents was included. The applicant models the fissile contents as 325 g pure Pu-239 as this is bounding or equivalent for the other allowable fissile contents. The staff agrees with the applicant and finds this acceptable. The staff has found this representation acceptable for the RH-TRU 72B which has been approved to transport similar contents (Case A in Reference 6-5).

The applicant uses polyethylene (CH_2) as a bounding hydrogenous moderating material. The applicant references the RH-TRU 72-B Safety Analysis Report as a reference and a basis for this assumption and included it as an attachment to the application (Reference 6-6). This document points to a study done by SAIC that concludes that polyethylene is the most reactive moderating material for the TRU waste. This document describing the SAIC study was reviewed by the staff in support of the TRUPACT-II and HALFPACT packages (Reference 6-3). The staff finds the contents of the Model No. 10-160B to be similar to that of the RH-TRU 72B and the TRUPACT-II and therefore finds the previous review of this document, with respect to finding the most reactive moderating material, applicable and acceptable for the Model No. 10-160B.

The package will include “special reflecting” materials. The applicant defines “special reflecting” materials as materials that can provide better reflection than 25% polyethylene/75% water equivalent reflection and include Be, BeO, C, D2O, MgO, and depleted uranium (less than 0.72 wt% and greater than or equal to 0.3 wt% U-235). The applicant uses beryllium as a bounding reflecting material. This analysis includes beryllium in an amount of 1% by weight of the total mass of the CH_2 and Pu-239. Any other special reflectors that exceed 1% by weight the total mass of the CH_2 and Pu-239 are not authorized to be shipped in the Model No. 10-160B.

The applicant again cites RH-TRU 72B (Reference 6-6) as the basis for this assumption. The inclusion of these “special reflectors” was also a part of the SAIC study used to determine the most reactive moderating material and was also reviewed by the staff in support of the TRUPACT-II and HALFPACT packages (Reference 6-3). Again, since the staff finds the contents of the Model No. 10-160B to be similar to that of the RH-TRU 72B and the

TRUPACT-II, the previous review of the SAIC document, with respect to these reflector materials, is applicable and acceptable for the Model No. 10-160B.

The applicant's model assumes a homogeneous sphere of Pu-239, CH_2 and Be. The applicant performed analyses to determine the optimal H/Pu ratio by varying the CH_2 mass in the fissile sphere. The Be mass was adjusted to be 1% by mass of the CH_2 mass.

The applicant assumed a 25% packing fraction for the CH_2. This assumption is also used in the RH-TRU 72-B (Reference 6-6) which was approved by the staff (Reference 6-5). This application states that a test was performed to determine that this is conservative for manually compacted TRU waste. The staff has reviewed the document describing this test (Reference 6-9) and has determined that the 25% CH_2 packing fraction is appropriate for use in the Model No. 10-160B.

The empty space within the sphere that results from the packing fraction assumption is filled with void for NCT or water for HAC. The applicant adjusted the H/X ratio to ensure that the most reactive H/X was found that accounted for the water in the sphere.

For NCT, the remaining CH_2 and Be that are not used in the fissile sphere comprise a reflector completely surrounding the fissile sphere. The remaining cask volume is filled with air. For HAC cases the applicant homogenizes the remaining CH_2 and Be with water and fills the entire space inside the cask not occupied by the fissile sphere.

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

The applicant placed the casks in a hexagonal (triangular) array. The applicant performed sensitivity studies to determine the most reactive location of the sphere within the package and determined that having it on a flat side (versus a corner) was more conservative.

For array conditions the applicant did not address the occurrence where the sphere could be in a different location for each array element and in some configurations could cause the fissile material to be closer together and more reactive. The staff finds that the conservatisms in the analysis (fissile material in the shape of a sphere with optimal moderation conditions) compensate for this possible non-conservatism.

The applicant also varied the cask spacing and interspersed moderation, for HAC arrays in Chapter 6, Table 6.9, of the application. The applicant determined that the most conservative configuration is when the casks are placed closely together and with a low water density (0.001 g/cm^3) between packages. The staff finds that this meets the requirements of 10 CFR 71.59(a).

6.3.5 Confirmatory Analysis

The staff performed independent calculations to confirm the applicant's results and that the most reactive conditions had been correctly identified. The staff performed calculations with the CSAS26 criticality sequence of the SCALE 5.1 suite of codes. SCALE 5.1 was developed by Oak Ridge National Laboratory for use in criticality and shielding analyses. The CSAS26 sequence is a criticality sequence that uses KENO-VI geometry and multi-group cross sections. The staff used the 238-group cross section library derived from ENDF-VI data.

The staff constructed its model based on design information in the application and the referenced drawing. The staff modeled the HAC configuration for both the single package and the infinite array. The staff modeled the limiting conditions as identified by the applicant. These are for a 13.611761 cm radius sphere that has a homogeneous mixture of CH_2, Be and H_2O. The compositions are the same as the applicant identified in Table 6-3 (Table 3 in the criticality section) of the application for this mixture. The H/Pu ratio is 900. The staff assumed 30cm reflection on all sides in a hexagonal geometry around the package. The results of the staff's evaluation are summarized in Table 6-2.

Table 6-2: Independent Staff Calculations of the Model No. 10-160B

Case	Staff k-eff from SCALE	Standard Deviation	k-eff from application	Standard Deviation from application
Single package, HAC	0.9311	0.0021	0.93252	0.00020
Infinite Array, HAC	0.9339	0.0020	0.93873	0.00020

As shown in Table 6-2, the staff's evaluation generally agrees with that of the applicant.

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. As stated in Sections 6.3.1 of this SER, damage to the cask is limited to the impact liner and the applicant neglects the presence of the impact limiter in all calculations. The applicant floods the empty spaces inside the cask with water when performing calculations for the damaged condition.

The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents.

The staff determined that water moderation is in the most reactive extent as required by 10 CFR 71.55(b).

The applicant stated on page 6-8 of the application that all single package analyses include full reflection of 30.48 cm (12 in) water on all sides. The staff finds that this meets the requirement in 10 CFR 71.55(b)(3).

6.4.2 Results

6.4.2.1 NCT

For the single package evaluation during normal conditions of transport, the applicant found the most reactive configuration to be with the fissile sphere in the base corner with water on the outside of the cask and an H/Pu ratio of 1400. The maximum k-eff for these conditions is 0.42656.

Since k-eff is less than the USL of 0.94 under 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 uses a bounding geometry to perform the criticality calculations, the staff verified 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 finds that the applicant meets 10 CFR 71.55(d)(2).

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 10cm cube. The staff finds that the applicant meets the requirements in 10 CFR 71.55(d)(4).

6.4.2.2 HAC

For HAC the applicant found the most reactive configuration to be with the fissile sphere in the lid corner with an H/Pu ratio of 900. The maximum k-eff is 0.93252.

The applicant did not perform the immersion in water test required by 10 CFR 71.73(c)(5) because they cited that “water inleakage has been assumed for the criticality analysis.” The staff finds this acceptable.

Since k-eff is less than the USL of 0.94 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 under HAC 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 0.0 therefore they performed calculations using an infinite array of packages for the normal conditions of transport.

The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents. This is discussed in Sections 6.1.1, 6.2, and 6.3.4 of this SER. The applicant did not model full water reflection on all sides of the array because it is an infinite array.

6.5.2 Results

6.5.2.1 NCT

For the evaluation of an array of packages during NCT, the applicant used the same configuration of fissile material as in NCT. The fissile sphere is in the base corner with an H/Pu ratio of 1400. For array considerations the applicant placed the sphere on the flat side of the hexagon surrounding each cask and used as array elements. The applicant also performed sensitivity studies (Table 6-8 of the application) to demonstrate that k-eff values decrease with cask spacing. Therefore the applicant assumed no space between the casks in the array. In addition the applicant assumed no interstitial moderation for NCT. The maximum k-eff for these conditions is 0.45328.

Since k-eff for an infinite array is less than the USL of 0.94 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 HAC

For the evaluation of an array of packages during HAC, the applicant used worst case single cask model under HAC and placed it in an infinite array that is the same as that specified for NCT. The difference between the two arrays is that for HAC the applicant added interspersed moderation between the packages. The applicant assumed a water density of 0.001 g/cm^3 between the packages and demonstrated that it is conservative in relation to other higher density values. The maximum k-eff for these conditions is 0.93878.

Since k-eff for an infinite array is less than the USL of 0.94 under the tests specified in 10 CFR 71.73, 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 HAC be subcritical.

6.6 Benchmark Evaluations

The applicant is using the MCNP5 code with the cross section data from the ENDF/B-VI library. This code is widely used in industry application for criticality calculations and has therefore been extensively benchmarked against critical experiments.

6.6.1 Experiments and Applicability

The applicant performed benchmark calculations using MCNP5 against 40 experiments documented in NEA/NSC/DOC (95)03, “International Handbook of Evaluated Criticality Safety Benchmark Experiments.” The applicant used the MCNP input files from this reference but modified the cross section data to be consistent with that used in the Model No. 10-160B evaluation.

The applicant chose cases that matched the parameters in the Model No. 10-160B criticality evaluation as closely as possible. The applicant chose cases that have plutonium in solution form. The applicant chose cases with low Pu-240 content. The H/X ratio range includes the H/X ratio used in the Model No. 10-160B evaluation. The AEF (average energy causing fission) range also includes that analyzed for the Model No. 10-160B.

The experiments used by the applicant to benchmark their code are Plutonium-nitrate systems in the thermal range. The staff has previously accepted the use of Plutonium-nitrate benchmarking experiments for Pu-CH_2-Water systems (Reference 6-1). The applicant did not submit any benchmarking calculations that have Pu and Be in the thermal energy range. However the staff has accepted these configurations before without specific benchmarking calculations for a thermal Pu with Be system (Reference 6-3). Previous acceptance was based on the small amount of Be and the conservative nature of the analysis. The analysis discussed in Reference 6-3 is similar to the Model No. 10-160B and therefore the staff finds that this justification is also applicable to the Model No. 10-160B. The analysis performed by the applicant contains many conservative assumptions such as the spherical shape of the fissile material, optimum mixture of moderating and reflecting materials and the exclusion of Pu-240. The staff finds that the benchmark experiments are therefore acceptable for the Model No. 10-160B.

6.6.2 Bias Determination

The applicant used 40 experiments to determine a criticality code bias. The staff finds that this is a sufficient number of appropriate benchmark experiments. The applicant states that there were no observable trends for the benchmark k_{eff} values that would impact the bias determination.

Most all of the benchmark comparisons yielded positive bias results. The staff found that the applicant appropriately treated the bias as zero.

The applicant also accounted for statistical and convergence uncertainties of both benchmark and package calculations and experimental uncertainties when determining the USL. The method for combining uncertainties to determine the USL is consistent with the guidance in NUREG/CR-5661 (Reference 6-2) and therefore the staff finds this approach acceptable.

6.7 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.

6.8 References

6-1 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, TRUPACT-II Shipping Container, Certificate of Compliance No. 9218, Revision 0, August 30, 1989 (ADAMS Accession No. ML030800039)

6-2 Oak Ridge National Laboratory, NUREG/CR-5661, ORNL/TM-11936, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages,* April 1997 (ADAMS Accession No. ML073461008)

6-3 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, Docket No. 71-9218, Model No. TRUPACT-II Package, Certificate of Compliance No. 9218, Revision 17, August 23, 2004 (ADAMS Accession No. ML042370439)

6-4 US DOE, RH-TRAMPAC Document, Rev. 0, June 2006

6-5 Letter from C. M. Reagan (US NRC) to P.C. Gregory (Washington TRU Solutions), Safety Evaluation Report, *Revision 4 of the RH-TRU 72B Shipping Package (Docket No. 95-19-9518)*, July 28, 2006 (ADAMS Accession No. ML062130059)

6-6 RH-TRU 72B, Safety Analysis Report, Revision 4, June 2006

6-7 American Nuclear Society, ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors,* Approved September 9, 1998, Reaffirmed May 16, 2007

6-8 American Nuclear Society, ANSI/ANS-8.15-1981, *Nuclear Criticality Control of Special Actinide Elements,* Approved November 9, 1981, Reaffirmed July 15, 2005

6-9 Washington TRU Solutions, LLC., *Test Plan to determine the TRU Waste Polyethylene Packing Fraction,* June 19, 2003 (ADAMS Accession No. ML090120825)

7.0 PACKAGE OPERATIONS

The staff reviewed the instructions for package operation in Chapter 7 of the application. The procedures include preparation of the Model No. 10-160B for loading, including visual inspection of the packaging components prior to loading. Procedures for leak testing the package are specified. Additionally, Chapter 7 includes procedures for unloading the package and preparing an empty package for transport. It is noted that the package will be prepared for transport and operated according to site-specific written procedures which will be consistent with the procedures in Chapter 7. The certificate has been conditioned to specify that the package must be prepared for shipment and operated in accordance with Chapter 7.

Based on review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff reviewed Chapter 8 of the application for the acceptance tests and maintenance program for the Model No. 10-160B. Section 8.1 provides the acceptance tests for the Model No. 10-160B. Section 8.2 describes the maintenance program. Maintenance includes periodic replacement of the seals, and fasteners. Periodic leakage rate tests are required.

The certificate has been conditioned to specify that the Model No. 10-160B package must meet the acceptance tests and be maintained in accordance with Chapter 8. Additionally, the certificate specifies seal replacement if inspection shows any defects or every 12 months, whichever occurs first, and that the containment system must be leak tested in accordance with Section 8.1.3 and Section 8.2.2, as appropriate.

Based on review of the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

The conditions specified in the Certificate of Compliance have been revised to include the new contents and incorporate changes to designate the certificate as "-96." These conditions are listed below:

Condition No. 5(a)(1) was revised to remove "CNS" from the package Model No. description.

Condition No. 5.(b)(1) has been revised to make clear that the prohibition on explosives, corrosives, and non-radioactive pyrophorics, and compressed gasses, and the limitations on pyrophoric radionuclides and potentially volatile organic compounds apply to all contents. Fissile TRU waste and Pu-239 as PuBe neutron sources have been added as contents types.

Condition No. 5(b)(2) has been revised to include the limitation on fissile contents to a maximum of the fissile gram equivalent of 325 grams Pu-239. Plutonium content exceeding 0.74 TBq must be in solid form. The PuBe neutron sources are limited to a maximum emission rate of 1.1E+8 n/sec. The decay heat limit for the package has been increased to 200 watts.

Condition No. 5(c) has been added to reflect the Criticality Safety Index for the added fissile contents. The Criticality Safety Index is 0.0.

Condition No. 10(d) has been added to specify that a helium leak test may not be employed when elastomeric silicone rubber O-ring seals are used on the packaging.

Condition No. 11(c)(1) has been revised to specifically include decay heat and hydrogen gas generation rate as waste characteristics which must be determined and limited in accordance with Appendix 4.10.2 of the application.

Condition No. 11(c)(4) was revised to reflect that "CNS" has been removed from the package Model No. description.

Condition No. 12 was added to specify that air transport is not authorized for fissile material.

Condition No. 14 was added to allow the package to be marked with the previous package identification number, USA/9204/B(U)-85, until October 31, 2010. This is to allow time to replace the packaging nameplate showing the revised package identification number, USA/9204/B(U)F-96. However, any package used to transport fissile material must be marked with the revised identification number.

Condition No. 15 authorizes use of the previous revision of the certificate until October 31, 2010.

As a consequence of the inclusion of the new Condition Nos. 12 and 14, the previous Condition No. 12 was renumbered to Condition No. 13, and the previous Condition Nos. 13 and 14 have been renumbered to Condition Nos. 15 and 16, respectively.

CONCLUSION

Based on the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. 10-160B package design has been adequately described and evaluated and that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9204, Revision No. 13,
on September 10, 2009.