Safety Evaluation Report for the

Consolidated Safety Analysis Report for Model 10-160B Type B RADWASTE Shipping Cask, Revision 3, January 2011

Docket Number: 11-30-9204

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Summary

On February 28, 2011, The U.S. Nuclear Regulatory Commission (NRC) issued Revision 15 of the NRC Certificate of Compliance (CoC) USA/9204/B(U)F-96, for the 10-160B packaging. Revision 15 of the NRC CoC had significant changes and NRC's Safety Evaluation Report (SER) addressed the rational for these changes.

Revision 0 of the Department of Energy (DOE) CoC USA/9204/B(U)F-96(DOE) was issued based on Revision 13 of the NRC CoC. Revision 13 of the NRC CoC was based on the *Consolidated Safety Analysis Report for Model 10-160B Type B RADWASTE Shipping Cask*, Energy*Solutions*, Revision 0, December 2007 and the four applications with minor changes to the Consolidated Safety Analysis Report (CSAR). The NRC CoC Revision 13 updated the 10-160B to the "-96" requirements and consolidated all of the previous applications to NRC for the 10-160B. The DOE CoC Revision 0, which was equivalent to the NRC CoC Revision 13, was issued on December 29, 2009. This revision was issued so that changes to 10-160B for DOE shipments would be made to the DOE CoC and not the NRC CoC.

The NRC issued Revision 14 to the NRC CoC on February 17, 2010. This revision was a renewal of the CoC, extended the expiration date by five years to October 31, 2015, and had only minor changes to text of the CoC (mainly the extension of the expiration date).

Revision 1 of the DOE CoC is based on *Consolidated Safety Analysis Report for Model 10-160B Type B RADWASTE Shipping Cask*, Energy*Solutions*, Revision 3, January 2011, which supersedes the previous revisions of the Safety Analysis Report. The NRC SER stated that the CSAR (Revision 3, January 2011) submitted under the January 24, 2011, application supersedes all previous revisions of the application; and the CoC only lists this application under references. The DOE Packaging Certification Program reviewed the changes in Revision 15 of the NRC CoC and the supporting NRC SER and concurs with the needs for these changes. Revision 1 of the DOE CoC 9204 is equivalent to Revision 15 of the NRC CoC and is being issued based on Revision 15.

The DOE SER for Revision 1 of the CoC is based on the NRC SER for Revision 15 of the NRC CoC and incorporates the NRC SER. A copy of the NRC SER is attached on the following pages.

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

SAFETY EVALUATION REPORT
Docket No. 71-9204
Model No. 10-160B
Certificate of Compliance No. 9204
Revision No. 15

SUMMARY

By application dated April 2, 2010, as supplemented on May 20, September 27, December 2, 2010, and January 24, 2011, Energy*Solutions* requested the addition of powdered solids to the list of approved contents of the Model No. 10-160B package. The consolidated application, Revision No. 3, dated January 24, 2011, submitted per 10 CFR 71.38(c), supersedes all previous revisions of the application.

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 to the Model No. 10-160B package design.

1.2.1 Contents

The powdered radioactive materials covered by this amendment request are radioactive powders that are encapsulated to form radioactive sources or items contaminated by liquid radioactive sources that have broken, spilled, or otherwise become uncontained followed by evaporation of the liquid carrier. Unless the source is capable of maintaining its integrity under Hypothetical Accident Conditions (HAC), i.e., as a "special form" source, the form must be considered a powder. Radioactive material is in various chemical forms including oxides, chlorides, sulfates, citrates, nitrates, and phosphates.

The applicant initially stated that the maximum activity of the contents was unchanged from the previously approved Certificate of Compliance (CoC), i.e., activity limited to the package category limit of $3000A_2$ while also being limited by the thermal limit of 200 watts and the package external dose rate limits of 10 CFR 71.47 and 10 CFR 71.51(a)(2). The staff disagreed with the interpretation of the applicant to (i) define contents in terms of A_2 and (ii) rely on measurements to demonstrate compliance with regulatory limits. Staff's rationale is explained in detail below:

Upon reviewing the history of this application, staff determined that defining contents in terms of A_2 was probably a carryover from legacy Low Specific Activity (LSA) waste packages under regulations prior to 1996. The International Atomic Energy Agency (IAEA) and the United

States Nuclear Regulatory Commission (NRC) have, since the publication of the 1996-edition of TS-R-1 and 10 CFR Part 71, stopped the use of A_2 as a unit for defining contents of transportation package. The staff believes that, for historical reasons, some legacy waste packages have not been updated to these revised regulations. Since the approval of LSA packages is under the purview of the Department of Transportation (DOT), NRC staff has not kept track of any LSA package updates. However, the review and approval of Type B package designs, such as the Model No. 10-160B package, must be done in compliance with the latest 10 CFR Part 71 regulations. This is also true for legacy LSA packages that are intended for use as Type B packages.

During its review of the current amendment request, staff also found that the quantity limits of the contents in the application would greatly exceed regulatory limits in 10 CFR 71.47 for normal conditions of transport (NCT). In addition, the shielding analysis provided in the application was insufficient in justifying these contents.

The applicant also stated that (i) it relies upon pre-shipment dose rate measurement as a means to demonstrate package design compliance with NCT, and (ii) calculations were only performed to demonstrate compliance with HAC. The staff's position is that it is not appropriate to demonstrate compliance with dose rate regulations in 10 CFR 71.47 purely based on measurements: 10 CFR 71.31 requires that the applicant submit a package evaluation, and 10 CFR 71.35 requires that this package evaluation include a demonstration that the package satisfies the limits set in 10 CFR 71.47.

Furthermore, the staff determined that it is not appropriate to define contents in terms of A_2 for the following reasons:

(1) Content with a given A_2 value can be any individual or combinations of a variety of radioactive isotopes. Using A_2 as unit to define the quantity limits of the content does not provide a unique description of the content;

(2) The A_2 value does not tell the nature of the of the source, i.e., neutron or gamma, nor the energy spectra of the content; and

(3) The A_2 value is determined by the weighted average of the A_2 value of individual isotopes in the content (Appendix A of 10 CFR Part 71). It is impossible to determine the A_2 value of the content without knowing the actual constituent nuclides in the content.

Thus, staff clarified that A_2 shall not be considered as an appropriate unit to define a source term and is intended solely for:

(1) Material selection purpose – the 3000 A_2 limit is directly linked to the structural robustness of the package, classified as a Category II package, and

(2) Control and verification of leak rates only, as stated in 10 CFR 71.51(a)(1) and (a)(2).

Staff requested the applicant to revise the application and define a maximum activity of the contents in gammas/second as a primary control method for the payload. The activity limit is now determined per a procedure referenced as Attachment 1 to Chapter No. 7 of the application. Neutron emitters are only permitted as previously approved.

A secondary container is now also required for all types of contents, including powdered solids.

Finally, staff requested a limitation, to prevent pyrophoric situations and support of the shipment of powders, as follows: powdered solid radioactive materials shall not include radioactive forms of combustible metal hydrides or elemental metals, i.e., magnesium, titanium, sodium, potassium, lithium, zirconium, hafnium, calcium, zinc, plutonium, uranium, and thorium, or non – metals, i.e., phosphorus. Such a limitation is included as a condition of the certificate.

2.0 STRUCTURAL EVALUATION

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

3.0 THERMAL EVALUATION

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

4.0 CONTAINMENT EVALUATION

4.1 Description of the Containment Design

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 package cavity surface and the lid surface to protect all accessible areas from contamination. The primary and secondary lids are sealed with two high-temperature elastomeric O-rings in the machine grooves. Both the vent port and the drain port are sealed with a seal and a cap screw.

4.2 Containment Evaluation

The applicant has proposed no changes to the containment design of the package but the addition of powdered solids to the allowable contents is conditioned on the demonstration that the package is leaktight in accordance with ANSI N14.5. Periodic leak test verifications are described in Section No. 4.9 of the application.

This test is conducted by evacuating the package's cavity to at least 90% vacuum, and then pressurizing the cavity with helium to +1 psig – 0 psig. The annulus between the O-rings is evacuated until the vacuum is sufficient to operate the helium mass spectrometer leak detector. The detector is used to monitor the helium concentration in the annulus, using a sensitivity equal to 5.0×10^{-8} atm-cm^3/sec or less. Calibration of the leak detector shall be performed using a leak rate standard traceable to NIST, according to Section No. 8.1.3 of the application. In order to demonstrate the leaktight condition, the acceptance criteria is 1.0×10^{-7} atm-cm^3/sec of air. Similar tests are performed on the vent and drain ports.

4.3 Evaluation Findings

Based on the review of the statements and representations 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 meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING

The Model No. 10-160B package shielding analysis was originally approved by staff and documented in a safety evaluation report (SER) dated November 2, 1990 (Reference 1). At this time, the maximum contents were identified as 2000 Type A quantity (A_2). Staff also reviewed and approved an increase in the contents from 2000 A_2 to 3000 A_2 and documented this approval in an SER dated August 10, 2001 (Reference 2). Despite multiple amendments and an increase in contents, the package shielding evaluation was not updated until a December 20, 2007, amendment request (Reference 3); however, the revised gamma shielding analysis had not been reviewed nor approved (Reference 4) as a part of that amendment request.

During its review of the current amendment, staff found that (i) the quantity limits of the contents in the package application, as submitted, would greatly exceed 10 CFR 71.47 regulatory limits for NCT, and (ii) the shielding analysis was insufficient in justifying these contents. The applicant stated that (i) it relied upon pre-shipment dose rate measurements to demonstrate the package design compliance for NCT conditions and (ii) calculations were only performed to demonstrate that compliance with hypothetical accident conditions (HAC).

The staff determined that it was not appropriate to demonstrate compliance with 10 CFR 71.47 purely based on measurements for several reasons, including (i) 10 CFR 71.31 requires the applicant to submit a package evaluation, and (ii) 10 CFR 71.35 requires a package evaluation including a demonstration of compliance with 10 CFR 71.47. Also, since the applicant relied upon measurements to meet regulatory dose rate limits, the current content limit of 3000 A_2 was obviously not intended to define maximum contents allowed in the package, but rather corresponded to a maximum limit for a structural categorization as described in Regulatory Guide 7.11.

Furthermore, the staff found that it was not appropriate to define contents in terms of A_2 for the reasons stated in Section No.1 of this SER, i.e., using A_2 as unit to define the quantity limits of the content does not provide a unique description of the content; the A_2 value does not tell the nature of the of the source, i.e. neutron or gamma, nor the energy spectra of the content; and it is impossible to determine the A_2 value of the content without knowing the actual constituent nuclides in the content.

The staff also found that the application, as submitted, did not include a safety analysis bounding all possible contents for the Model No. 10-160B package, and requested a more specific definition of, and a safety analysis for, the contents (Reference 5).

In the revised application, in response to staff's two requests for additional information, the applicant identified 13.4 Ci of Co-60 as the bounding source term for the gamma emitter contents and determined that the dose rate for this package is 9.96 mrem/hr at 2 meter from the surface of the transportation vehicle enclosure.

The following sections document the staff's evaluation of the revised application, and of its supplemental materials.

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The Model No. 10-160B package shielding design consists of a 1-1/8-inch thick carbon steel inner shell, 1-7/8 inches of lead, and a 2-inch thick carbon steel outer shell. The inner and outer shells are welded to a 5-1/2-inch thick carbon steel bottom plate. A 12-gage stainless steel thermal shield surrounds the package outer-shell in the region between the foam-filled impact limiters that cover the top and bottom of the vertically oriented package. The package lid is a 5-1/2-inch thick carbon steel plate with a 31-inch diameter opening equipped with a secondary lid. A sealed secondary container is used for powdered contents.

The applicant's structural analysis of the package under HAC conditions demonstrates that no loss of shielding occurs at the package's ends. Lead slump, however, may occur along the package wall, resulting in a loss of lead shielding at the end of the package lead layer. The applicant considered this lead slump in the shielding analysis.

5.1.2 Summary Table of Maximum Radiation Levels

The applicant performed a shielding analysis using the SAS4 sequence of the SCALE 5.1 computer code (Reference 6). In the SAS4 shielding model, the applicant used both a point source and distributed sources under NCT. The applicant calculated NCT dose rates with a 13.4 Ci Co-60 source, and an average gamma energy of 1.25 MeV, located in the center of the package cavity. The applicant also calculated HAC dose rates by assuming that lead slump occurs under a 30-foot drop accident and that the point source has been relocated to the inner liner adjacent to the location of the lead slump and in contact with the lid.

Table No. 5.1 of the application provides a summary of the shielding calculation results for both NCT and HAC conditions:

Summary of Dose Rates of the 10-160B package (mrem/hr)

	Package Surface		1 m from package Surface		2 m from 8-foot trailer
Condition	Side	Top/Bottom	Side	Top/Bottom	Side
NCT					
Neutron Source	114	83.3	N.A.	N.A.	9.44
Gamma Source	126	479	N.A.	N.A.	9.96
HAC					
Neutron Source	N.A.	N.A.	82.7	39.5	N.A.
Gamma Source	N.A.	N.A.	144	99.9	N.A.

5.2 Source Specification

Because of the wide variety of contents, the applicant cannot reasonably provide a description for the exact constituents of the isotopes, nor the energy spectra of all potential contents. For this reason, the applicant developed, during staff's review of the application and at staff's request, an approach that can be used by package users to determine the maximum allowable source term of the package contents.

The method developed to determine the maximum activity for a given energy spectra of the sources in the package contents is based on the assumption that dose rates at points outside the package are directly proportional to the activities and spectra of the sources in the package's contents. With this assumption, if the dose rates from a unit source are known for the points where regulatory limits are specified, the users will be able to determine the maximum allowable activity in the package contents by dividing the regulatory dose rate limits by the dose rates contributed from a unit source.

Thus, the applicant developed SAS4 models for the package with unit source strengths and varying source energies. The results of the shielding calculations of the SAS4 models are a set of data relating dose rates to unit source strengths and spectra. With such data, the user can determine the maximum source activity that will meet regulatory dose rate limits for a given source energy. As already mentioned above in this SER, a point source of Co-60 with an activity of 13.4 Ci will give a dose rate of 9.96 mrem/hr at two meters from the side surface of the package trailer enclosure. For contents with multiple energies or multiple radioactive isotopes, the users will calculate the fractional dose rate contribution from each energy bin of the source. Attachment 1 to Chapter No. 7 of the application provides details on this methodology and examples.

For distributed sources in the package, the dose rate versus source energy for unit activity is somewhat more complicated because of the content's self-shielding. In order to take credit for the self-shielding effect of the contents, the applicant developed models with various densities and source energies. Considering the fact that contents may have various materials, the applicant further studied the impact of the atomic number, i.e., the Z value, of the contents to the self-shielding because higher Z values will have higher gamma attenuation. Through a series of shielding calculations, the applicant concluded that the contents will have the lowest self-shielding effect when Z=40 (zirconium, Zr). The application does not provide further details on how the Z value was determined.

To justify the use of Zr in the above calculations, the applicant performed a series of calculations using different Z values (from 7 to 92) for the source media, while maintaining the density of the source constant at 1.46 g/cm^3. The density, i.e., 1.46 g/cm^3, corresponds to the density needed to fill the package cavity while maintaining the package content weight limit to 14,500 lbs.

The applicant performed separate calculations for gamma energies of 0.5, 0.7, 0.9, and 1.5 MeV. The dose rates of gamma radiation at each energy bin were then plotted against the Z of the material, with Z varying from 7 to 92. Plots of the dose rates at each gamma energy versus the Z value of the material showed that a Z=40 (zirconium) value is conservative, i.e., gives a larger dose rate for all energies except 0.5 MeV. Since gammas of 0.5 MeV and less do not contribute significantly to the package dose rates, the applicant determined that using a Z=40 for the source is appropriate.

The applicant also studied the impact of the content density to its self-shielding by varying the content density from 0.5 g/cc to 8 g/cc. Based on the results of this study, the applicant developed an empirical equation for the content density correction factor (DCF),

$$DCF = 0.7 \times \ln(\rho) + 0.98,$$

where ρ is the material density of the content.

The applicant then developed two curves for the maximum allowable source activity versus source energy, one for point sources and one for distributed sources. These two curves are shown in Figure No. 5.3 and in Attachment 1 to Chapter No. 7 of the application.

5.3 Model Specification

The applicant used nominal dimensions of the package and did not consider any manufacturing tolerances in the shielding analysis. Impact limiters and the package inner container are conservatively neglected for shielding evaluation purposes. However, the applicant states that the package has a gamma scan as part of the acceptance criteria for the lead shield and allows no more than a 10% reduction in shielding from its nominal thickness. Future amendment requests should consider the package design and manufacturing tolerances, if safety margins are reduced.

The applicant considered the lead slump resulting from the 30-foot drop, represented as a gap 0.05-cm high at the top of the lead shell. This is consistent with the information presented in Section No. 2.7.1.1.3 (and previously reviewed and approved by the staff) of the application that the lead slump is predicted to be less than 0.02 inches (0.05 cm). The staff finds these models acceptable because they are consistent with the results from the accident analyses described in Chapter Nos. 2 and 3 of the application.

5.3.1 Configuration of Source and Shielding

5.3.1.1 Normal Conditions of Transport

The walls of the Model No. 10-160B package (1.125-inch thick inner and 2-inch thick outer steel walls with a 1.875-inch thick lead layer between) were modeled as cylindrical shells around the cavity cylinder. The base and lid of the package is a 5.5-inch thick steel plate. As said above, the impact limiters were conservatively neglected. A sketch of the model is shown in Figure No. 5.1 of the application. Since the package lid and bottom are identical, only one end was modeled for shielding purposes.

The package is transported upright in an exclusive use conveyance. Shielding results show that, with a 13.4 Ci Cobalt-60 source, the gamma dose rate is 9.96 mrem/hr at 2 m from the enclosure surface of the trailer in compliance with 10 CFR 71.47(b)(2).

5.3.1.2 Hypothetical Accident Conditions of Transport

HAC conditions do not affect the geometry of the steel shells or of the base or lid. Only lead slump resulting from the 30-foot drop (< 0.02 inches), as discussed in Section No. 2.7.1.1 of the application, is included in the HAC model as a void 0.05-cm high at the top of the lead shell. Dose rates are determined at 1 m from the sidewall and the lid. The HAC model is shown in Figure No. 5.2 of the application.

5.3.2 Material Properties

Table No. 5.2 of the application provides a summary of the materials and their properties used in the model. The staff verified the materials and mass densities of the shield and finds that (i) they are consistent with material properties as commonly used and (ii) the shielding materials are not expected to degrade over the service life of the package under NCT. No temperature sensitive material is present in the packaging.

The staff reviewed the SAS4 input decks in Section No. 5.7 and confirmed that materials data used in the input are correctly selected.

5.4 Shielding Evaluation

The staff reviewed the information presented in the revised application and its supplemental documents. The applicant demonstrated, with the results shown in Table No. 5.1 of the application, that contents under NCT will meet HAC dose rate limits. As such, the maximum activity was determined by the applicant based on only NCT conditions.

5.4.1 Methods

The staff reviewed the methodology used by the applicant to determine the maximum activity of the contents to meet the dose rate limits of 10 CFR 71.47. As stated in Section No. 5.2 of this SER, the applicant determined the maximum activity by assuming that dose rates at points outside the package are directly proportional to the activities and spectra of the sources in the contents. Neglecting self-shielding, the staff finds this methodology acceptable since it will yield the most conservative results for a given source. Including self-shielding involves determining the material yielding the most conservative results when including a density correction factor to take credit for self-shielding. Staff finds that a low Z material would ordinarily be a plausible choice for performing self-shielding calculations, while the applicant concluded from its analyses that a material with a Z value of 40 would be most conservative.

In addition, the applicant assumed that the source and the materials are uniformly distributed in the cavity of the package. The staff considers this to be a very impractical assumption because, unless the contents are well mixed, the uniform distribution of the sources and density of the contents in the package are not guaranteed.

The staff does not agree with the results and conclusions made by the applicant on (i) the selection of Z=40 and (ii) the density correction factor calculated based on the selected Z value for the following reasons:

1. The density of the content may be not uniformly distributed in the package. Assuming an homogeneous density as a basis from which to calculate the density correction factor will definitely produce non-conservative dose rates because anything that is non-uniformly distributed will result in less self-shielding.

2. It is physically incorrect to search for the Z that produces the highest dose rate by varying the Z value, while keeping the density constant. The problem is that, when the applicant searched for the Z that produces the highest dose rate, it might have allowed the material density to go beyond the theoretical density of that material, which is physically impossible and incorrect. For example when Z=3 (lithium), the maximum possible material density is 0.534 g/cc. When Z=40, the maximum density is 6.4 g/cc. Therefore, for Z=40, using a density of 8.0 g/cc is erroneous since one can never have zirconium with a density exceeding the theoretical density.

For these reasons, the staff determined that comprehensive measurements of the package dose rates prior to shipment is necessary as a condition for using this package for generic contents. Also, unless further analysis can demonstrate the validity of the results, the density correction factor equation is, in general, not considered valid for general purposes. The current analyses are not considered sufficient for the purpose of validating the density correction factor equation as presented in Section No. 5.5 of the application.

In addition, the staff found that uncertainties of the SAS4 code were not considered in the shielding analyses. Because the SAS4 code was initially developed for spent fuel transportation shielding analysis and the MORSE model employed in the code may produce results with significant uncertainty, using SAS4 for general-purpose shielding analysis should also include an appropriate uncertainty analysis via code benchmarking.

5.4.2 Key Input and Output Data

The staff performed a review of the SAS4 input deck to ensure that the geometry and materials were appropriately specified. The staff reviewed the output files provided and determined that the results were properly represented in the application.

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 reviewed the model input and determined that the selected detector locations are appropriate and can detect possible radiation streaming path due to lead slump.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant performed the flux-to-dose-rate conversion using the factors provided in the SAS4 code. The staff confirmed that the applicant used the ANSI/ANS 6.1.1-1977 standard and found this acceptable because it is consistent with NUREG-1609.

5.4.5 Confirmatory Analysis

The staff performed confirmatory analyses with a 13.4 Ci of Co-60 point source in the package. The staff used MCNP 5.1 (Reference 7) with the continuous energy library. The results of the staff's confirmatory analyses show reasonable agreement with the applicant's shielding analysis for the limiting point source case.

5.5 Evaluation Findings

Based on its review and the results of staff's confirmatory analyses, the staff determined that the design of the Model No. 10-160B package, with contents of 13.4 Ci of Co-60 or equivalent provides a reasonable assurance to meet 10 CFR Part 71 regulatory requirements. Staff determined that, although the methodology for determining the material density correction factor developed by the applicant is not accurate, the comprehensive dose rate measurements, added as a condition of the certificate, will provide reasonable assurance for detecting possible hot spots of radiation.

5.6 References

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.

2. 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.

3. Letter from P.L. Paquin (Energy*Solutions*) to R. A. Nelson (NRC/SFST), Amendment Request for the 10-160B, Certificate No. 9204, December 20, 2007, ADAMS Accession No. ML090540866.

4. Certificate of Compliance No. 9204 for the Model No. 10-160B Package, Revision 13, September 10, 2009, ADAMS Accession No. ML092540028.

5. NRC NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," May 1999.

6. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 6, Vols. I-III, January 2009.

7. MCNP – A General Monte Carlo N-Particle Transport Code Version 5, X-3 Monte Carlo Codes Applied Physics Division, Los Alamos National Laboratory, April 24, 2003 (Revised 2/1/2008).

6.0 CRITICALITY EVALUATION

The applicant has proposed no changes to the authorized fissile contents for the Model No. 10-160B package.

7.0 PACKAGE OPERATIONS

The staff reviewed the instructions for package operations which include preparation for loading, including visual inspection of the packaging components prior to loading, procedures for leak testing, procedures for unloading the package and preparing an empty package for transport. The package will be prepared for transport and operated according to site-specific written procedures which will be consistent with the procedures in Chapter No. 7 of the application, except those mentioned in Section No. 7.4.

The applicant specifies that hydrogen and oxygen, produced by the radiolysis decomposition of a small amount of water remaining after draining the package cavity, and the hydrogen concentration must be limited to less than 5% (by volume) in the package. Staff did not agree with the nitrogen inerting operations, as shown and planned in Section No. 7.4 of the application, because:

- The described operations do not clearly show that the process will prevent the development of flammable gas mixtures in any confined area of the package throughout the entire transport period.
- The applicant did not provide a detailed evaluation analysis to demonstrate that there are no flammable gas mixtures (considering the worst case concentrations of hydrogen or any other flammable gases, and oxygen) during shipment.
- The applicant did not demonstrate that the inerting process effectively prevents the creation of flammable mixtures during the time of the shipment.

- The applicant did not provide a detailed configuration of the secondary container to ensure that the nitrogen could be introduced effectively to the innermost packaging or other confined areas within the containment system of the Model No. 10-160B package.

The applicant also states that "If a leak path can develop between the secondary container and the cask, the cask will also be inerted." It is not clear how the leak path between the secondary container and the package is defined and the applicant did not show that both the secondary container and the package cavity can be properly inerted prior to shipment. Thus, one of the conditions of the certificate is that inerting is not allowed for contents other than TRU waste.

The applicant added an attachment to Chapter No. 7 to allow users to determine the maximum allowable source term of the package contents, based on the known source strength, source spectrum, and the specific weight of the contents. The methodology is based on the assumption that dose rates are directly proportional to the activities and spectra of the sources in the package contents. As such, if dose rates from a unit source are known for those points where regulatory limits are defined, users will be able to determine the maximum allowable activity in the package contents by dividing regulatory dose rate limits by the dose rates contributed from a unit source. For contents with multiple energies or multiple radioactive isotopes, users will calculate the fractional dose rate contribution from each energy bin of the source. The staff reviewed the calculations and derivation of the curves and equation and found that the curves and equations were derived based on shielding analyses that did not include considerations of uncertainties in of the SAS4 computer code and the manufacture tolerances of the cask in the computer models. As such, it is assumed that significant bias and errors might have been built into the curves and equations that are to be used by the users.

More importantly, the shielding analyses for the non-point sources were performed with assumption that the content and source are homogenously distributed in the package cavity. Considering the intended uses of this package, the contents of this package for non-point source are likely to be inhomogeneous for most of the cases. To assure the safety of the general public and the shippers and the handlers, a dose rate measurement requirement, condition No. 10, is added to the Certificate of Compliance as a condition for using this package. These measurement requirements are excerpted from the draft ANSI 14.36 standard, still under review by the voting committee. However, staff considers them appropriate for use and adequate for this application at this time. Upon publication of the standard, this CoC should be revised to reference directly the ANSI 14.36 standard.

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, particularly of 10 CFR 71.41 and 51, 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 package. Section No. 8.1 provides the acceptance tests while Section No. 8.2 describes the maintenance program, including 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 No. 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 Nos. 8.1.3 and 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 several changes as indicated below:

Item No. 3.b has been revised to identify the consolidated EnergySolutions application dated January 24, 2011.

Condition No. 5.(b)(1)(v) has been added to include powdered solids in secondary containers as contents types.

Condition No. 5(b)(2)(i) has been added to include a new definition of the maximum quantity of material per package. Contents are limited to 13.4 Ci of Co-60. For contents other than Co-60, the maximum quantity of radioactive material is determined by the methodology described in Attachment 1 to Chapter No. 7 of the application.

Condition Nos. 5(b)(2)(ii), 5(b)(2)(iii), 5(b)(2)(iv), 5(b)(2)(v), 5(b)(2)(vi), 5(b)(2)(vii), and 5(b)(2)(viii) have been added to clearly spell out several conditions stated under Condition No. 5(b)(2) of the previous certificate.

Condition No. 5(b)(2)(ix) has been added to limit the powdered materials that can be shipped. Combustible metal hydrides or combustible elemental metals, i.e., magnesium, titanium, sodium, potassium, lithium, zirconium, hafnium, calcium, zinc, plutonium, uranium and thorium, or combustible non-metals such as phosphorus cannot be shipped in this package.

Condition No. 5(b)(2)(x) has been added to prohibit powdered solids with neutron emitters.

Condition No. 6, known as condition No.9 in the previous certificate, has been modified to prevent inerting as called in Section No. 7.4 of the application.

Condition No. 7 was numbered Condition No. 12 in the previous certificate.

Condition No. 8 specifies that, for contents other than TRU waste, inerting is not allowed to limit the concentration of flammable gases.

Condition No. 9 replaces condition No. 11(c) of the previous certificate. TRU waste characteristics must be determined and limited in accordance with Appendix 4.10.2 of the application.

Condition No. 10 has been added to include comprehensive dose rate measurements prior to shipment to detect hot spots caused by the non-uniform distributions of the sources and the density of the contents.

Condition No. 11 requires devices or measures to secure the contents inside the secondary container, if necessary.

Condition No. 13 authorizes use of the previous revision of the certificate until February 28, 2012.

The expiration date of the certificate was not modified. Condition Nos. 6, 7, 8, 9(c), 10, 11, and 14 of the previous certificate were either rewritten or deleted as either no longer applicable or because they referenced Chapter Nos. 7 and 8 of the application.

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. 15,
on February 28, 2011.