Safety Evaluation Report for Revision of DOE CoC 9204 based on NRC C0C 9204 Revision 20

Docket Number: 12-29-9204

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Background

Energy*Solutions* (ES) was the certificate holder for the Department of Energy (DOE) CoC 9204 Revision 3. Revision 3 of DOE CoC 9204 was equivalent to and based on Revision 16 of the Nuclear Regulatory Commission (NRC) CoC 9204, except for approval of one additional content for shipment by DOE. ES is also the certificate holder for the NRC CoC 9204 for the 10-160B and the NRC CoC 9168 for the CNS 8-120B package.

ES informed NRC and DOE that it had voluntarily removed all of the 10-160B packages (NRC CoC 9204 and DOE CoC 9204) from service effective April 27, 2012.

ES requested a revision to NRC CoC 9204 for the 10-160B package under the provisions of 10 CFR 71.41(c) by letter dated May 15, 2012, as supplemented on May 17, 2012. The regulations in 10 CFR 71.41(c) states that the NRC may authorize a package using environmental and test conditions different from those specified in either 10 CFR 71.71, *Normal Conditions of Transport*, or 10 CFR 71.73, *Hypothetical Accident Conditions*, if the controls proposed by the shipper are demonstrated to be adequate to provide an equivalent level of safety.

NRC issued Revision 19 to CoC 9204 and its SER with a condition to allow for shipments to be made for 90 days after the issue date of May 25, 2012 (expiring on August 23, 2012). The DOE Packaging Certification Program (PCP) reviewed the ES submittals of March 15 and March 17, 2012, and the NRC SER for Revision 19 of the NRC CoC and concurred with the finding and actions of NRC. The DOE SER for Revision 4 of the CoC was based on Revision 19 of the NRC CoC, and incorporated the NRC SER. The DOE CoC Revision 4 was issued on June 6, 2012, with equivalent conditions to those listed in Revision 19 of the NRC CoC 9204 and the supporting NRC SER. Revision 4 of the DOE CoC was issued with an expiration date of August 28, 2012, to closely coincide with the 90 day NRC restriction.

Summary

On August 23, 2012, NRC issued Revision 20 of the NRC CoC 9204 (10-160B) and the supporting SER. This revision approved the design changes requested by Energy*Solutions* Services, Inc (ES) (Note: NRC CoC 9204, Rev 20 reflected a name change to Energy*Solutions* Services Inc.) and removed the special 90 day conditions for transport that were stated in Revision 19 of the NRC CoC. When Revision 20 was issued, DOE PCP put Revision 4 of the DOE CoC 9204 under timely renewal until Revision 5 of the DOE CoC, based on Revision 20 of the NRC CoC, could be issued.

DOE PCP has reviewed the ES supplements that were submitted to NRC to justify the request for revision. The ES supplements included the *Consolidated Safety Analysis Report for Model 10-160B Type B Radwaste Shipping Cask, Revision 4, July 2012,* which was submitted to the NRC on July 20, 2012; changes to Chapter 7 of the consolidated safety analysis report (SAR) which was submitted to NRC on July 26, 2012; and the revised drawing DWG-CSK-12V01-EG-0002-01, Rev 3, *Cask Secondary Lid Thermal Shield Details* submitted to NRC on August 26, 2012. The consolidated SAR incorporated the information for all the supplements submitted to NRC for the previous revisions.

DOE PCP has reviewed the NRC SER for Revision 20 and concurs with the finding and actions of the NRC. The DOE SER for Revision 5 of the CoC is based on Revision 20 of the NRC CoC and incorporates the NRC SER. Revision 20 of the NRC CoC also contains elements and conditions that were added in Revisions 17 and 18 of the NRC CoC and were not shown in Revision 4 of the DOE CoC.

The justification for these changes made in Revisions 17 and 18 are addressed in the consolidated SAR. DOE PCP reviewed the NRC SERs for Revisions 17 and 18 and concurs with the finding and actions of these SERs. Elements and conditions of Revisions 17 and 18 have also been incorporated into Revision 5 of the DOE CoC. Revision 5 of the DOE CoC is equivalent to the NRC CoC Revision 20 except for approval of one additional content for shipment by DOE. Copies of the NRC SER for Revision 20, Revision 18, and Revision 17 are attached on the following pages.

SAFETY EVALUATION REPORT
Docket No. 71-9204
Model No. 10-160B
Certificate of Compliance No. 9204
Revision No. 20

SUMMARY

By application dated July 20, 2012, supplemented July 26 and August 10, 2012, Energy*Solutions* requested an amendment to Certificate of Compliance (CoC) No. 9204 to incorporate the addition of a thermal shield as an integral component of the package.

The applicant found that, under an Hypothetical Accident Condition (HAC) of a puncture test followed by a thermal test, the sheet metal covering the hollow region of the impact limiters may rupture and provide a direct heat path to the secondary lid and the baseplate of the package, thus exposing the seals to unacceptable temperatures and eventually leading to a loss of containment for the package. Therefore, a thermal shield plate has been attached to the secondary lid in order to protect the seals.

Revision 19 of the CoC was granted under the provisions of Title 10 Code of Federal Regulations (10 CFR) 71.41(c). The regulation in 10 CFR 71.41(c) states that the NRC may authorize a package using environmental and test conditions different from those specified in either 10 CFR 71.71, “Normal Conditions of Transport” and 10 CFR 71.73, “Hypothetical Accident Conditions,” if the controls proposed by the shipper are demonstrated to be adequate to provide the equivalent level of safety.

In order to demonstrate compliance with the requirements of 10 CFR 71.41(a) for this self-identified design issue and to add the thermal shield to all packages prior to each Type B shipment, Energy*Solutions* provided a revision to the consolidated safety analysis report, Revision No. 4, and updated drawings as supporting information to this amendment request.

NRC staff reviewed the applicant’s request and found that the addition of the thermal shield allows the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Package Description

There are no changes to the Model No. 10-160B package as described in the January 24, 2011, application other than (i) the addition of the thermal shield, and (ii) a limitation of the weight of the contents to 14,250 lbs from 14,500 lbs in previous revisions of the CoC.

The thermal shield protects the secondary lid. The thermal shield, consisting of two polished stainless-steel plates separated by a thin air gap, provides an additional air gap above the secondary lid. The thermal shield is attached to the secondary lid lifting lugs with hitch-pins.

1.2 Licensing Drawings

The staff reviewed Licensing Drawing No. DWG-CSK-12CV01-EG-0002, Rev. 3 for the secondary lid thermal shield and Energy*Solutions* Drawing No. C-110-D-29003-010, sheets 1 through 5, Rev. 16, for the packaging itself, and determined that the submitted drawings are adequate.

1.3 Findings

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. 10-160B package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL EVALUATION

The staff reviewed the application to verify that the addition of the thermal shield did not change the results of the previously approved structural analysis.

2.1 Structural Evaluation

The staff reviewed the structural evaluation of the deformation and/or damage to the shield in a scenario of a puncture bar going through the top hollow portion of the impact limiter's sheet-metal cover and contacting the thermal shield and the secondary lid bolts. The applicant showed that, although (i) the puncture bar causes minor damage to the central portion of the shield, and (ii) the shield plates may deform all the way to the lid with only minor damage, the top and bottom plates remain intact over most of their surface area and, as such, provide adequate thermal resistance during the thermal test. In addition, the applicant demonstrated that the secondary lid bolts remain covered by the thermal shield under this scenario.

The applicant also evaluated a scenario of a rod striking the bolt-head, assuming that the thermal shield does not provide any cover to the bolts. Even if the secondary lid comes into contact with the primary lid, the rod does not cause any damage to the lid. Further, the applicant demonstrated that a shear-out of the bolt head is not even possible in a scenario of a rod, inclined at a 27°angle from the lid surface to cause the maximum shear load onto the bolt head: the bolt shear strength is greater than that of the rod, i.e., 216,450 lbs versus 156,420 lbs. The staff reviewed the information provided and determined that the puncture bar will not cause any damage to the bolts even under this scenario.

2.2 Materials Evaluation

The thermal shield is made of ASTM A240 Type 304 austenitic stainless steel plates joined by ASTM A276 Type 304 sectioned pipes. The staff has confirmed that the material properties are acceptable. The staff notes that the minimum ASTM elongation strain at rupture for ASTM A276 Type 304 is 35%, while the finite element analysis (FEA) of the sectioned piping predicts a 40% strain, in the plastic regime following a puncture test. The FEA strain is reported in terms of true stress and strain, while the ASTM specification requires a minimum engineering strain. As such, the equivalent minimum engineering strain of the ASTM A276 Type 304 will be sufficient to prevent rupture of the sectioned pipes of the thermal shield during a puncture test.

Staff also noted a typographical error related to the schedule of the pipe used for the optional shield insert: the inner 8 inches SCH 60 steel pipe for the insert is in fact a SCH 40 pipe.

2.3 Findings

The structural analyses and calculations that were submitted for the addition of the thermal shield in this amendment request, have adequately confirmed that the design of the thermal shield is acceptable and in compliance with the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The addition of a thermal shield to the secondary lid of the package reduces the heat load on the package body during an HAC fire event and allows the seal temperatures to be kept below their maximum allowable temperature limits. The applicant performed a 2-D thermal evaluation, predicted a maximum O-ring seal temperature of 375°F when considering a scenario in which the puncture bar impacts the thermal shield, and concluded that the package is in compliance with 10 CFR 71.51.

3.1 Thermal Evaluation

In order to protect the containment seals during an HAC thermal event, a thermal shield, consisting of two stainless steel plates covering the entire secondary lid surface (the plate and the bolts) and separated by a thin air gap, was installed onto the secondary lid of the package. The thermal shield bottom plate is insulated with air-pocket from the “outside” surface of the secondary lid. Seven pipe stubs are welded to the thermal shield plates to act as stand-offs providing an additional air gap above the secondary lid. A 0.104 inch thick fire shield, with a 0.156 inch thick air gap between the shield and the outer structural shell of the packaging, also covers the exposed portion of the body of the packaging.

3.2 Modeling

The 200-watt internal heat load is applied as a constant heat flux over the exposed inner surface of the secondary lid using a 2-D finite element model ANSYS code that includes only the secondary lid. The total insolation is modeled to be 400 gcal/cm^2 for a 12-hour period for curved surfaces during the post-fire cooldown, according to 10 CFR 71.71. The ambient temperature is set up as 1475°F during the 30-min fire transient and 100°F during post-fire cool-down. For radiation heat transfer between the thermal shield and the environment, an emissivity of 0.9 is specified for the 30-minute fire transient, and a calculated emissivity of 0.7347 is used during post-fire cool down. Heat transfer is enhanced from the ambient fire to the thermal shield by forced convection during the 30-minute fire transient and is reduced from the thermal shield to the ambient air by natural convection during the cool-down period. The proposed thermal model is acceptable for predicting the secondary lid seal temperatures.

The major assumptions used in the applicant’s 2-D model (excluding the primary lid in the model) are listed below.

(i) The scenario in which the puncture bar impacts the thermal-shield is addressed. The damaged thermal-shield model assumes that, even in a deformed shape, the thermal shield remains in close contact with the lid and hence, transfers a large amount of heat to the secondary lid during a 30-minute fire transient.

(ii) The heat conduction between the primary and secondary lids is neglected to eliminate the heat loss from the secondary lid to the primary lid.

(iii) The radiation heat transfer between the primary and secondary lids is neglected to reduce the heat loss from the secondary lid to the primary lid.

(iv) The two circular plates of the secondary lid are assumed to be totally connected as a solid plate to increase the heat input during 30-minute fire transient.

(v) A larger radiation emissivity of 0.3 is used, instead of 0.15, between two thermal shield plates during the 30-minute fire transient. The use of a radiation emissivity of 0.3 increases the heat input into the secondary lid and the containment seals.

The staff reviewed these assumptions and found them acceptable.

3.3 Findings

The staff confirmed that (i) the thermal-shield design features are adequately described and evaluated, (ii) the 2-D thermal model, including only the secondary lid, is discussed and described in sufficient detail for verification of thermal-shield effectiveness, (iii) the assumptions, used in the analysis, provides conservative predictions and, (iv) a calculated maximum seal temperature of 375°F is obtained from the damaged thermal-shield model.

The staff finds that the model analysis addressing the scenario in which the puncture bar impacts the thermal-shield is acceptable and that all temperatures in the package components due to HAC conditions are below their maximum allowable limits. The maximum HAC seal temperature of 375°F, during the cool-down period of the fire transient, has a "built-in" margin of safety. The maximum temperature in the lead shielding is calculated to be 274°F, well below the melting point of 622°F, and the steel body of the package is also well below the materials' service limits. Thus, the staff finds that the secondary lid seal will maintain its containment function during a fire transient and a post-fire cool-down period.

The staff has reasonable assurance that the thermal shield is capable of substantially reducing the secondary lid seal temperatures during an HAC fire accident and that the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The applicant has proposed no containment changes to the Model No. 10-160B package design. The staff determined there is reasonable assurance that the containment seals will maintain their containment function during HAC.

5.0 SHIELDING

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

6.0 CRITICALITY EVALUATION

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

7.0 PACKAGE OPERATIONS

The applicant has proposed minor changes to the operating procedures of the package to account for the installation or removal of the thermal shield during package loading and unloading, and of its associated anti-tamper seals.

The staff reviewed the changes and determined they were acceptable.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant has proposed no changes to the general acceptance tests and maintenance program of the package.

CONDITIONS

The conditions specified in the Certificate of Compliance have been revised to incorporate several changes as indicated below:

Item No. 3(a) was revised to include the new name of the Certificate holder, Energy*Solutions* Services, Inc.

Condition No. 5(a)(2) has been revised to (i) include the thermal shield in the description of the package, (ii) the reduced weight of the contents to 14,250 lbs from 14,500 lbs, and (iii) a typographical error related to the schedule of the pipe used for the optional shield insert (the inner 8 inches SCH 60 steel pipe for the insert has been replaced with a SCH 40 pipe).

Condition No. 5(a)(3) has been revised to include Drawing No. C-110-D-29003-010, sheets 1 through 5, Rev. 16, and the new drawing related to the Secondary Lid Thermal Shield, Drawing No. DWG-CSK-12CV01-EG-0002-01, Rev. 3.

Condition No. 13 on the use of CoC Revision No. 17 until December 31, 2012, was removed because it became obsolete. Since all existing packages have been modified to include the required thermal shield, no previous revision of the certificate is authorized.

The reference section was updated to include the supplements dated July 20 and August 10, 2012. The expiration date of the certificate was not changed.

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. 20,
on August 23, 2012.

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. 18

SUMMARY

By letter dated September 9, 2011, supplemented October 28, 2011, Energy*Solutions* requested the addition of a Source Insert containing up to 10,000 Ci of ^{60}Co as authorized contents in the Model No. 10-160B package.

NRC staff reviewed the applicant's request and the supplement dated October 28, 2011, and found that it did not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Package Description

There are no changes to the Model No. 10-160B package as described in the January 24, 2011, application, as supplemented April 6, September 9 and October 28, 2011. The applicant is proposing an additional component to the Model No. 10-160B package design with a Source Insert included into the cavity of the package.

The Source Insert adds photon (gamma) shielding to satisfy Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) dose rates. The side walls of the insert consists of lead with a total thickness of 6.0 inches, located between an inner 8.0-inch (nominal) schedule (SCH) 60 steel pipe and an outer 24.0-inch (nominal) SCH 60 steel pipe. The bottom consists of 6.0 inches of lead supported by a 0.75-inch thick steel base plate. The lid includes a steel encased lead plug (nominal lead thickness 8 7/8 inches), steel bolting plate, and flat silicon rubber gasket. Figures 5-2 and 5-3 of the August 2011 addendum to the application illustrate the source insert.

The steel cribbing used to support the source insert inside the package weighs less than 5,000 lbs. Therefore, the combined maximum load inside the Model No. 10-160B package will be 10,500 lbs, i.e., much smaller than the licensed weight of 14,500 lbs. The Source Insert is placed in the package cavity in such a way that its center of gravity is located near the center of gravity of the package. Therefore, the combined centers of gravity of the Model No. 10-160B package will remain at the same location as that already analyzed in the application.

1.2 Licensing Drawings

The staff reviewed Licensing Drawing Nos. C-038-145083-004, Rev. 0 and C-038-145083-005, Rev. 0, for the Source Insert assembly and details and the Source Insert steel cribbing, respectively. The staff determined that the submitted drawings are adequate. In particular, the staff noted that the safety classification for each component of the Source Insert is included on the Licensing Drawings.

1.3 Contents

With the addition of a shielded source insert, the Model No. 10-160B package can transport up to 10,000 Ci of ^{60}Co (approximately 9 grams) normal form radioactive material. The total mass of the contents, including any material encapsulating the normal form ^{60}Co radioactive material, dunnage, optional secondary containers, etc., does not exceed 500 pounds.

Staff noted that the structural calculations use 500 pounds as the mass of the contents. Staff also noted that no specific configuration of the radioactive material contents is stipulated because the shielding evaluation, in Chapter 5 of the Addendum, assumes that the radioactive material collapses to essentially a point source during HAC, and is also conservatively assumed as air in NCT.

1.4 Finding

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. 10-160B package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL EVALUATION

The staff reviewed the application to verify that the changes made to the package design, as part of this amendment request, meet the structural requirements of 10 CFR Part 71 under NCT and HAC. The staff also reviewed the application to determine whether the package fulfills the acceptance criteria of NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material.”

2.1 Evaluation

The applicant provided a structural evaluation in Report ST-663, Rev. 1, and demonstrated that the Source Insert, when placed inside the package, survives NCT and HAC loadings, while still satisfying the regulatory dose limits of 10 CFR 71.47 and 10 CFR 71.51.

Section 6.1 of ST-663, Rev. 1, evaluates the Source Insert and contents against the 10 CFR 71.45 regulatory lifting requirement of 3 against yield, and concludes that the new design has an ample margin (with a safety factor of 1.29). The package lifting and tiedown criteria are not re-evaluated as part of this amendment request since the previously approved application (weights) bounds the addition of the Source Insert (including the contents), as indicated in Section 2.4 of Addendum, August 2011. Staff also noted that, per Note 2 of licensing drawing No. C-038-145083-004, Rev. 0, the applicant will be testing the lifting lugs to 150% of the service load, and will also inspect them.

Section 6.3 of ST-663, Rev. 1, evaluates the package with the new insert against the 10 CFR 71.71 NCT drop cases. The applicant uses deceleration ratios of the previously approved package application to claim that HAC scenarios bound NCT scenarios. Staff noted that the applicant did not encompass the different service levels of ASME allowable stresses, per Regulatory Guide 7.6, but makes a ratio-based quantitative comparison. Staff believes this is a reasonable approach and concludes that NCT conditions are met.

Section 6.2 of ST-663, Rev. 1, evaluates the package with the new insert against the 10 CFR 71.73 HAC drop cases. The applicant uses the global deceleration values determined in the original package evaluation for the various drop cases, and applies them to the applicable components of the Source Insert. Staff noted that the applicant has not provided an acceptance criterion (other than against maximum allowable stresses) for the containment of the insert and that there was no discussion on the HAC crush or puncture tests (10 CFR 71.73(c)(2) and (3)) in the application. However, such conditions are bounded by the results of the original application (defining the containment boundary of the package) and the new Source Insert is not susceptible to failure under these test conditions.

Section 6.4 of ST-663, Rev. 1, evaluates the new insert against lead slumping and the potential loading induced to the shielded insert from the cribbing. The analysis of the lead shielding concludes that the flow stress generated during the HAC end drop is lower than the allowable, and that the lead shielding will not permanently deform. Staff concurs with this determination that the lead shielding will not slump and cause an increase in the HAC dose rates. Section 6.4.2 of ST-663, Rev. 1, details the cribbing load analysis. The analysis considers the effects of the bottom, middle, and top segments of the cribbing mass, and conservatively uses the cribbing yield stress (while not considering ASME code allowables) as the acceptance criteria for qualifying the cribbing members. The analysis considers the effects of HAC end and side drop tests (inputs of 176g and 121g respectively). The analysis also considers the damage that may occur from the effect that the cribbing members exert against the source insert. In general, NRC staff concludes that the analysis provides a reasonable justification that the new package configuration has adequate safety margin and meets the HAC drop test conditions and corresponding acceptance criteria.

2.2 Findings

All structural analysis and calculations that were submitted have adequately confirmed that the new design is acceptable and in compliance with the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The staff reviewed the application to verify that the changes made to the package, as part of this amendment request, meet the thermal requirements of 10 CFR Part 71 under NCT and HAC. The staff also reviewed the application to determine whether the package fulfills the acceptance criteria of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

3.1 Evaluation

For the configuration evaluated in this application, the heat load of the ^{60}Co source is 153.9 W, as discussed in Section 3.1 of the addendum, but a generic heat load of 200 W is used for the thermal evaluation as a conservative value. The thermal analysis for NCT and HAC has been performed using a 2-dimensional axisymmetric finite element model.

The resulting temperatures for the NCT and HAC analysis are presented in Tables 3.1-3 and 3.1-4 of the application, respectively. The temperature values for the various components analyzed remain under the allowable limits specified in the original application. However, it was noted that the resulting values of bulk air temperature within the package were higher than the ones presented in the original application. The staff noted that the bulk air temperature reported in the evaluation (298°F) corresponds to the maximum temperature of the air inside the package reached anywhere during the HAC fire event. This temperature is conservatively compared with the average bulk air temperature reported in the application. These maximum bulk air temperatures were used as conservative values to determine the maximum internal pressure of the package for NCT and HAC. The maximum internal pressures calculated were 3.6 psig and 6.4 psig, for NCT and HAC, respectively. Both pressures were significantly under the allowable values presented in the original application.

3.2 Findings

After reviewing this amendment request, the staff determined that the materials temperatures and internal pressures are acceptable for NCT and HAC. Staff concludes that this thermal evaluation satisfies the regulatory requirements of 10 CFR 71.71 and 10 CFR 71.73.

4.0 CONTAINMENT EVALUATION

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

5.0 SHIELDING

The staff reviewed the application to verify that the changes made to the package, as part of this amendment request, provide adequate protection against radiation and meet the external radiation requirements of 10 CFR Part 71 under NCT and HAC. The staff also reviewed the application to determine whether the package fulfills the acceptance criteria listed in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

5.1 Description of Shielding Design

5.1.1 Shielding Design Features

The shielding design features of the Source Insert include the following: (i) inner shell sidewalls of cast lead (density of 11.3 g/cm^3 and a total thickness of 6.72 inches) located between an inner 8.0-inch (nominal) SCH 40 steel pipe and an outer 24.0-inch (nominal) SCH 60 steel pipe, (ii) a base consisting of 6.0 inches of lead supported by a 0.75-inch thick steel base plate, and (iii) an upper shield lid consisting of 8.625 inches of lead encased with steel.

Additional shielding is also provided by the 10-160B overpack shielding itself consisting 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. The package lid is a 5-1/2-inch thick carbon steel plate with a 31-inch diameter opening equipped with a secondary lid.

5.1.2 Summary Tables of Maximum Radiation Levels

The applicant performed the shielding analysis using the MCNP5 computer code (Reference 2) and calculated NCT dose rates with a 10,000 Ci ^{60}Co distributed source located in the center of the package cavity.

Table 5-3 of the application summarizes the dose rates for NCT. Although the 10 CFR 71.47(b)(3) limit is at 2 meters from the external surface of the vehicle, the applicant calculated a maximum dose rate at 2 meters from the outer surface of the package and conservatively used this value to satisfy the 10 CFR 71.47(b)(3) limit.

Peak NCT Dose Rates for the 10-160B Package with Centered Source Insert (mrem/hr)

NCT	Package Surface			2 Meters from Package Surface		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	2.9	0.8	6.5	0.2	0.1	NA[1]
Neutron	NA[2]	NA[2]	NA[2]	NA[2]	NA[2]	NA[2]
Total	2.9	0.8	6.5	0.2	0.1	NA[1]
10 CFR 71.47(b) Limits	200	200	200	10	10	10

[1]The package is assumed centered on a 8-foot wide conveyance with the outer surface of the package approximately 40 inches (101.6 cm) above ground level. A dose rate +2m from the base is not available.
[2]The source consists solely of Co-60 and contains no neutron-emitting radioisotopes.

The applicant also calculated HAC dose rates (see Table 5-4 of the application) by conservatively assuming a 10,000 Ci ^{60}Co point source with the point source being relocated to the worst-case geometric configuration following an accident. Although the 10 CFR 71.51(a)(2) limit is at 1 meter from the external surface of the package, the applicant calculated a maximum dose rate at any point on contact with the outer surface of the package and conservatively used this value to satisfy the 10 CFR 71.51(a)(2) limit.

Peak HAC Dose Rates for 10-160B Package with Source Insert at Worst-case Geometric Configuration Following an Accident (mrem/hr)

HAC	1 Meter from Package Surface		
Radiation	Top[1]	Side[2]	Bottom[2]
Gamma	44.5[3]	27.8[3]	27.5[3]
Neutron	NA[4]	NA[4]	NA[4]
Total	44.5	27.8	27.5
10 CFR 71.51(a)(2) Limit	1000	1000	1000

[1]The bounding top HAC dose rate occurs with the insert inverted and positioned against the package inner top surface.
[2]The bounding side and bottom HAC dose rates occur with the insert lying sideways on the package inner bottom surface.
[3]Dose rates are conservatively calculated on contact with the package surface.
[4]The source consists solely of Co-60 and contains no neutron-emitting radioisotopes.

The staff reviewed Tables 5-3 and 5-4 of the application to ensure that the package meets the requirements in 10 CFR 71.47 and 10 CFR 71.51. The staff also verified that the dose rate in any normally occupied space (i.e., driver location) under normal conditions is less than the allowable limit of 2.0 mrem/hr, thus demonstrating compliance with 10 CFR 71.47(b)(4) for exclusive use shipments.

5.2 Radiation Source

5.2.1 Gamma Source

The maximum quantity of radioactive material that will be shipped with the Source Insert is 10,000 Ci of ^{60}Co. Table 5-5 of the application lists the photon emission probability for ^{60}Co as a function of energy. The applicant references the "NuDat" database at the National Nuclear Data Center of BNL. Conservatively assuming two photons per decay, the photon source activity for 10,000 Curies of ^{60}Co is 7.4 10^{14} photons/sec.

As shown in Table 5-5, only photons with energies above 0.347 MeV are included in the shielding calculations because photons with energies below 0.347 MeV are too weak to penetrate the steel of the package.

The staff reviewed the gamma source term for the proposed contents and also verified that the applicant specified the gamma source term as a function of energy.

5.2.2 Neutron Source

The ^{60}Co source term contains no neutron emitting radioisotopes.

5.3 Shielding Model

Chapters 2 and 3 of the application show that NCT tests do not impact the geometry of the source insert's shielding and that there will be no damage to the shielding of the source insert as a result of HAC. Thus, the packaging, previously reviewed and approved by the staff and documented in Reference 1, is not impacted by the addition of the Source Insert. The staff finds that the shielding model is consistent with the effects of the tests performed in compliance with 10 CFR 71.71 and 10 CFR 71.73. The staff finds that the application provides the basis for finding that the package has been adequately described and evaluated against NCT and HAC, as specified in 10 CFR 71.71 and 10 CFR 71.73.

There are no changes in the lead slump evaluation, which was previously evaluated by staff in Reference 1. The applicant's structural analysis of the Source Insert under HAC conditions demonstrates that the effects of lead slump are negligible (maximum deformation of 0.0525 in.) and that most of this deformation is recovered resulting in a very small lead slump, if any. Therefore, the applicant did not model lead slump in the shielding analysis. The staff finds such an approach to be reasonable.

5.3.1 Configuration of Source and Shielding

The staff examined Figures 5-1 through 5-11 of the application, the description of the modeling, and the MCNP input decks to determine how the shielding is modeled. The staff verified that the Source Insert dimensions are consistent with the licensing drawings. The applicant used nominal dimensions to model the package. The staff finds this approach acceptable because of the low dose rates.

5.3.1.1 Normal Conditions of Transport

The insert is modeled as concentric cylinders surrounding a source cavity 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 10,000 Ci gamma source is modeled as a distributed source over the cavity volume. The composition of the source is conservatively modeled as air to conservatively eliminate any self-shielding due to the source material. The shoring is not included in the model to conservatively eliminate the small amount of added internal shielding provided by the shoring.

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

- External surface of the transport vehicle (axial and radial), and
- 2 meters from the surface of the transport vehicle.

The applicant did not calculate a dose point at 2 meters from, or on contact with, the external surface of the transport vehicle but calculated a maximum dose rate at 2 meters from the outer surface of the package and on contact with the external surface of the package and conservatively used these values to satisfy the 10 CFR 71.47(b) limits. Because these limits were met at 2 meters from, or on contact with, the packaging, the staff finds that it would meet the same limits with regard to 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).

The staff verified that the applicant considered potential streaming effects from the Source Insert. As part of this verification, the staff reviewed the applicant's model of the drain line for the Source Insert, which by its nature is a potential streaming path. The drain line allows water to drain from the insert after loading ^{60}Co sources stored underwater. The applicant calculated the peak dose rate at the discharge to the drain line to be 1,058 mrem/hr, which is less than the peak dose rate of 1,250 mrem/hr (see Tables 6-1 and 6-4 of the application) on contact with the outer radius, which occurs at a higher elevation. Based on this comparison, the staff finds the drain line was adequately designed and does not adversely impact external radiation levels.

5.3.1.2 Hypothetical Accident Conditions of Transport (HAC)

As noted in Section 2.7.1.1.3 of the application, HAC conditions do not affect the shielding of the source insert, and there is little, if any, lead slump in the insert. Therefore, no lead slump was modeled in the shielding calculations. However, under HAC, neither the shoring nor the ^{60}Co source is expected to maintain structural integrity, and thus the source insert was modeled at various locations in the package cavity. The applicant selected four configurations for evaluation as the bounding geometries during HAC conditions as shown in Figures 5-8 through 5-11 of the application. Dose rates were determined at 1 meter from the package sidewall, top, and bottom. The staff finds this acceptable.

For all HAC cases, the ^{60}Co source term was modeled as a 1-cm sphere positioned against the interior side of the insert closest to the associated package exterior surface. The 1-cm sphere is essentially a point source, which models an extreme collapse of the ^{60}Co pins and conservatively maximizes the resulting HAC dose rates.

The staff verified that the applicant has a dose point 1 meter from the surface of the package for HAC as specified by 10 CFR 71.51(a)(2). The dose point was located in line with the point source to give the highest results. The staff finds this acceptable.

5.3.2 Material Properties

The staff verified that the applicant identified the materials and mass densities of the shielding materials for the source insert. Table 5-6 of the application provides a summary of the materials and their properties used in the shielding models. The applicant identified the inner and outer shell as carbon steel with a density of 7.82 g/cm^3, and the lead shield as commercial grade lead with a density of 11.30 g/cm^3. Such values are typical for shielding materials and are reasonable for use in the shielding analysis. The lead shielding was conservatively modeled with impurities at maximum values and minimum lead content. In addition, because the applicant assumes a point source for HAC conditions, no assumption is made regarding self-shielding, and thus, the staff finds that this is conservative. The staff also finds that the selected material compositions and densities are appropriate and provide reasonable assurance that the materials densities are adequately modeled for the shielding of the Source Insert.

The staff reviewed the MCNP input decks described in Section 5.5.2 of the application and confirmed that materials data used in the input are correctly selected.

5.4 Shielding Evaluation

5.4.1 Methods

For the shielding analysis the applicant used the MCNP5 computer code with photon and neutron cross-section sets designated ".04p," obtained from the ENDF/B-VIII photon data library provided with MCNP5. MCNP is a three dimensional code, widely used for shielding analyses, that staff has accepted for similar shielding evaluations.

Six models (two for NCT and four for HAC, as shown in Figures 4-1 through 4-6 of the application) were developed to determine external radiation levels at the package surface, at 2 meters from the package surface for NCT, and at 1 meter from the package surface for HAC.

For NCT, the applicant assumed a distributed source, which is appropriate due to the nature of the Co-60 pin sources being evenly distributed inside the Source Insert. For HAC, the applicant assumed the Co-60 pins are collapsed to approximately a point source (i.e., modeled as a 1-cm sphere), conservatively maximizing the resulting HAC dose rates. The four HAC configurations were selected to investigate various bounding geometry and source placement configurations of the source insert within the central cavity of the package.

5.4.2 Key Input and Output Data

The staff performed a review of the MCNP input decks provided by the applicant to ensure that the geometry and materials were appropriately specified. The staff determined that the selected detector locations are appropriate to detect possible radiation streaming paths. The staff also reviewed the output files provided by the applicant 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).

5.4.3 Flux-to-Dose-Rate Conversion

The staff confirmed that the applicant used the ANSI/ANS 6.1.1-1977 (Reference 3) standard and finds this acceptable. The applicant performed the gamma flux-to-dose-rate conversion using the MCNP code and Table 3 from the standard.

5.4.4 Radiation Levels

The staff verified that the analysis showed that the locations selected are those of maximum radiation levels and include any radiation streaming paths. 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 independent confirmatory analyses using the MAVRIC sequence of the SCALE6 code (Reference 4) with the v7-27n19g cross section library to model gamma radiation. 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

The staff reviewed the description of the package design features related to shielding and the source term for the insert and found them acceptable. The methods used are consistent with accepted industry practices and standards. The staff reviewed the maximum dose rates for NCT and HAC and determined that the reported values were below the regulatory limits in 10 CFR 71.47 and 10 CFR 71.51 for an exclusive use package.

Based on the staff's review of the statements and representations in the application and the results of staff's confirmatory analyses, the staff concludes that the design of the Model No. 10-160B package, with contents of 10,000 Ci of ^{60}Co placed in the Source Insert, provides a reasonable assurance to meet 10 CFR Part 71 requirements.

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. 15, February 28, 2011, ADAMS Accession No. ML110610496.
2. MCNP5 – 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).
3. American Nuclear Society, ANSI/ANS 6.1.1 1977, Neutron and Gamma-Ray Flux-to-Dose-Rate Factors, La Grange Park, Illinois.
4. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 6, Vols. I-III, January 2009.

6.0 CRITICALITY EVALUATION

Not applicable.

7.0 PACKAGE OPERATIONS

A specific procedure for loading the Source Insert and loading the Insert into the package with the specified cribbing has been added to the general procedures for loading and unloading the Model No. 10-160B package. It is to be noted that the Source Insert will not be unloaded.

In particular, the package cavity shall be vacuum dried if the insert is loaded underwater. Radioactively contaminated liquids are pumped out, removed by the use of an absorbent material or via the drain line. Leak testing is required when seals are replaced (including seals on the optional vent and drain ports): the pressure drop leak test of the package primary lid, secondary lid, vent line, or drain line (as applicable) is performed in accordance with the Chapter 8 of the application.

The staff reviewed the operating procedures and concludes that the operating procedures both meet the requirements of 10 CFR Part 71 and 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 applicant has proposed no changes to the general acceptance tests and maintenance program of the package. However, acceptance tests of the Source Insert, pertaining to visual examination, structural testing, load testing, and shielding integrity testing, have been included.

There are no routine or periodic maintenance activities for the Source Insert which is a single use container.

CONDITIONS

The conditions specified in the Certificate of Compliance have been revised to incorporate several changes as indicated below:

Item No. 3.b has been revised to identify Energy*Solutions*' request dated April 6, 2011.

Condition No. 5(a)(2) has ben revised to describe the Source Insert. The source insert design weight is 8,000 lbs; it has side walls consisting of 6.0-inch thick lead, sandwiched between an inner 8 inches nominal SCH 60 steel pipe and an outer 24.0-inch SCH 60 steel pipe. The bottom of the source insert also consists of lead supported by a 0.75-inch thick steel base plate. The lid includes a steel encased lead plug, steel bolting plate and flat silicon rubber gasket.

Condition No. 5(a)(3) has been revised to include two new drawings: Energy*Solutions* Drawing No. C-038-145083-004, Rev. 0, and Drawing No. C-038-145083-005, Rev. 0.

Condition No. 5(b)(1)(vi) has been added to include byproduct material as normal form solid metal loaded into the new Source Insert.

Condition No. 5(b)(2)(i) has been revised to add maximum authorized contents with a limit of 10,000 Ci of ^{60}Co.

Condition No. 5(b)(2)(vi) has been added to specify that the contents of the source insert have a maximum weight of 500 pounds.

Condition No. 14 has been added to authorize use of the previous revision of the certificate for a period of approximately one year.

As a result of the addition of Condition No. 14, Conditions No. 12 and 13 of the previous certificate were renumbered.

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. 18, on January 11, 2012.

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. 17

SUMMARY

By application dated April 6, 2011, Energy*Solutions* requested that (i) the maximum content limits for the Model No. 10-160B package be defined for a Co-60 point source, and (ii) a provision for a shipment of low specific activity (LSA) waste within 10 days of preparation, or within 10 days after venting of drums, be added to the Certificate of Compliance (CoC).

NRC staff reviewed the applicant's requests and found that these requests did 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

Staff's Safety Evaluation Report for the Model No. 10-160B CoC, Revision No. 15, explained in detail why specifying maximum allowable contents only in multiples of A_2 was not sufficient. Staff mentioned 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 a 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 source, i.e., neutron or gamma, nor the energy spectra of the content, and the A_2 value is independent of the shielding performance for the specific source;

(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;

(4) The A_2 value can be modified with revised regulations independent of the package approvals;

(5) Thermal evaluations rely upon knowing the decay heat of the source. This value is difficult to verify by direct measurement so it is typically calculated based on the individual nuclides allowed in the maximum source specification;

(6) Content's limits in terms of thousands of A_2 will significantly exceed the shielding capability of the package.

Thus, contents cannot be defined as "Type B quantity of radioactive material, not to exceed 3,000 A_2." On the other hand, staff also recognized that it is still appropriate to use A_2 values to (i) provide an appropriate categorization of the package and (ii) limit the prescribed leak rate, i.e., 3000 A_2 in the case of the Model No. 10-160B package.

Attachment 1 to Chapter No. 7 of the application provides a methodology that allows 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. For contents with multiple energies or multiple radioactive isotopes, users calculate the fractional dose rate contribution from each energy bin of the source. Thus, condition No. 5(b)(2)(i) of the CoC now states that "the maximum quantity of radioactive material is determined to be the lesser of the quantity found by the methodology described in Attachment 1 to Chapter No. 7 of the application or the 3000 A_2 limit prescribed by the package leak rate."

The applicant requested that the CoC also includes a limitation of the contents for a Co-60 source as follows: "Maximum contents are limited to 0.495 TBq (13.4 Ci) for a Co-60 point source." Staff disagreed with this request because "point sources" are not licensed or certified as contents. Further, the condition as written could have different regulatory interpretations such as (i) there is only a radioactivity limit for Co-60, but only if loaded as a point source; thus no Co-60 limits exist if it is distributed across a waste volume, (ii) only a single Co-60 point source up to 13.4 Ci is allowed in the package, and waste with distributed Co-60 is not allowed, etc. Staff evaluated a 13.4 Ci Co-60 point source using the methodology described in Attachment 1 to Chapter No. 7 of the application. Co-60 emits two gamma rays with energies of 1.17 and 1.33 MeV. Calculations performed by staff show that a 13.4 Ci point source of Co-60 will have an activity lower than the maximum activity allowed by the point source curve. Therefore, the staff finds that there is no need to include any specific activity limit on Co-60 as a point source in Condition No. 5(b)(2)(i), since the limit is already bounded by the point source curve.

Condition No. 5(b)(2)(i) of the CoC, as now written, includes de facto the maximum activity value for Co-60 as a point source as found from the point source curves corresponding to the shielding analysis.

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

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

5.0 SHIELDING

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

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 applicant requested staff to review and re-insert in the CoC one of the conditions of the CoC Rev. No. 14, specifically related to the shipment of LSA waste materials. The staff reviewed Information Notice (IN) 84-72, the Safety Analysis Report for the Model No. 10-160B package (Rev. 0) and its accompanying Safety Evaluation Report, as well as some of the referenced reports of BNL-NUREG-28682, NUREG/CR-2830, and DOE GEND-041 (issued by EG&G Idaho, Inc., in 1986). The staff also performed a confirmatory analysis to validate this request. The staff's findings are delineated below:

(1) The staff reviewed IN 84-72, and related reports such as BNL-NUREG-28682 "Review of Recent Studies of the Radiation Induced Behavior of Ion Exchange Media (1980)," NUREG/CR-2830 "Permissible Radionuclide Loading for Organic Ion Exchange Resins from Nuclear Power Plants (1983)," and DOE GEND-041 "A Calculational Technique to Predict Combustible Gas Generation in Sealed Radioactive Waste Containers (1986)." The staff confirmed that the 10-day condition for shipment of LSA materials, as mentioned in IN 84-72, was (i) first established by on-site sampling tests and measurements from a DOE/NRC joint gas generation research program started after the Three Mile Island (TMI) accident and, (ii) further validated by analyses which addressed the correlation of hydrogen generation with the storage time period of LSA materials. As a result of its review, the staff confirmed that the 10-day condition had an appropriate technical basis.

(2) An NRC approved Type B Package for LSA material with a dose rate greater than 1 Rem/hr at three meters from the unshielded source is required by 49 CFR 173.427 while a DOT approved Industrial Package is allowed for LSA material with a dose rate less than 1 Rem/hr at three meters from the unshielded source. Staff reviewed the methodology developed by the DOE/NRC joint gas generation research program and concluded that the potentially different packages used to ship LSA material should not affect the hydrogen generation methodology (including those parameters used in the calculations) which conservatively predicts a low hydrogen generation within twice the

shipping period. In view of the performed analyses, the staff validates that there is no safety concern with hydrogen generation in a package if LSA material shipment takes place within 10 days of preparation or within 10 days after venting.

(3) Given the LSA material's characteristics and corresponding properties for radiolytic reactions, there should be no significant chemical, galvanic, or other reactions with LSA material, as required by 10 CFR 71.43(d). Even if some organic material is contained in the LSA waste, the quantity of hydrogen generated through radiolysis will be very limited if the package is shipped within the 10-day condition.

(4) The staff performed a confirmatory analysis of hydrogen generation with LSA material in the Model No. 10-160B package. The bounding analysis was performed using the maximum decay heat (200 watts), the maximum allowable limit of hydrogen in volume (5%), the maximum energy emission and absorption fraction (1.0), and the conservative effective G value in radiolysis (0.6 molecules per 100 eV for LSA material) to minimize the allowable shipping time for hydrogen generation. Calculations were performed with the weight fractions of water, 1% and 2%, respectively, contained/absorbed in the LSA materials. Both calculations show that (i) it would take significantly longer than 100 days to reach the 5% hydrogen generation limit for the LSA material, and (ii) it should not generate hydrogen above the 5% limit, if the package is shipped within the 10-day condition.

Based on these findings, the staff accepts the applicant's request to re-insert the 10-day condition in the CoC and concludes that the operating procedures both meet the requirements of 10 CFR Part 71 and 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 applicant has proposed no changes to the acceptance tests and the maintenance program.

CONDITIONS

The conditions specified in the Certificate of Compliance have been revised to incorporate several changes as indicated below:

Item No. 3.b has been revised to identify Energy*Solutions*' request dated April 6, 2011.

Condition No. 5(b)(2)(i) has been revised to clarify the definition of the maximum quantity of material per package: the maximum quantity of radioactive material is determined to be the lesser of the quantity found by the methodology described in Attachment 1 to Chapter No. 7 of the application or the 3000 A_2 limit prescribed by the package leak rate.

Condition No. 8 has been revised to add that for contents with a radioactivity concentration not exceeding that for Low Specific Activity material, the hydrogen concentration can be assumed to be less than 5% provided the package is shipped within 10 days of preparation, or within 10 days after venting of drums or other secondary containers.

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. 17,
on August 26 , 2011.