SAFETY EVALUATION REPORT

Docket No. 71-9338
Model No. 3977A
Certificate of Compliance No. 9338
Revision No. 0

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SUMMARY

By application dated September 29, 2012, as supplemented on December 20, 2012, April 23, September 20, November 21, and 28, December 13, 16, and 17, 2013, and January 6, 10, 27, and 31, and February 11, 2014, Croft Associates Limited requested that the Nuclear Regulatory Commission approve the Model No. 3977A package as a Type B(U) package for the transport of radioisotopes used in a wide range of therapeutic and diagnostic applications and research.

The packaging consists of an inner containment vessel, which also provides shielding, and the outer stainless steel keg and insulating cork packing, which provides impact and thermal protection. The containment vessel has stainless steel inner and outer shells enclosing depleted uranium (DU) shielding in both the body and the containment vessel lid. The dimensions of the package are approximately 424 mm in diameter at the top and bottom rims, and 585 mm in overall height. The maximum weight of the package and contents is 163 kg.

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, and the conditions listed in the certificate of compliance (CoC), the staff concludes that the package meets the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging

The Model No. 3977A packaging, the Safkeg-HS 3977A, consists of an outer stainless steel keg enclosing insulating cork packing, and an inner containment vessel. There are three shielding inserts designed for use in the Model No. 3977A, designated as Shielding Insert Design Nos. 3982, 3985, and 3987. However, only Shielding Insert Design Nos. 3982 and 3985 will be authorized for use with the Safkeg-HS 3977A with the original issue of the certificate of compliance. Therefore, further discussion of the shielding inserts in the safety evaluation report will only address these shielding inserts. The outer keg provides impact and thermal protection. Containment is provided by the containment vessel. Shielding is provided by the containment vessel and shielding insert.

The keg has a stainless steel outer shell and a stainless steel liner, between which insulating cork is fitted. The keg lid is attached to the body by eight stainless steel studs and nuts, with a single O-ring weather seal. An inner cork liner is fitted between the keg liner and the top and sides of the containment vessel, consisting of a cork body and cork top, with no cork between the bottom of the containment vessel and the keg liner.

The containment vessel consists of a body and lid. The body has a stainless steel outer wall, base, and flange/cavity wall. The flange/cavity wall is welded to the outer wall to form a cavity into which DU shielding is placed. The DU shielding thickness is approximately 46 mm at the base of the CV and 47.6 mm along the side of the CV except at the top where the lid seats. The DU shielding thickness at the top is 22.25 mm thick. After the DU shielding is installed, the base is then welded to the outer wall. The containment vessel lid top and lid shielding casing are stainless steel, with 45.9 mm thick DU inside. The containment vessel lid is secured by eight, M-10x1.5x20, alloy steel recessed hexagon socket head cap screws. The containment vessel is sealed by two concentric fluoroelastomer O-rings, and the lid is equipped with a leak test port.

There are two Shielding Inserts authorized for use in the Model No. 3977A packaging. Design Insert No. 3982, HS-12x95-Tu, is a tungsten insert with inner cavity size of 12 mm diameter by 95 mm height. The approximate mass of the insert is 9.2 kg. Design Insert No. 3985, HS-31x114-Tu, is a tungsten insert with inner cavity size of 31 mm diameter by 114 mm in height. The approximate mass of the insert is 7.9 kg.

The radioactive material shall be enclosed in a convenient product container such as a quartz vial or aluminum capsule. Irradiated items may be carried in a plastic or metal can or wrapping to minimize the contamination of the insert.

The approximate dimensions and weight of the package are:

Overall package outer diameter	424 mm
Overall package height	585 mm
Containment vessel outer diameter	200 mm
Containment vessel height	302.5 mm
Containment vessel cavity inner diameter	65.8 mm
Containment vessel cavity inner height	157.1 mm
Maximum package mass	163 kg

1.2 Drawings

The packaging is constructed and assembled in accordance with Croft Associates Limited Drawing Nos:

1C-5940, Rev. E	Cover Sheet for Safkeg HS Design No. 3977A (Licensing Drawing)
0C-5941, Rev. D	Safkeg HS Design No 3977A (Licensing Drawing)
0C-5942, Rev. B	Keg Design No. 3977 (Licensing Drawing)
0C-5943, Rev. B	Cork Set for Safkeg HS (Licensing Drawing)
1C-5944, Rev. C	Containment Vessel Design No. 3978 (Licensing Drawing)
1C-5945, Rev. C	Containment Vessel Lid (Licensing Drawing)
1C-5946, Rev. D	Containment Vessel Body (Licensing Drawing)
2C-6920, Rev. A	Silicone Sponge Rubber Disc (Licensing Drawing)

The shielding inserts are constructed and assembled in accordance with Croft Associates Limited Drawing Nos:

2C-6173, Rev. D	HS-12x95-Tu Insert Design No. 3982 (Licensing Drawing)
2C-6174, Rev. D	HS-31x114-Tu Insert Design No. 3985 (Licensing Drawing)

1.3 Contents

The contents may be solids, or gases. Gaseous contents are normal form while solid contents may be either normal form or special form. The decay heat for both solid and gaseous contents may not exceed 30 watts per package. Specific radionuclides and quantity limits for each radionuclide are applied according to the insert used and the form of the radioactive material. The maximum mass of contents is based on the mass of a steel cylinder that would completely fill the cavity of the insert. Contents are limited by maximum activity on a per radionuclide basis, by maximum gross mass and net radioactive material mass, except for gaseous contents which are limited by volume. Dose rate considerations are the primary limiting factor for contents.

Approval of liquid contents was requested; however, because important supporting hydrogen generation analyses were not provided by the applicant, liquid contents are not authorized with the original issue of the certificate of compliance. Thus, further discussion of the contents in this safety evaluation report will only address solid and gaseous contents.

1.4 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the package has been adequately described to meet the requirements of 10 CFR Part 71.

2.0 STRUCTURAL

2.1 Description of Structural Design

2.1.1 Overview

The Safkeg transportation package is comprised of the 3977 keg, inner cork packing, and the 3978 containment vessel. Various inserts containing approved contents are designed to fit within the containment vessel inner cavity.

3977 keg – The keg assembly is made up of a rolled and welded austenitic stainless steel outer shell, and upper flange, a base, and a rolled austenitic stainless steel liner which contains the outer cork material. The lid is secured by way of eight closure studs and a lockpin assembly.

Inner Cork and Top Cork Packing Material – The inner cork and top cork material is placed inside the inner keg cavity to provide additional support under impact loads and also to provide some thermal insulation.

3978 Containment Vessel – The containment vessel is a right circular cylinder weldment consisting of an inner and outer shell, an alloyed DU fill, a bottom plate and a bolted removable closure.

Insert – Two inserts are authorized for this particular package; each is comprised of machined tungsten shell bodies and lids.

2.1.2 Design Criteria

2.1.2.1 Loading and Load Combinations

Loading and load combinations were developed using Regulatory Guide 7.8 and are summarized in Tables 2-1 and 2-2 of the SAR. Loadings include heat, cold, pressure, vibration, and free drop for normal conditions of transport (NCT) and free drop for hypothetical accident conditions (HAC). These loads, combined with ambient temperature, insolation, decay heat, internal pressure, and fabrication stresses, achieve a full range of loading combinations.

2.1.2.2 Acceptance Criteria

The acceptance criteria are based on the allowable stress criteria in Regulatory Guide 7.6 for primary membrane, primary membrane plus bending and primary plus secondary stresses for NCT and HAC. In the case of bearing stresses and bolt stresses, the acceptance criteria are based on the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel (ASME B&PV) Code, Section III, Division 3 (2003).

In order to extract stresses to compare against allowable stress criteria, the finite element method is used. Linearized stress intensities are obtained from critical stress locations within the containment vessel and those values are compared against the allowable stress criteria to determine a margin of safety, referred to by the applicant as the design margin. Furthermore, the adequacy of the package and packaging is confirmed by physical testing including package leak testing.

2.1.3 Weights and Centers of Gravity

Table 2-7 of the SAR summarizes several content configurations with varying weights and center of gravity locations.

2.1.4 Identification of Codes and Standards for Package Design

The codes and standards used for the design of this package and packaging are presented in Table 2-8 for major components and Important-To-Safety designations are included in the design drawings. In general, the package containment is designed to the ASME B&PV Code, Section III, Subsection NB, and the non-containment structural components to ASME B&PV Code, Section III, Subsection NF. Regulatory Guides 7.6 and 7.8 are used to develop the design criteria and load combinations, respectively as previously indicated in 2.1.2.1 and 2.1.2.2 of this safety evaluation report (SER).

2.2 Materials Evaluation

2.2.1 Overview

The Safkeg-HS 3977A package description is discussed in Section 1.2 of the application. The principal structural members of the Safkeg-HS 3977A package are as follows:

- Keg
- Cork Packing
- Containment Vessel

KEG:
The keg has a stainless steel (SS) outer shell and SS liner, both American Society for Testing and Materials (ASTM) A240, Type 304L. The shell of the keg is rolled and welded to form a cylinder. A SS base plate ASTM A240, Type 304L, SS top flange ASTM A240, Type 304L, top/bottom SS skirts ASTM A240, Type 304L, and top/bottom SS rims ASTM Type MT304 are welded to the rolled cylinder to form the keg body.

The keg is closed by a flat SS lid, ASTM A240, Type 304L which is bolted down with 8 SS studs, ASTM A276, Type 304L, and nuts ASTM A2-70, Type 304 against a single weather preventing (water) O-ring Nitrile (NBR). All closure studs are fitted with seal holes for incorporating a tamper indicating device (i.e., lock wire) in accordance with 10 CFR 71.43(b). In addition, a padlock can be attached to a SS lock pin, ASTM A276, Type 304L, welded to the keg SS closure flange, ASTM A240, Type 304L, to prevent unauthorized removal.

There is a low melting SS fuse plug, ASTM A2 alloy fitted at the center/bottom of the keg which will vent to provide over pressure relief during HAC fire conditions. This alloy has a melting point of 95°C ± 5°C.

CORK PACKING:
An inner cork (IC) liner or the SS keg liner ASTM A240, Type 304L, is fitted between the IC packing and the outer cork (OC) packing. The IC packing consists of a body and a top cork. There is no cork directly underneath the CV as it sits on the base of the keg liner. The OC packing is placed in between the SS keg liner and SS keg shell. Both the IC and the OC packing is machined from resin bonded agglomerated cork (i.e., granulated cork bound by a resin and coated in a water base varnish). The cork packing may be formed from one

agglomerated piece or from several glued together with a contact adhesive. The OC is not intended to be replaced. The IC packing is readily removable and intended to be replaced if required at pre-shipment or annual maintenance.

CONTAINMENT VESSEL:
The CV consists of a body and a removable lid assembly bolted together with 8 closure bolts and sealed with an inner and outer O-ring.

The CV body is fabricated from three pieces of solid SS as follows:

- CV flange/cavity wall
- CV outer wall
- CV base

The CV flange/cavity wall ASTM A276, Type 304L, is welded to the CV outer wall ASTM A511, Type MT304L which is welded to the SS base ASTM A240, Type 304L or A276, 304L and forms the cavity into which the CV body DU alloy shielding (2% molybdenum by weight) is placed. The DU alloy forms the shielding for the walls and base of the CV body shielding.

The CV removable lid is fabricated from two pieces of SS as follows:

- CV lid top
- CV lid shielding casing

The SS CV lid shielding casing ASTM A276, Type 304L, has the DU alloy shielding placed inside and is then welded to the SS CV lid top ASTM A276, Type 304L. The CV lid is held in position by eight recessed alloy steel screws ASTM A320, Type L43. The seal between the CV body and the CV lid is achieved by two O-ring Fluoroelastomer (Viton GLT) O-ring Rubber ASTM D2000 seals (3 mm cord diameter). Access to the interspace between the two O-rings is provided for operational and maintenance leak testing required in 10 CFR 71.51.

PAYLOAD INSERTS:
The radioactive contents are transported within payload containers or inserts placed inside the CV. Either of the two inserts specified in the CoC shall be used to provide further shielding and confinement for the contents. The two inserts are machined from tungsten. Both inserts consist of a body and a lid which are machined from a solid piece. The lid screws onto the body with an O-ring seal. Each insert has a different cavity size and provides varying levels of shielding.

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

2.2.2 Effects of Radiation on Materials

Section 2.2.3 of the application discusses the effects of radiation. The contents of the package emit one or all of alpha, beta, gamma, and neutron radiation. Austenitic stainless steel, DU and cork were chosen for the construction of the package because they are durable materials that are able to withstand the damaging effects from radiation. The Fluoroelastomer O-ring seals fitted to the containment system are the only material on which the radiation may have an effect;

however, it has been shown that for the radioactive contents the maximum dose to the containment seal is less than 1 x 10^6 rad whereas no change of physical properties of the Fluoroelastomer containment seal is expected at radiation levels up to 1 x 10^6 rad. These seals are also required to be replaced annually at maintenance.

Staff finds that radiation effect on the elastomeric seals is not a concern as the seals are replaced on an annual basis. In addition, finite element analysis predicts the maximum temperature of the elastomeric seal during accident conditions reached is 116°C which is below the acceptable stated heat resistance for Fluoroelastomer rubber.

2.2.3 Brittle Fracture

Section 2.1.2.5 of the application discusses material brittle fracture concerns. All the structural components of the package are fabricated from austenitic SS which is ductile at low temperatures. Regulatory Guide 7.11 states austenitic SS is not susceptible to brittle facture at temperatures encountered in transport. The HAC drop tests have been conducted at minus 40°C to determine if brittle fracture has any effect on the package, with compliance demonstrated if the containment vessel is undamaged and leak tight on completion of testing.

The staff finds that by avoiding the use of ferritic steels brittle fracture concerns are precluded. Specifically, most primary structural packaging components are fabricated of Type 304L SS. Since this material does not undergo a ductile-to-brittle transition in the temperature range of interest (down to minus 40ºC), it is safe from brittle fracture. The staff states that in austenitic SS metal the force required to move dislocations is not strongly temperature dependent and dislocation movement remains high (i.e., will deform more readily under load before breaking) even at low temperatures and the material remains relatively ductile.

2.2.4 Acceptance Tests and Maintenance Program

Section 8 of the application discusses acceptance tests and maintenance program. The metallic materials of construction (keg and CV) are procured and fabricated to consensus industry standards. The various structural components are fabricated from ASTM standard materials in accordance with American Society of Mechanical Engineers Sections III, V, and IX. A summary of maintenance requirements is discussed in Table 8-1 of the application. The maintenance program includes periodic testing, inspection, and replacement schedules.

The staff finds that visual inspections at various timed intervals provide additional reasonable assurance against corrosion occurring unnoticed.

2.3 General Standards for All Packages

2.3.1 Minimum Package Size

The smallest overall dimension exceeds the specified requirement of 4 inches; therefore, the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.3.2 Tamper Indicating Feature

Each closure stud includes a milled hole which allows a wire security seal to be affixed to the studs. Furthermore, the existing lockpin is designed to accept a padlock; therefore, the requirements of 10 CFR 71.43(b) are satisfied.

2.3.3 Positive Closure

Positive closure is demonstrated by using two bolted closure lids, one on the outer keg and one on the containment vessel. The package is adequately analyzed for maximum internal and external differential pressures as well as expected external and internal pressures during NCT and HAC. Thus, the requirements of 10 CFR 71.43(c) are satisfied.

2.3.4 Chemical and Galvanic Reactions

Section 2.2.2 of the application discusses reactions due to chemical, galvanic, or other reactions. The applicant states that the package has been evaluated for all the material interactions of chemically or galvanic dissimilar materials and there is no potential for chemical, galvanic, or other reactions between the components of the package which are SS and cork in dry conditions, and SS and encapsulated DU alloy which is sealed and therefore dry.

The staff concludes that, during normal conditions of transportation, the CV internals will not be subject to continuous or frequent exposure to moisture or that any water intrusion is not likely to occur in great quantities. The number of and galvanic potential between the different metals used in fabrication is low. Therefore, the conditions required to create the possibility for galvanic corrosion is small. Further, visual inspections to be performed of the Important-to-Safety CV payload cavity at various timed intervals provide reasonable assurance against any significant corrosion occurring unnoticed.

2.4 Lifting and Tie-down Standards for All Packages

2.4.1 Lifting Devices

The package does not incorporate any structural feature that is used as a lifting device. The requirements of 10 CFR 71.45(a)(1) for lifting devices are met.

2.4.2 Tie-Down Devices

The package does not incorporate any structural feature that is used as a tie-down device. The requirements of 10 CFR 71.45(b)(1) for lifting devices are met.

2.5 Normal Conditions of Transport

2.5.1 Heat

2.5.1.1 Pressures and Temperatures

The maximum calculated temperatures for the containment vessel, including a 30W maximum heat load, are illustrated in Table 3-2. The maximum reported temperature is approximately 163°C; however stress calculations are performed using 158°C. Given that the maximum gauge pressure of 700kPa, assumed in the stress calculations, exceeds the calculated maximum pressure of 180kPa, the discrepancy of the maximum temperature used for the purposes of structural performance is inconsequential.

2.5.1.2 Differential Thermal Expansion

Differential thermal expansion is evaluated with a finite element model. Interference between the stainless steel shell and the DU shielding is evaluated as well as interference between the containment vessel, cork, and the keg shell.

Given the gaps present between the containment vessel and cork, as well as between the cork and keg lid, differential thermal expansion did not challenge the structural integrity of the package.

The analysis did however show some induced stresses due to interference between the containment vessel shell and DU shielding. The stress calculations and comparison with allowable stresses are discussed below.

2.5.1.3 Stress Calculations and Comparison with Allowable Stresses

The finite element model applied a uniform temperature of 158°C across the containment vessel which contained an internal pressure of 700kPa. Linearized stress intensities at various section cuts are reported in Tables 2-16 and 2-17 of the SAR and compared with allowable stresses. All cases showed a margin greater than zero.

Staff notes that in one case, a design margin of 0.04 is reported. As indicated above, the use of a 700kPa operating pressure with a design capacity greater than 1000kPa and a calculated actual pressure of 180kPa indicates significant conservatism in the calculation method.

Additionally, containment vessel buckling and fatigue are evaluated and compared with requisite acceptance criteria.

The requirements of 10 CFR 71.71(c)(1) are satisfied.

2.5.2 Cold

The applicant evaluates the package for the cold condition (-40°C) including the determination of maximum stresses in the containment vessel as reported in Table 2-20. In all cases, the stress results, when compared with allowable stress, demonstrate positive margins. In addition, further evaluations are completed for bolts, gross buckling, and thermal cycle stress ratcheting. In all of these cases, the analytical results demonstrate that the package is sufficiently robust in a cold environment.

The requirements of 10 CFR 71.71(c)(2) are satisfied.

2.5.3 Reduced External Pressure

The applicant evaluated the package for the reduced external pressure including the determination of maximum stresses in the containment vessel as reported in Table 2-24. In all cases, the stress results, when compared with allowable stress, demonstrate positive margins. In addition, further evaluations are completed for bolts, gross buckling, and fatigue. In all of these cases, the analytical results demonstrate that the package is sufficiently robust in a cold environment.

The requirements of 10 CFR 71.71(c)(3) are satisfied.

2.5.4 Increased External Pressure

The applicant evaluated the package for increased external pressure including the determination of maximum stresses in the containment vessel as reported in Table 2-24. In all cases, the stress results, when compared with allowable stress, demonstrate positive margins. In addition, further evaluations are completed for bolts, gross buckling, and fatigue. In all of these cases, the analytical results demonstrate that the package is sufficiently robust in a cold environment.

The requirements of 10 CFR 71.71(c)(4) are satisfied.

2.5.5 Vibration

The applicant used the peak G load expected during NCT due to vibration as the means to evaluate the performance of the package under this requirement. The stress analysis performed for this evaluation demonstrates that the package would not be adversely affected due to vibratory loads. Furthermore, the fatigue evaluations performed for heat, cold, and pressure demonstrate in aggregate that the package is sufficiently robust to withstand all direct and indirect vibration effects.

The requirements of 10 CFR 71.71(c)(5) are satisfied.

2.5.6 Water Spray

Due to the materials of construction, including seals, the staff has determined that water spray is not a significant challenge to the structural design of this package.

The intent of 10 CFR 71.71(c)(6) is satisfied.

2.5.7 Free Drop

The keg and containment vessel were subjected to a 1.2 meter drop test for side, top corner, and top end orientations. Furthermore, the three drop orientations were modeled using the finite element method under both hot and cold conditions to determine the stresses in the containment vessel. The stress results and margins of safety are illustrated in Tables 2-31, 2-32, and 2-33.

Staff notes that the stresses reported by the applicant in the containment vessel show negative margins of safety for several load and stress combinations. The explanation provided by the applicant for these negative margins is that the material model (modulus of elasticity) for cork was modified to allow for the numerical simulation to converge. This is a common artifact of quasi-static finite element analysis programs when attempting to model failure in a structural material. The technique used by the applicant to overcome this numerical instability was to artificially increase the modulus by a given factor such that failure of the material is not initiated. In this case, the modulus is increased by three orders of magnitude, thereby creating a significantly worse impact environment for the keg and the containment vessel. In general, this is a conservative approach; however, given that this is not considered a best practice and there is no reasonable way to determine if increasing the modulus by a factor of 1000 is sufficiently conservative, the analysis must be partially set aside in making the safety determination in the cases where negative margin exists. In order to demonstrate that this approach is reasonable

for this particular configuration, the applicant would, at a minimum, need to perform a sensitivity study on the cork modulus in order to draw conclusions regarding the margin of safety. Staff therefore does not fully accept the results of the finite element methodology used by the applicant as a proxy for the structural performance of the package and containment vessel. In regions where there is a positive margin of safety, it is reasonable to utilize the results because of the conservative nature of the analysis, but only from the perspective of defense in depth and not as a primary means of demonstrating structural performance.

The staff does have reasonable assurance from the physical drop tests, informed by the analytical approach, that the package and containment vessel are capable of satisfactory structural performance.

The drop testing and analytical modeling in aggregate satisfies the requirements of 10 CFR 71.71(c)(7).

2.5.8 Corner Drop

The corner drop is not applicable to this package design. The requirements of 10 CFR71.71(c)(8) are satisfied.

2.5.9 Compression

The compression drop test is performed and the package exhibited no permanent change in external dimensions. The requirements of 10 CFR 71.71(c)(9) are satisfied.

2.5.10 Penetration

The applicant subjected the package to a penetration test which consisted of dropping a 6 kg steel bar 3 cm in diameter from a height of 1 meter. The reported damage to the outer keg consisted of a dent approximately 8 mm in depth. As such, this dent is not sufficient in size to challenge the functional characteristics of the package.

The requirements of 10 CFR71.71(c)(8) are satisfied.

2.6 Hypothetical Accident Conditions

2.6.1 9-meter Free Drop

The keg and containment vessel are subjected to a 9 meter drop test for end, side, and corner drop orientations. Furthermore, the three drop orientations are modeled using the finite element method under both hot and cold conditions to determine the stresses in the containment vessel. The stress results and margins of safety are illustrated in Tables 2-34 to 2-42.

Staff notes that the stresses reported by the applicant in the containment vessel show negative margins of safety for two load and stress combinations. As with the NCT drop condition, the explanation provided by the applicant for these negative margins is that the material model (modulus of elasticity) for cork is increased by three orders of magnitude to allow for the numerical simulation to converge. As with the NCT evaluations, this is a conservative approach; however, given that this is not considered a best practice and there is no reasonable way to determine if increasing the modulus by a factor of 1000 is sufficiently conservative, the analysis must be partially set aside in making the safety determination. In order to demonstrate

that this approach is reasonable for this particular configuration, the applicant would, at a minimum, need to perform a sensitivity study on the cork modulus in order to draw conclusions regarding the margin of safety. Staff therefore does not fully accept the results of the finite element methodology used by the applicant as a proxy for the structural performance of the package and containment vessel. In regions where there is a positive margin of safety, it is reasonable to utilize the results because of the conservative nature of the analysis, but only from the perspective of defense in depth and not as a primary means of demonstrating structural performance.

Staff does however have reasonable assurance from the physical drop tests, informed by the analytical approach, that the package and containment vessel are capable of satisfactory structural performance.

The drop testing and analytical modeling in aggregate satisfies the requirements of 10 CFR 71.73(c)(1).

2.6.2 Crush

This evaluation is not applicable due to the package density exceeding the minimum allowable. The requirements of 10 CFR 71.73(c)(2) are met.

2.6.3 Puncture

The applicant subjects the package to a penetration drop test which consists of dropping the package from a height of 1 meter onto a fixed steel puncture bar. The package is dropped from three different orientations including side, top rim, and top of the outer keg. The reported damage to the outer keg consists of a dent approximately 11 mm in depth. As such, this dent is not sufficient in size to challenge the functional characteristics of the package and no penetration of the keg is observed.

The requirements of 10 CFR 71.73(c)(3) are met.

2.6.4 Thermal

2.6.4.1 Summary of Temperature and Pressures

The applicant reports that the containment vessel reaches a maximum temperature of 208°C during the fully engulfing fire including a heat load of 30W from the contents. This results in a calculated internal pressure of 164kPa which is an order of magnitude lower than the 1000kPa (gauge) assumed for design.

2.6.4.2 Differential Thermal Expansion

The applicant notes no significant thermal gradient over the uranium shielding and stainless steel; therefore the NCT heat conditions are considered bounding.

2.6.4.3 Stress Calculations

The applicant notes that there is no significant thermal gradient over the containment vessel; therefore the NCT heat conditions are considered bounding.

2.6.4.4 Comparison with Allowable Stresses

As discussed, NCT heat condition thermal stresses are bounding.

The requirements of 10 CFR 71.73(c)(4) are met.

2.6.5 Immersion – Fissile

Although the applicant originally sought authorization to transport fissile material in this package, this content type has been withdrawn from consideration. The requirements of 10 CFR 71.73(c)(5) are met.

2.6.6 Immersion - All Packages

The applicant demonstrates through a scaled calculation (increased external pressure) that the margins of safety remain positive for an equivalent pressure of 150kPa. The requirements of 10 CFR 71.73(c)(6) are met.

2.6.7 Deep Water Immersion Test -Special Requirements for Type B Packages Containing More Than 10^5 A_2

The contents have less than 10^5 A_2; therefore this test is not applicable.

2.7 Evaluation Findings

Based on a review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated, meets the regulatory requirements for mitigating galvanic or chemical reactions, is unaffected by cold temperatures, is constructed with materials and processes in accordance with acceptable industry codes and standards, and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL

The staff reviewed the Safkeg-HS 3977A (Safkeg) transportation package application to verify that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC and that the package design satisfies the thermal requirements of 10 CFR Part 71. The application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," as well as associated Interim Staff Guidance (ISG) documents.

3.1 Description of Thermal Design

Design features that are significant with respect to heat transfer in the Safkeg package are the stainless steel keg outer skin, the stainless steel keg inner liner, the outer cork, top cork and inner cork, the stainless steel containment vessel, and the DU shielding in the containment vessel. Under a HAC fire event, the keg skin is designed to heat up very quickly with the cork providing insulation to the containment vessel due to its low thermal conductivity and ablation properties. Since heating of the cork during the HAC fire causes gas production within the keg cavity, a fuse plug is provided in the bottom of the keg. On heating above 95°C the fuse plug melts allowing pressure relief of the keg cavity.

The staff finds the Safkeg package thermal design description acceptable.

3.2 Material Properties and Component Specifications

Material property tables for the Safkeg components are included in SAR Section 3.2. Materials present in the package include 304L stainless steel, DU, and cork. Thermal properties provided in the SAR include thermal conductivity, density, heat capacity, and emissivity. Except for cork, the temperature range for the material properties covers the range of temperatures encountered during the thermal analysis.

During a fire, the cork experiences temperatures up to 800°C. However, the applicant stated that no measurements of cork properties at high temperatures are available. Therefore, the applicant performed a HAC thermal test using the Safkeg-LS 3979A package. This package uses the same cork specified for the Safkeg-HS 3977A package. A thermal computer model of the Safkeg-LS 3979A package simulated the HAC test results both to validate the model and to demonstrate the acceptability of the thermal properties assumed for the cork. It is found that, in order to obtain agreement with the measured temperatures, the thermal conductivity of the cork needed to be increased by 50%. The applicant stated that these thermal properties, validated against the furnace test, are “effective” properties that include any effects of charring and shrinkage of the cork. The applicant also performed a NCT thermal test using a Safkeg-HS 3977A package. The applicant simulated this test using the Abaqus computer model described in Section 3.3. To produce the best agreement with the measured temperatures for the NCT thermal test, the thermal conductivity of the cork needed to be reduced by 15%. Because cork is a natural material, this degree of variation in conductivity may well be possible. To ensure that all the calculations performed with the model are conservative, the lower, fitted conductivity is used for temperature calculations during NCT and the higher, measured thermal conductivity is used for the temperature calculations during HAC.

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

3.3 Description of Safkeg Thermal Model

To perform the thermal evaluation of the Safkeg-HS package, the applicant developed an axisymmetric model. The model explicitly represented the keg outer skin, the keg inner liner, the outer cork (the cork between the inner liner and outer skin), the top cork within the keg cavity, the inner cork within the keg cavity, the CV, and the DU shielding in the inner containment vessel. The Abaqus code is used to build the thermal model. The model contained 7841 nodes and 5022 elements. Each of the components listed above is generated separately and thermally joined using either tied constraints or interactions (representing narrow air gaps). The thin outer skin of the keg is modeled using “shell” elements while all the other components are modeled using solid elements.

The applicant validated the thermal model by comparing against both an experimental self-heating test performed on the 3977A package (simulating NCT), and a furnace test (simulating HAC) performed on a Safkeg-LS 3979A package.

The staff finds the Safkeg-HS thermal model description acceptable.

3.4 Thermal Evaluation under Normal Conditions of Transport

3.4.1 Heat and Cold

The applicant used the thermal model described earlier to perform the package thermal evaluation under NCT and subject to solar insolation. Figure 3-3 of the SAR showed the transient temperature at various locations on the outer surface of the keg with a 30W heat load. The highest temperatures occurred on the top of the container because the insolation flux is greater on the top than on the side. The maximum predicted temperature, which occurred on the top of the package, is 102°C. Figure 3-4 of the SAR showed the transient temperature at the inner containment vessel lid seal. It can be seen that the maximum temperature is reached after 1-½ days. The maximum seal temperature is predicted to be 151°C. Figure 3-5 of the SAR shows the maximum temperatures throughout the package under NCT. The peak temperatures experienced during NCT conditions with insolation are shown in Table 3-2 of the SAR along with the allowable maximum temperatures for each component listed. All predicted temperatures are below the allowable limit. As shown in Table 3-2, the maximum temperature of the accessible surface is 42°C which is reached on the keg lid. The base of the keg reaches 45°C. However, this surface is not accessible; therefore, it is not considered. This demonstrates that the package is capable of fulfilling the requirements of 71.43(g) as the accessible surface temperature is less than 50°C with a maximum heat load of 30W.

For the NCT cold evaluation the applicant assumed that the package is in an ambient of -40°C, with zero insolation and zero heat decay. No analysis is performed because the applicant assumed that the package and all the components will eventually reach thermal equilibrium at -40°C. The applicant states that this temperature is within the allowable service limits for all the components.

3.4.2 Maximum Normal Operating Pressure

The applicant calculated a maximum normal operating pressure of 800 kPa based on an assumed maximum temperature of 160°C.

3.4.3 Maximum Thermal Stresses

The maximum thermal stresses during NCT event are addressed in Chapter 2 of the SAR.

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

3.5 Thermal Evaluation under Hypothetical Accident Conditions

3.5.1 Initial Conditions

The applicant took the initial conditions for the thermal model HAC fire test from the NCT analysis at the end of a 12-hour period of insolation with a content decay heat of 30W. All temperatures used as initial conditions are at their maximum values, per NCT analysis. The applicant stated that the package experienced minimal damage after the penetration and drop tests and that the damage did not affect the package thermal performance.

3.5.2 Fire Test Conditions

The applicant used the finite element model described earlier to determine the temperature of the container during the fire accident specified in 10 CFR Part 71. A 30 minute, 800°C fire is simulated followed by a 12 hour cooling period. The thermal model is modified as described below.

A transient calculation is made with heating from a constant 800°C fire for 30 minutes followed by cooling for 12 hours. The calculation started from the temperature profile obtained for NCT with insolation. No solar insolation is applied in the transient calculation during the heating phase of the fire test; however, insolation is applied during the 12 hour cooling phase following the fire. During the heating phase of the fire test, all exterior surfaces of the keg are assumed to receive heat by forced convection and radiation from the fire. A heat transfer coefficient of 15 W/m^2-K is assumed (the value suggested in the "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material," 2005 Edition, IAEA Safety Guide No. TS-G-1.1 (ST2) 2002). The absorptivity of the surface of the keg is assumed to be 0.8, which is the value specified in 10 CFR Part 71. During the cooling phase, heat is modeled as being lost from all exterior surfaces of the keg by radiation and natural convection. The emissivity of the surface of the keg is assumed to remain at 0.8, the value used during the heating phase. Established correlations for natural convection are again used to derive the appropriate convection coefficient, as described in the SAR.

3.5.3 Maximum Temperatures and Pressures

The maximum temperatures for the Safkeg package components in a HAC fire event with an ambient temperature of 38°C and insolation, as calculated by the applicant, are provided in Table 3-3 of the SAR. All the temperatures predicted during the HAC are below the allowable limit. The applicant calculated a maximum design pressure of 11 MPa based on an assumed maximum temperature of 200°C during HAC. The applicant calculated a maximum pressure within the containment vessel of 0.4 MPa which is well within the design envelope of 11 MPa.

3.5.4 Maximum Thermal Stresses

The maximum thermal stresses during HAC fire event are addressed in Chapter 2 of the SAR.

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

3.6 Evaluation Findings

Based on a review of the statements and representations in the application, the staff concludes that the Safkeg-HS 3977A transportation package thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT

4.1 Description of the Containment System

The containment boundary of the SAFEKEG-HS 3977A package is formed from the CV flange/cavity wall, lid top and containment seal O-ring, as shown in Figure 4-1 in the SAR. The containment seal O-ring, which is fitted in a face seal configuration with the O-ring recessed into the flange, seal the lid top to the flange/cavity wall. The lid top is held in position with 8 alloy steel closure screws, which screw into the containment vessel flange/cavity wall and lid and are tightened to a torque of 10±0.5 Nm. On tightening the closure screws, a uniform and repeatable compression of the O-rings is provided.

The closure screws are recessed into the lid top to physically protect them from damage. There is also a shear lip in the lid top and flange protecting the screws from shear failure due to transverse impact loads. The closure screws are positive fasteners, that cannot be opened unintentionally, or by any pressure that may arise within the package.

There are no welds, valves, or pressure relief devices present in the containment boundary and the package does not rely on any filter or mechanical cooling system to meet the containment requirements.

The containment system is designed, fabricated, examined, tested, and inspected in accordance with ASME B&PV Code Section III, Subsection NB. The complete specifications such as closure screw torques, materials of construction, O-ring specifications, and design dimensions for the containment system are given in CoC Drawing Nos. 1C-5944, 1C-5945, and 1C-5946.

The flange/cavity wall and lid top are machined from 304L solid stainless steel. The containment O-ring is manufactured from silicone. The materials of construction of the containment system are evaluated in Section 2.2.2 of the SAR. All the materials have been selected for compatibility with each other, the inserts, and the payload in order to avoid chemical, galvanic, or other reactions.

Viton GLT is selected as the containment O-ring material because it offers a temperature range of -40°C to 205°C. The radiation dose to the containment seal, assuming that the package is loaded with maximum contents as specified in Section 1.2.2 of the SAR for a full year, is estimated to be much less than 1×10^4 Gray (1×10^6 rad). This estimate is based on the dose rate data presented in Section 5.5.4.1.1 of the SAR for cesium-137 (Cs-137) contents. It is judged that Cs-137 would produce the highest dose rate to the containment seal (which is outside the shielding) as it has a penetrating radiation. The maximum dose rate at the containment seal for each of the two inserts authorized for use in the Safkeg-HS, for the maximum Cs-137 contents and as limited by the package maximum allowable surface dose rate, is given in Table 4-1 in the SAR. The Parker Handbook reported the impact on all elastomers is minor at radiation levels up to 1×10^6 rad.

Consequently, staff concluded that the containment O-ring seal will not be unduly affected by the radiation from the contents of the package. Staff also noted that the containment O-ring seal is required to be replaced either during the periodic maintenance activities found in Section 8.2 of the SAR or after a maximum period of one year.

Figure 4-2 of the SAR shows the two additional O-ring seals fitted to the CV: a test point seal and a test seal. These seals are present to facilitate the leak test of the containment seal during the pre-shipment leak test. The test point is a tapped hole that allows connection of a pressure drop leak tester to the interspace volume between the test seal and the containment seal. The test seal is located close to the containment seal to provide a small interspace volume thus increasing the sensitivity of the pressure rise leakage test. The inserts, specified in the CoC, are also fitted with an O-ring seal. The test point seal, the test seal and the insert seals are not relied upon for containment.

4.2 Containment under Normal Conditions of Transport

The Safkeg-HS 3977A has been designed specifically to meet the criteria for leak tightness during NCT. The CV containment boundary is tested to demonstrate it is leaktight during the testing, fabrication, and maintenance of the packaging. The maximum internal pressure of the CV under NCT is taken as the design pressure of 800 kPa or 8.0 bar absolute. This value is calculated and found in Section 3.3.2 of the SAR. The contents are carried within the inserts specified in the CoC. These inserts are required for all contents. Under NCT, the shielding inserts, together with the user defined product containers, provide confinement of the radioactive material (solid, or gas). However, the CV containment seal provides containment.

Prototype testing and analysis has demonstrated the structural performance of the CV containment boundary. A prototype Safkeg-HS 3977A package was subjected to the NCT and HAC tests, as reported in Sections 2.6 and 2.7 of the SAR, in an uninterrupted test series; the containment seals are shown to be leaktight on conclusion of the tests. The structural analysis showed that there would be no permanent deformation of any of the containment system components under NCT conditions. The structural analysis also showed that there would be no permanent deformation of any of the containment system components under NCT; therefore, there would be no effect which could cause any reduction in the effectiveness of the containment system.

4.3 Containment under Hypothetical Accident Conditions

The thermal evaluation in Section 3.4 in the SAR demonstrates that the seals, bolts, and containment system materials do not exceed their temperature limits under HAC. The testing and analysis reported in Section 2 of the SAR show that the containment system would be unaffected by HAC and provide complete containment for all contents. The containment system has been shown to be unaffected by HAC and the seals are within their working temperature for the short duration high temperatures associated with the HAC thermal test. Therefore, it is concluded that the containment system meets the requirement for providing containment of the solid radioactive contents, within the allowable leakage rate limits, for HAC. For gaseous radioactive material, under HAC the gas is assumed to leak from both the shielding inserts and the user defined product containers into the cavity of the CV. The gas is assumed to leak from the CV containment seal at the leakage rate to which the containment is proved (i.e., 1×10^{-7} ref.cm^3/s or leaktight).

Containment of gases is based upon the assumption that the closure of the containment system (i.e., the containment seal and the CV lid and top flange) would leak at the leaktight rate of 1×10^{-7} ref.cm^3/s.

The maximum amount of the radioactive gases that may be carried has been calculated based upon the allowable leakage rate limits specified in 10 CFR Part 71 and the assumed leak rate

from the containment seals of 1 x 10^{-7} ref.cm^3/s. The calculation of the size of a single leak having a leakage rate of 1 x 10^{-7} ref.cm^3/s is given in report CS 2012/04. The calculated hole diameter, for a single leak path in the 3 mm O-ring, with a hole length of 0.26 cm, is 1.1 x 10^{-4} cm. The gas leakage rates (in terms of mass flow and A_2/hr and A_2/week) are given in report CS 2012/05. The allowable leakage rates under HAC are taken as no escape of Kr-85 exceeding 10 A_2, and no escape of other radioactive material exceeding a total amount of A_2 in a week, as given in 10 CFR 71.51(a)(2).

4.4 Leakage Rate Tests for Type B Packages

The materials and components used to manufacture the containment boundary are required to be helium leak tested during fabrication with a pass leakage rate of 1 x 10^{-7} ref.cm^3/s. These tests ensure the fabricated components and the materials used meet the required level of containment prior to the approval of the package for use.

The requirements for the fabrication leak rate test are specified in Section 8.1.4 of the SAR. The confinement capability of the inserts is assured by the requirements specified in Section 8.1.4.

If any maintenance activities are undertaken on the containment boundary, a helium leak rate test is required to confirm that any repairs or replacements have not degraded the containment system performance. The required leak rate has a pass leakage rate of 1 x 10^{-7} ref.cm^3/s. The requirements for the maintenance leak rate test are specified in Section 8.2.2 of the SAR. The confinement capability of the inserts is assured by the requirements specified in the same section.

A periodic helium leak rate test is required to be carried out annually with a pass leakage rate of 1 x 10^{-7} ref.cm^3/s. This test confirms that the containment boundary capabilities have not deteriorated over an extended period.

The requirements for the periodic leak rate test are specified in Section 8.2.2 of the SAR. The confinement capability of the inserts is assured by the requirements specified in the same section.

Prior to shipment, each package is required to be leak rate tested using the gas pressure rise or gas pressure drop method, with a pass leakage rate of 1 x 10^{-3} ref.cm^3/s. This test confirms the CV is correctly assembled prior to shipment.

The requirements for the pre-shipment leak rate test are specified in Section 7.1.3 of the SAR. The confinement capability of the inserts is assured by the requirements specified in Section 7.1.2 of the SAR.

4.5 Evaluation Findings

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

5.0 SHIELDING

The purpose of the shielding review is to verify that the package design meets the external radiation requirements of 10 CFR Part 71 for NCT and HAC. The applicant has proposed a new package design designated the Safkeg-HS. The proposed contents include specific nuclides up to the specified quantities that are derived from consideration of dose rate, mass, and heat limitations. The quantities also account for decay of the proposed nuclides and the contribution to both dose rates and decay heat from the resulting buildup of progeny in the contents. The applicant's shielding evaluation method is to derive contents quantities from calculations of quantities that result in dose rates at the most limiting regulatory limit(s) at the most limiting location(s), considering dose rate limits on the package surface and at 1 meter from the package for NCT and HAC for non-exclusive use, with the quantities derived from the dose rate analyses adjusted as described later in this chapter of the SER. The staff reviewed the submittal using the guidance in Section 5 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

5.1 Description of the Shielding Design

5.1.1 Design Features

The staff reviewed the SAR general information chapter (Chapter 1) in the application, the proposed CoC drawings, the model drawings, and the design information in SAR Chapter 5 of the application. The staff also considered relevant information in the attachments to the application. The shielding design features of the Safkeg-HS include three major components: the insert, the CV, and the outer packaging that is referred to as the keg. The dimensions for the inserts being authorized for use with the Safkeg-HS are specified in the proposed CoC Drawing Nos. 2C-6173 and 2C-6174.

The contents are placed in the inserts, which are placed inside the CV cavity. The inserts are sealed by screwing the insert lid tight and aligning the match mark on the insert lid with the match mark on the insert body. A rubber disc (CoC Drawing No. 2C-6920), which the applicant incorporated into the design based upon a staff information request, is placed on top of the insert in the CV cavity. The purpose of the disc is to prevent movement of the insert and loosening of the insert lid due to phenomena such as vibration. The applicant demonstrated with a vibration test that the manner of insert closure and use of the rubber disc are adequate to ensure the insert retains the contents. In addition, the inserts will be leak rate tested as part of the acceptance testing and maintenance programs to ensure they meet the specified leak rate criterion. This is important because the applicant's shielding analysis relies on the inserts retaining the radioactive contents. Based on the vibration test results, the acceptance testing and maintenance programs, and the design features to ensure proper closure of the insert during package operations and that the closure is maintained during transport, the staff finds the assumption that the contents remain in the insert to be acceptable.

The CV shielding is provided by the CV steel and DU shielding, with the dimensions and specifications in the proposed CoC Drawing Nos. 1C-5944, 1C-5945, and 1C-5946. The CV has DU in both the CV body and the CV lid. The DU thickness is of relatively uniform thickness for most of the CV body and lid (45.7 to 47.65 mm), the DU thickness is significantly reduced in the top portion of the CV body to 22.25 mm. The CV lid has a ring of extra steel on its top to provide additional shielding above any gaps that may exist between the CV body and the portion of the lid that fits into the body. There is also a raised portion of steel on the top of the

lid at the radial center. However, there is a hole where the steel is significantly reduced, that can be an area for increased radiation levels. To address this, the applicant has added a shielding screw to the package design and operations. The shielding screw is present except when using the hole as a means to lift the CV or CV lid. Thus, the screw will be part of the package as presented for transport. With the screw in place the staff finds the applicant's analysis is acceptable for addressing dose rates for the package's axial top. The CV is placed inside the keg.

The keg is composed of steel and cork material that is placed between the keg outer shell and cavity liner. When the CV is placed inside the keg, it is inserted into an inner cork and then a top cork is placed over the CV. This cork fits within the keg cavity. A steel lid encloses the keg cavity. The cavity liner and the outer shell are steel. The space between the cavity liner and the outer shell contains a cork component referred to as the outer cork. For the purposes of the shielding evaluation, the cork is only credited with maintaining the position of the CV and the dimensions of the package. The cork material is neglected. Proposed CoC Drawing Nos. 0C-5941, 0C-5942, and 0C-5943 show the keg components and their dimensions.

5.1.2 Maximum Radiation Levels

Table 5-1 of the application summarizes the maximum radiation levels for the package under NCT, and Table 5-2 of the application summarizes the maximum radiation levels for the package under HAC. The NCT limits are for exclusive use shipments in a closed vehicle. As discussed later in this SER, the proposed contents' activity limits are derived from, or back-calculated from, the regulatory dose rate limits for non-exclusive use shipments and from package thermal limits and mass limits. The more restrictive activity is the proposed contents activity limit. Thus, some contents activity limits are based on the regulatory dose rate limits. Because of the method for determining the contents activity limits, which includes assuming the contents are a point source, the package surface dose rate limit is the most restrictive dose rate limit. Thus, the maximum NCT dose rate at the package surface is the limit of 200 mrem/hr (2 mSv/hr) for purposes of defining the dose rate-based maximum contents activity.

This analysis assumes, however, that the package is in the as-fabricated condition at nominal dimensions and does not account for uncertainties in the analysis associated with dimensional and material tolerances, package damage due to NCT and HAC tests, and the dose conversion factors (DCFs) used. Accounting for these factors, the calculated dose rates exceed the non-exclusive use limits. Therefore, the applicant demonstrates that the package dose rates meet the NCT limits for exclusive use in a closed vehicle. The analysis assumes the package is in a closed vehicle, with the package secured in a position that is at least 20 cm from any vehicle surface. The analysis accounts for most of these factors and also accounts for the impact of the source location, as described in the following sections. For some contents, the activity limits were adjusted. The applicant also included an additional margin by ensuring the calculated package surface dose rate limits did not exceed 800 mrem/hr (8 mSv/hr).

5.2 Source Specification

Tables 1-3-1, 1-3-2, 1-3-6, 1-4-1, 1-4-2, and 1-4-6 of the application describe the proposed contents, including the proposed activity limits. The application defines the proposed contents as specific radionuclides for three proposed contents types in the last three aforementioned tables, which also give the activity, or quantity, limits for each radionuclide. The limits account for the contribution to both dose rates and to decay heat resulting from the buildup of the proposed contents' decay progeny with time as well. Two content types are solid material as

either special form or normal form. Each of these content types is specific to one of the two inserts. One content type is for gaseous material that is loaded into the thinner tungsten insert.

As discussed later in this SER, the proposed activity limits are derived from the regulatory dose rate limits and the package thermal and mass limits. For the shielding evaluation, the applicant calculated the activity for each nuclide in each content type that meets the more restrictive regulatory dose rate limits for non-exclusive use shipments under both NCT and HAC. Radioactive progeny build up in the contents for several of the proposed radionuclides. This buildup occurs quickly for those nuclides with short half-lives. The applicant's evaluation includes these progeny as part of the radiation source associated with each nuclide as described in Section 5.4.1 of this SER. The specific source is the proposed nuclide and its progeny at the decay time that maximizes the dose rates. These dose rates are used to determine the activity limits for the parent nuclides that are the proposed contents as explained later in this SER chapter.

The proposed contents and their progeny include gamma-, neutron-, and beta-emitting nuclides. The applicant did not perform neutron dose rate calculations for the proposed contents, or their progeny, for the two tungsten inserts. The staff evaluated the proposed contents for consideration of neutron sources. While the decay of some nuclides includes spontaneous fission, the rates are quite small and the resulting neutron sources are negligible. The staff also considered the potential for alpha-n reactions in the contents, including the possible materials in which the radionuclides might be mixed. For example, americium with beryllium or boron can be a very good neutron source. Thus, since this source was not considered in the application, the staff is conditioning the CoC to prohibit beryllium and boron in the materials which are loaded as part of the proposed contents.

The proposed contents and their progeny include beta-emitting nuclides. The applicant used a conservative method to address the contribution to dose rates from bremsstrahlung resulting from these nuclides. The calculation method is based on the method described in Chapter 5 of Cember's *Introduction to Health Physics* text [1]. As part of this method, an appropriate material must be identified as the beta absorber. The applicant selected the insert material as the absorber based on the thickness of the absorber. The applicant demonstrated that this is appropriate because the range of beta particles with energies up to 2.28 MeV is smaller than the insert's minimum thickness. The staff considered a beta energy of 3.3 MeV, considering the proposed contents and the progeny with sufficient beta emission rates. The staff's evaluation indicates that even for betas at this energy, the use of tungsten as the absorber is appropriate.

The applicant's approach to calculating the bremsstrahlung dose rates is described in CTR 2011/01, Issue D, "Safkeg-HS 3977A: Package Activity Limits Based on Shielding." The applicant's method only considers betas with maximum energies above 0.8 MeV and provides a basis for this threshold. The method also considers all betas above this threshold that are emitted by a nuclide and their emission intensities. The applicant added the resulting dose rates to the gamma dose rates from the proposed nuclides and their progeny in its analyses. The approach is somewhat different for the proposed contents and their progeny that are pure beta emitters in that there is no energy threshold below which betas were not considered for these radionuclides. Based on the proposed nuclides and the activity limits along with the conservative nature of the method, the staff finds the method to be acceptable.

Based on its review of the application's source descriptions, its own independent evaluations of the proposed radionuclide contents' decay schemes and resulting progeny as well as the

radiation types requiring analysis, as described above, the staff finds that the application adequately captures the radiation sources for the proposed contents.

5.3 Shielding Model

5.3.1 Source and Shielding Configuration

The applicant uses two shielding methods in its evaluation. The models for each method are different. The first method uses the McBend code. The second method uses the MicroShield code. The source is modeled as a point source for each method. No credit is taken for material properties of the source that could provide self-shielding.

For the calculations with the McBend code, three-dimensional models of the package and the tungsten inserts were created. There is also a model for a package with no insert. The application includes model drawings in SAR Chapter 1 which show nominal dimensions for the packaging components. The models used these nominal dimensions. The models include the gaps between components, such as between the CV lid and CV body. The source was positioned in various locations within the inserts' cavities and within the CV cavity in the model without an insert to determine the location of maximum dose rates. Initially, these locations included the center of the cavity base, the cavity wall at the cavity's axial mid-plane, and the center of the top of the cavity. For the model without an insert, the source was also positioned at the top of the cavity next to the wall. This position aligns the source with the small gap that is modeled between the CV lid and flange. It also aligns the source with the reduced DU shielding in the CV body. The package cork was modeled as void. The purpose of the McBend calculations was to determine the location of the maximum dose rates. Thus, the models used nominal, as-designed properties for the package. The calculations used a 3000 Curie (Ci) Cs-137 source. Based on the results for the McBend analyses, the applicant determined that the highest dose rates for packages with the inserts occur on the surface of the package base.

The staff reviewed the McBend models as described in the application, including the model drawings. The staff's evaluations included independent calculations, using the MicroShield code, of different nuclides as the source. These nuclides include cobalt-60 and sodium-24. The purpose of selecting these nuclides was to confirm that the location of maximum dose rates would not change for gammas of different energies. Based on these calculations, the staff finds that, of the locations initially evaluated by the applicant, the maximum dose rates will occur at the base surface of the package for the proposed contents.

The staff, however, determined that not all appropriate configurations were evaluated for determining the maximum dose rate location. Thus, the applicant modified the analysis to include calculations for the source in the top of the insert cavity next to the cavity wall (referred to as the "top eccentred" configuration) with the insert up against the CV lid. This location resulted in the maximum dose rates for the package. To address this result, the applicant assessed the increase of dose rates for the source in this configuration versus the dose rates for the source at the cavity base. From this assessment, the applicant determined a factor for each insert by which the dose rates (determined for the source at the cavity base) were to be increased. The applicant based the factors on calculations for the Cs-137 contents. The actual impact for different contents that emit gammas of different energies will vary. The applicant looked at some of these other contents and determined that the impact for the Cs-137 will bound the impact for the other contents. The staff also did some independent evaluations of the possible impacts for other nuclides. Based on its review of the applicant's analysis and its own evaluation, including consideration of the margins to the dose rate limits for the contents, the

staff finds that using the Cs-137 factors for all the proposed contents is adequately bounding and therefore acceptable.

The applicant performed calculations using the MicroShield code to determine the maximum quantities of the proposed radionuclide contents based on the non-exclusive use dose rate limits. The models are only for the configuration of the point source at the base of the insert cavity since the applicant's initial McBend calculations identified the package base surface as the location of maximum dose rates. The MicroShield calculations account for the proposed nuclide and its progeny at the decay time that maximizes the dose rates. The bremsstrahlung contribution is also included in the source definition in the models for the nuclides and their progeny. The staff finds that using a point source is acceptable and may be conservative. However, since the proposed contents descriptions would allow the shipment of sources that are in effect point sources, the extent of the conservatism depends on the configurations of the package's actual contents. The models include the keg, the CV, and the inserts. The staff reviewed the model dimensions in Section 5.4.1 of the application and finds they are consistent with the nominal dimensions for the inserts, CV, and keg through the base of the package in the proposed CoC drawings. The applicant's analysis used the same model for both the NCT and HAC calculations. The use of nominal package dimensions introduces uncertainties into the models. Section 5.4.5 of this SER describes these uncertainties and how the applicant accounted for them.

The models for both McBend and MicroShield calculations assume the contents remain in the inserts. This is an important assumption that can significantly impact the package dose rates. This assumption attributes a containment capability to the inserts, though the inserts are not relied upon for demonstrating the package's containment capability (see Chapter 4 of this SER). It also relies upon the ability of any product container in which the contents will be shipped to retain the materials. The package operations in Chapter 7 of the application describe the closure of the insert and the checks on that closure (see SAR Section 7.1.2, steps 4 and 7). Tables 1-3-1, 1-3-2, and 1-3-6 of the application provide some information regarding product containers that may be used.

The staff considered the package operations and information in the listed tables. The staff also considered information provided by the applicant that describes drop tests and their results for the inserts. The tests indicate that contents will not leak from the insert. However, it was not clear from the material presented that the impacts in the test, done in some kind of testing device, bound the impacts of the same tests for an insert in a package. Further, the staff had concerns regarding the consistency of the closure based on a “hand tight” criterion. Thus, the applicant modified the insert design to include match marks on the insert body and lid. The operations have been modified to use the criterion of aligning the match marks to ensure proper closure. The applicant has also modified the package design and operations to include a rubber disc that is placed on top of the insert in the CV cavity to prevent motion of the insert, including any loosening of the insert closure due to package vibration during shipment. The applicant performed a vibration test for a package with these design features (i.e., match marks on the inserts and the rubber disc) and assembled as the package would be used. The results of the tests demonstrate that the inserts will not leak. The staff finds that the vibration test may be one of the more challenging conditions for the insert closure for both NCT and HAC. Thus, based on the tests and their results, the design and operations changes to ensure a consistent closure of the insert lid during package operations, the operations checks to ensure the insert closure features, including the O-ring, are present and undamaged, and the use of product containers, the staff has reasonable assurance that the contents will remain within the inserts.

Consequently, the staff has reasonable assurance that the shielding models are appropriate with regard to this design aspect.

5.3.2 Material Properties

As described earlier in this SER, the materials relied on for shielding are the tungsten of the inserts, the stainless steel and the DU of the CV, and the stainless steel of the keg. The cork material is not relied on for shielding. Any material properties of the proposed contents are also neglected.

Tables 5-7 and 5-8 of the application describe the material properties used in the McBend and MicroShield analyses, respectively. The steel and tungsten components are code materials that are specified in the proposed CoC drawings. Thus, the material properties for these materials are governed by the codes under which they are fabricated and designated. The DU is not a code material; thus, the proposed CoC drawings specify the material properties, including chemical composition and minimum density.

The staff reviewed the material specifications in Table 5-7 of the application for the McBend analysis. For the steel and tungsten materials, the properties are consistent with those reported for these materials in publicly available references. The DU composition is consistent with the specification in the CoC drawings while the density exceeds the minimum stated in the CoC drawings. The staff finds the model's DU density acceptable for the McBend analyses since the only purpose of these calculations is to identify the location where maximum dose rates occur.

The staff reviewed the material specifications in Table 5-8 of the application for the MicroShield analysis. The ability of MicroShield to perform analyses with alloyed or composite materials is somewhat limited. The default materials in the code include a limited number of elemental materials (e.g., iron and tungsten) and an even more limited number of composite materials (e.g., air). The user may create custom materials; however, the definition of that material goes beyond merely describing the material's constituent elements. Additional aspects of the material must also be defined that influence the material's shielding performance. Thus, great care must be exercised in the definition and use of such custom materials. Therefore, the applicant represented steel with iron, the DU shield as solely uranium, and the tungsten alloy as solely tungsten. For each of these materials, the applicant used a density that was less than the alloys' densities in the McBend model. The applicant adjusted the material properties, namely the densities, so that the MicroShield results would match the McBend results for the same Cs-137 source.

In its review of the material properties of the MicroShield model, the staff considered the properties and amounts of the different constituents of each alloy. Based on these considerations, the staff finds the specification of the steel as iron with the selected density to be acceptable. This density is more than the iron density in the steel; however, considering the proximity of the other constituents to the iron in the periodic table, the staff expects that the overall shielding behavior of the steel will be close to that of iron. Thus, the staff has reasonable assurance that the analyzed density, which is less than the density of steel, adequately accounts for the uncertainty in that assessment. The staff considers the density used for the uranium in the model to be a source of uncertainty and possibly non-conservative given the material specifications in the proposed CoC drawings. The staff also considers the density used for the tungsten to be a source of uncertainty and possibly non-conservative given the density and composition of the tungsten alloy as used in the McBend analyses. While the staff recognizes that the applicant adjusted the MicroShield model so that its results match that of the

McBend analyses, the staff notes that this was done for a single gamma energy. The two models may diverge at different gamma energies. Also, as noted previously, the staff finds the uranium density in the McBend model acceptable based on the purpose for which that model was used. The staff does not find it acceptable for determining package dose rates for the same reasons the staff has concerns with the uranium density used in the MicroShield model. The staff included the uncertainties due to the density and composition specifications for the uranium and tungsten in its evaluation of the overall analysis uncertainties described later in this SER chapter.

5.4 Shielding Evaluation

5.4.1 Methods

As already noted, the applicant performed its analyses using two computer codes, McBend and MicroShield. The McBend code is a 3-D Monte Carlo code that is capable of dealing with neutron, gamma, and charged-particle radiation. It has the capability of analyzing 3-D problems and includes the ability to analyze complex geometries such as streaming paths. The code is the product of the ANSWERS Software Service of AMEC. The code has a twenty-five year history of uses in a wide variety of applications, including the shielding analyses of various NRC-approved transportation packages and spent fuel dry storage systems. The latest version of the code, version 10A_RU1, was used for the Safkeg-HS package shielding analyses. The applicant used this code to determine the location of maximum package dose rates only. Based on its capabilities and its history of use in other approved applications, the staff finds the code to be acceptable for use in the Safkeg-HS application.

The MicroShield code is a 1-D point kernel code capable of dealing with gamma radiation only. It does have some unique useful features such as the ability to select from a list of many radionuclides as sources and to decay these nuclides to account for the gamma contributions from any of a nuclide's progeny. Although it is widely used, it has important limitations and sources of uncertainty. These include the materials and material properties already noted. It is a 1-D code; therefore, it cannot handle complex geometry issues such as streaming paths. In addition, the code uses an approximation to deal with radiation buildup in the shielding materials. Whereas each material has its own buildup properties, MicroShield is only able to use a single material's buildup properties for a calculation. It applies these properties to all the materials in the model. This can be a source of uncertainty and the user must wisely select the material that defines the buildup for the problem. This is often the material that is closest to the detector with a thickness of at least three mean free paths. Thus, the staff finds that the use of this code may be acceptable for shielding analyses for simple geometries and shielding configurations, including materials. Additionally, the code uses ICRP 51 DCFs to calculate dose rates. The issue with using these DCFs is described in Section 5.4.3 of this SER. Based on the applicant’s use of the code, calculations for the base of the package (a simple geometry) and other analyses and techniques to address the uncertainties and limitations of the code (see Sections 5.4.3 and 5.4.5 of this SER), the staff finds the code’s use is acceptable for this application.

As already described, the applicant accounted for bremsstrahlung using a method described in Cember [1]. This method converts the beta source into an equivalent gamma source with energy equal to the maximum beta energy. The applicant applied this method to the applicable proposed contents and progeny. The applicant then either added this equivalent gamma source into the MicroShield source input portion of the model to obtain dose rates that included both gamma and bremsstrahlung contributions or calculated the bremsstrahlung dose rates

separately in MicroShield and added these dose rates to the gamma dose rates for the applicable proposed contents and progeny. The staff finds this method to be a conservative and an acceptable approach.

For the proposed contents, the form may be special or normal form. Thus, the applicant accounted for the contribution of the progeny to the dose rates. The applicant used the exposure rate versus decay time tool available in MicroShield to predict the decay time at which dose rates reached their maximum level. The applicant then calculated the gamma dose rates for a 1 Ci amount of the parent nuclide (except for Cs-137, for which 3000 Ci was used) decayed to this time, using MicroShield's decay tool for the source. MicroShield uses data from ICRP Publication 38 [2] for the decay calculations. The applicant also consulted the data available in DOE/TIC-11026 [3], particularly for the bremsstrahlung contributions from the progeny. Data used from DOE/TIC-11026 was manually input into MicroShield.

Tables 10 and 11 of CTR 2011/01, Issue D, show the gamma dose rates and bremsstrahlung dose rates from each parent and its progeny for each insert. Gamma dose rates are only shown for the parent nuclide; however, the dose rate is from the decay time that maximizes the gamma dose rates. So, the gamma dose rates shown in the table include the contributions from the progeny. Only the bremsstrahlung from each of the progeny is shown differently. The bremsstrahlung dose rate is the dose rate for 1 Ci of the nuclide being present in the package, regardless of the actual quantity present at the decay time of maximum dose rates. The staff finds this approach acceptable given the conservative nature of the bremsstrahlung calculation method and the fact that for many nuclides, if not all, the quantity of each daughter will be less than the original 1 Ci of the parent due to decay. The gamma dose rates and bremsstrahlung dose rates are added together. The applicant attributes this dose rate to the initial, undecayed, quantity of the parent nuclide (i.e., the proposed nuclide), which is 1 Ci. This quantity is then scaled up to the maximum shielding-based quantity by the ratio of the most restrictive dose rate limit to the calculated dose rate. The most restrictive dose rate limit is the 2 mSv/hr limit for the package surface. Thus, the maximum quantity of the proposed nuclide is 1 Ci times the ratio of 2 mSv/hr to the calculated total dose rate for that nuclide and its progeny given in Table 10 or 11 of CTR 2011/01, Issue D, for the appropriate insert.

To modify the shielding analysis to account for the uncertainties and factors such as the source position that maximizes package dose rates, the applicant applied adjustment factors to the dose rate, 2mSv/hr, to determine the final dose rate of the package surface, which is compared against the exclusive-use limit for a package in a closed vehicle. The applicant included additional margin for the dose rates by limiting the calculated package dose rates to 8 mSv/hr. If the adjusted dose rates exceeded 8 mSv/hr, the nuclide quantity limits were reduced by the factor necessary for the dose rates to be below this value. The resulting quantity was the shielding-based quantity limit for the proposed radionuclide content. The staff finds this approach to be acceptable since it addresses the decay of the proposed contents and the dose rate contributions from progeny that are themselves radioactive.

For some of the proposed contents, the progeny contribute significantly to or can dominate the dose rates attributed to the proposed contents. As noted previously, the dose rate is attributed to the proposed radionuclide contents for the condition when no progeny are present (i.e., at a time prior to any decay of the proposed contents). Thus, the CoC condition describing the contents limits clarifies the condition of the contents to which the quantity limits apply. That condition is the condition with no progeny present with the proposed nuclide. This is important because an analysis that attributes the quantity limit to some point of decay where the proposed nuclide content is at the quantity limit and decay progeny are present indicates that dose rates

for the contents will exceed those used in the applicant's analysis. This in turn would mean a lower quantity limit is needed for the proposed contents than is listed in the CoC's approved contents limits.

The applicant's overall method is to use the non-exclusive use dose rate limits at the location initially identified to be the most limiting (in terms of dose rates) to back-calculate the acceptable contents activity limits in terms of shielding. The final activity limits are based on the more restrictive limits of dose rate, heat rate, and mass limit. With regard to deriving contents limits from the regulatory dose rate limits, the staff finds that this is an acceptable approach. However, in using this approach, the applicant must account for the method's uncertainties to assure that the dose rates from the proposed contents will not exceed the regulatory limits. The applicant did perform an uncertainty analysis and re-analyzed the source position for maximum package dose rates. Based on these analyses, the applicant adjusted its overall analysis to demonstrate that the package meets the dose rate limits for exclusive use in a closed vehicle with added margin. For that analysis, the applicant assumes the package is shipped in a closed vehicle and its position is fixed such that it is at least 20 cm from the vehicle's surfaces. The distance from the vehicle's surfaces was selected to ensure compliance with the NCT limit for those surfaces of 2 mSv/hr (200 mrem/hr) with margin as well. The dose rate-derived contents limits were adjusted as needed as a result of this analysis. The dose rate-based (or shielding-based) quantity limits for the light tungsten insert's contents were reduced by a factor of two because the calculated package surface dose rate exceeded the 8 mSv/hr limit (and the regulatory limit) when the 2 mSv/hr dose rate was adjusted to account for the uncertainties and factors described in this chapter of the SER. The staff reviewed this approach and finds it to be acceptable based on the description of the approach, the margins to the exclusive use limits, and its own, independent evaluation of the uncertainties. The staff's evaluation of the uncertainties is described later in this SER chapter.

5.4.2 Input and Output Data

The applicant provided sample files for the MicroShield calculations used to determine the dose rates for different nuclides. The staff reviewed these input files and finds that the information regarding material properties and dimensions used in the calculations is consistent with the descriptions of the calculations given in the application.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant used the DCFs from ICRP 51 to determine the package dose rates. These are the DCFs that are used in the MicroShield code. In particular, the DCFs for the anterior-posterior orientation were used. The staff finds that the use of these DCFs is not acceptable. These DCFs are used to calculate effective dose equivalent (EDE). The 10 CFR Part 71 dose rate limits are for dose equivalent (DE). The models used to derive these two dose quantities are very different. The differences are more notable with lower gamma energies, with EDE always being less than DE. In addition, the acceptability of a particular shipment is demonstrated by dose rate measurement. EDE is not a measured quantity and so the DCFs for EDE are not appropriate for calculating package dose rates. The DCFs of the ANSI/ANS-6.1.1, 1977 Revision, are accepted by the staff as the staff finds these DCFs to be the most appropriate DCFs for the determination of DE. As a result of this issue with the DCFs, the applicant analyzed the differences between the ICRP 51 and ANSI/ANS-6.1.1, 1977 Revision DCFs and treated those differences as part of the analysis uncertainties. The staff finds this approach to be acceptable based on the description of this analysis, the margins to the dose

rate limits and staff's own, independent evaluation of the DCFs and the uncertainties. The staff's evaluation of the uncertainties is described later in this SER chapter.

5.4.4 External Radiation Levels

As already described, the applicant used the non-exclusive use dose rate limits to initially determine the shielding-based quantity limits for the proposed contents. This analysis did not include the impacts due to the NCT and HAC tests described in 10 CFR 71.71 and 71.73. The applicant included these effects as part of its uncertainty analysis, which it performed to determine the uncertainties' impacts on the method and the dose rates. Based on this analysis, the applicant adjusted its analysis to demonstrate that the package meets the exclusive use dose rate limits for shipment in a closed vehicle plus an additional margin. The final quantity limits for the proposed contents are based on the most restrictive of the dose rate, heat rate, and mass limits for the package. A number of the proposed contents quantity limits are based on the dose rate limit. The maximum dose rates, when the described factors and uncertainties are accounted for, are summarized in Table 5-1 of the application for NCT and Table 5-2 of the application for HAC. The applicant selected the location of the package within the closed vehicle to ensure compliance with the NCT regulatory limits of 2 mSv/hr (200 mrem/hr), with margin, for the vehicle surfaces. Based on its review and evaluation of the other aspects of the applicant's shielding analysis, as described throughout this chapter of the SER, the staff has reasonable assurance that the package meets the regulatory dose rate limits.

5.4.5 Uncertainties and Conservatisms

In its review of the applicant’s analysis the staff finds that the shielding models introduce uncertainties and do not address the impacts of the NCT and HAC tests in 10 CFR 71.71 and 71.73. The uncertainties arise from the dimensional tolerances of the inserts, the CV and keg components. Also, it is not clear that the gaps between the CV's steel liner and DU shield and between the CV's DU shield and steel base will be there in an as-fabricated package when tolerances are accounted for, or for a package that has experienced NCT or HAC test conditions. Furthermore, the package operations and maintenance programs described in Chapters 7 and 8 of the application allow for continued use of the package with some level of damage without requiring repair.

To address these issues, the applicant performed analyses of the impacts of the tolerances and damage due to the NCT and HAC tests. In particular, the applicant performed calculations with different nuclides that represented gamma sources of different gamma energies. With the exception of thorium-228 (Th-228), the selected nuclides had peak energies at or below 0.662 MeV. The Th-228 peak energy was at 3 MeV. The applicant’s analysis considered the dimensional tolerances for the package, the removal of the small void areas, and the damage due to the NCT and HAC tests. The applicant’s analysis also included the differences between the ICRP 51 anterior-posterior DCFs and the ANSI/ANS-6.1.1, 1977 Revision DCFs. Accounting for these uncertainties resulted in significantly increased package surface dose rates.

In addition, the applicant analyzed the impacts resulting from the source position. As noted previously, the initial analysis indicated that the configuration with the source at the base of the insert cavity resulted in maximum package dose rates. Further analysis indicated that the maximum package dose rates occur with the source at the top of the insert cavity at the cavity wall and the insert placed against the CV lid (the “top eccentred” case). Since MicroShield is unable to handle such a configuration, analysis of the proposed contents in this configuration

would require calculations with a code like McBend due to the complicated geometry involved. Instead of analyzing all the proposed contents at this position, the applicant used the latest McBend analysis for Cs-137 to determine the source position resulting in maximum dose rates. The applicant used the ratio for each insert of the dose rate for the “top eccentred” case versus the bottom centered case that is the basis for all the contents activity limit calculations as an adjustment factor for the dose rates calculated in MicroShield. The applicant also performed calculations for other representative nuclides; these calculations indicated that the effect of source position for Cs-137 bounded the effect for other nuclides with different gamma energies. In addition, though the applicant’s analysis is for demonstrating compliance with the dose rate limits for exclusive use transport in a closed vehicle, the applicant adds some margin by restricting the package surface dose rates to 800 mrem/hr (8 mSv/hr) vs. the 1000 mrem/hr (10 mSv/hr) limit.

The staff reviewed the applicant's analysis. The staff finds that the dimensions used in the analysis are consistent with the tolerances in the proposed CoC drawings and with the damage described in the structural evaluation (see Chapter 2 of this SER). They are also consistent with the maximum amount of damage that does not require repair for continued package use that is described in Chapter 7 and Section 8.2 of the application. With regard to the selected nuclides, the staff finds that they should be adequate to characterize the greatest differences in the two sets of DCFs; however, for other purposes, additional nuclides with peak energies between that of Cs-137 and Th-228 should also have been considered. The staff notes that the impacts of the dimensional tolerances and NCT and HAC damage were analyzed with MicroShield and the configuration of the source being centered at the insert cavity base.

The staff performed its own independent evaluation of the uncertainties for the same configuration used by the applicant. In addition to the nuclides the applicant considered, the staff included cobalt-60 and sodium-24 as these nuclides have peak gammas at energies between Cs-137 and Th-228. The staff's calculations resulted in a dose rate increase that was similar to the applicant's though a little larger. The staff also considered the uncertainties due to the densities of the uranium and tungsten materials in addition to those considered by the applicant. For the uranium, the staff’s analysis used the minimum density and the composition specification in the proposed CoC drawings (18 g/cc and 2% molybdenum). Based on its evaluation, the staff finds that accounting for the materials tolerances causes the dose rates to increase further by several percent.

As mentioned previously, the choice of a buildup material in MicroShield can also be a source of uncertainty. The staff considered this as well. The concerns with the buildup material selection are primarily a concern at low gamma energies. The change in dose rates with the change in buildup material for the selected nuclides with the lower peak energies indicated a change that was consistent with or less than the change for Cs-137. The staff recognizes that the applicant adjusted its MicroShield model so that the results for Cs-137 would match those of the McBend model. Based on this fact and the results of its own analysis, the staff finds that any uncertainty due to the buildup material selection is adequately addressed in the original model.

The staff also considered the use of an adjustment factor to account for the maximum dose rates occurring for the “top eccentred” source configuration instead of the bottom centered case. Due to the configuration of the package shielding in this location and the location of the highest dose rate on the package surface versus the source location, the staff expects that the effects of the dimensional and material tolerances and the damage due to NCT and HAC will increase beyond those analyzed by the applicant. The staff considered these effects as well as evaluated different nuclides to see how the impacts of source position changed with gamma

energy. The staff also took into account that the applicant's analysis methods include some conservative approximations such as neglecting the cork material. In addition, the applicant has limited package surface dose rates to no more than 80% of the relevant limit. Furthermore, not all contents activities are limited by the dose rates; some are limited by the applicant's proposed mass and thermal limits. For some of those contents, the resulting margins to the dose rate limits are quite significant. Thus, the staff notes there are aspects of the method that need to be improved and there are uncertainties which the applicant's method should more properly address. However, based on the identified conservatisms and margins, the staff finds there is reasonable assurance that the uncertainties of the applicant's analysis method have been adequately addressed for this application.

5.4.6 Corrections to Shielding-Based Contents Limits

The staff noticed several inconsistencies and errors in the shielding-based quantity limits for several of the proposed contents and the results used to derive these limits. These inconsistencies were noticed between documents of the same supplement to the application and between different revisions of the same documents. As a result, the staff used the applicant's own analysis method, as already described in this SER chapter, and consulted the different revisions of the shielding analysis documents submitted by the applicant over the course of the review to determine the appropriate contents limits. Given, the method changed up until the response to staff's second request for additional information (RAI), the documents submitted with that response were used as a starting point. The staff recognizes that some contents' limits still needed to be changed at this point too; thus, the staff accounted for those changes in its evaluation. The staff compared the analysis results among the different documents provided in response to its second RAI. The appropriate or the conservative value of the dose rate results for the analyzed quantity (i.e., 1 Ci prior to decay for all nuclides except Cs-137, which used 3000 Ci), were used. The more conservative value was selected in cases where the basis for the difference in results was unclear. This process was repeated in comparing later versions of the same documents to those provided in response to the second RAI. The result was a listing of the package surface dose rates (gamma plus bremsstrahlung) for each proposed nuclide at the analyzed quantities for both inserts, without addressing the analysis uncertainties.

As part of this process, the staff noticed that the bremsstrahlung of some nuclides and their progeny were not included in the results. For nuclides for which the dose rate contribution had not been calculated, the staff used the applicant's method to calculate the contributions and add them to the proposed contents. These included the bremsstrahlung for gold-198 (Au-198) and iridium-194 (Ir-194). In the case of Ir-194, the applicant pointed out that there is a metastable version of Ir-194. Information indicated that the metastable nuclide should not be present in the contents. The staff accepts that information; however, the staff found that, for the bremsstrahlung analysis, the applicant had used the information for the metastable nuclide and not the Ir-194 of interest. In addition, the contribution of thallium-210 (Tl-210) was missing from radium-226 (Ra-226). Tl-210 also contributes to the gamma dose rate; however, the contribution is negligible. Consequently, the staff found it acceptable not to include the gamma dose rate contribution of Tl-210 to Ra-226.

The staff used the ratio of the 2 mSv/hr dose rate limit to the dose rates from the analysis to scale the analyzed quantities of each nuclide up to the activities that would result in a dose rate at the package surface of 2 mSv/hr prior to accounting for the analysis uncertainties. The resulting activities for the heavy tungsten insert (Insert 3982) are the shielding-based, or dose rate limit-based, quantity limits since they result in dose rates less than the 8 mSv/hr value when

the analysis uncertainties are included. For the light tungsten insert (Insert 3985), the staff reduced the quantities by half as was done by the applicant for the reason described previously in this SER chapter. The resulting quantities are the shielding-based quantity limits for the light tungsten insert. There are some quantities that may still be incorrect; however, because of conservatisms and other factors, they are acceptable. This analysis does not include nuclides beyond those nuclides listed as proposed contents in the two tungsten inserts and included in Tables 1 through 3 of the CoC. Thus, for an application to include the steel insert and its contents as part of the approved contents in a later CoC revision, the staff will need to ensure that the quantity limits for the proposed contents for that insert, including neutron-emitting nuclides such as the plutonium and uranium isotopes, are adjusted to account for the uncertainties and other factors described in this SER chapter as well as account for any modifications needed in the analysis that are unique to those nuclides (e.g., accounting for neutron dose rates as well as gamma dose rates of the parents and the progeny).

The CoC quantity limits are based on the overall more limiting activity derived from the package dose rate, heat, and mass limits. Accounting for the corrections described here, the limits in the CoC for nuclides that are limited by dose rate will differ from those in the application as supplemented. The correct shielding-based quantity limits for each nuclide in the proposed contents for the tungsten inserts are provided in the table below.

Table of Corrected Shielding-Based Quantity Limits

	Quantity Limit			Quantity Limit			Quantity Limit	
Nuclide	3982 insert (Bq)	3985 insert (Bq)	Nuclide	3982 insert (Bq)	3985 insert (Bq)	Nuclide	3982 insert (Bq)	3985 insert (Bq)
Ac-225	2.51E+12	5.47E+11	I-131	4.11E+15	4.97E+14	Re-186	1.56E+14	2.66E+13
Ac-227	7.25E+11	1.63E+11	In-111	1.45E+28	6.43E+27	Re-188	7.25E+11	2.61E+11
Ac-228	4.28E+11	9.30E+10	Ir-192	2.71E+15	3.59E+14	Rh-105	1.05E+31	5.84E+28
Am-241	3.51E+11	7.92E+11	Ir-194	2.05E+12	4.83E+11	Se-75	8.89E+18	3.94E+18
As-77	9.43E+19	5.72E+18	Kr-79	6.02E+13	1.15E+13	Sm-153	6.12E+11	9.54E+14
Au-198	5.10E+14	5.69E+13	Lu-177	1.55E+24	6.89E+23	Sr-89	1.22E+13	2.59E+12
Ba-131	1.88E+14	3.06E+13	Mo-99	5.29E+13	9.56E+12	Sr-90	1.72E+12	4.15E+11
C-14	6.92E+27	3.06E+27	Na-24	2.63E+10	6.38E+09	Tb-161	1.61E+13	3.70E+12
Co-60	2.39E+11	4.68E+10	Np-237	3.57E+12	7.92E+11	Th-227	1.01E+13	2.09E+12
Cs-131	8.38E+35	3.71E+35	P-32	5.56E+12	1.25E+12	Th-228	6.79E+10	1.67E+10
Cs-134	7.05E+12	1.32E+12	P-33	5.10E+37	1.80E+37	Tl-201	1.59E+22	7.05E+21
Cs-137	1.59E+15	1.58E+14	Pb-203	1.18E+17	1.23E+16	W-187	2.24E+13	4.28E+12
Cu-67	6.49E+25	1.42E+24	Pb-210	8.04E+12	1.66E+12	W-188	6.43E+11	2.19E+11
Hg-203	3.57E+13	2.66E+34	Pd-109	2.96E+14	4.81E+13	Xe-133	2.25E+33	9.97E+32
Ho-166	2.04E+12	4.60E+11	Ra-223	1.02E+13	2.07E+12	Y-90	1.72E+12	4.15E+11
I-125	4.48E+35	1.99E+35	Ra-224	8.86E+10	2.19E+10	Yb-169	1.87E+19	8.30E+18
I-129	3.30E+26	1.46E+26	Ra-226	9.91E+10	2.34E+10	Yb-175	6.02E+23	3.03E+22

5.4.7 Heat Load-Based Contents Limits Calculations

The staff also notes that the heat limit-based quantities are derived from a calculation of watt/curie using MicroShield. The code's calculation method, while indicating an option for addressing decay and contributions from progeny, does not account for the progeny in

determining the heat output. Thus, the applicant performed new calculations with MicroShield at varying decay times to determine the maximum heat output. The calculations used an initial 1 Ci amount of proposed radionuclide contents decayed to multiple times. The remaining amount of the proposed nuclide and the amounts of each of the progeny were then manually input into MicroShield to determine the decay heat at those times. The maximum decay heat was identified. The allowable quantity of the proposed nuclide based on heat limits was determined by multiplying the initial 1 Ci amount of the proposed nuclide by the ratio of the heat limit for the content type to the calculated maximum decay heat. This method is similar to what was done for determining the allowable quantity based on dose rate, or shielding, limits (see SER Section 5.4.1). As with that method, since the heat load is attributed to the initial, un-decayed, proposed radionuclide content, the CoC condition is written to clarify that this is the content condition for which the quantity limits apply. The overall quantity limits for the proposed nuclides in the different content types were adjusted accordingly. The limits in the CoC account for these heat calculations.

5.5 Evaluation Findings

Based on its review of the statements and representations in the application and independent confirmatory calculations, the staff has reasonable assurance that the shielding design has been adequately described and evaluated and that the package meets the external radiation requirements of 10 CFR Part 71.

5.6 References

1. Cember, Herman, *Introduction to Health Physics*, Third Edition, McGraw-Hill, New York, 1996. (See pages 129-131 for estimating bremsstrahlung).
2. ICRP, 1983, *Radionuclide Transformations – Energy and Intensity of Emissions*, ICRP Publication 38, Ann. ICRP 11-13.
3. Kocher, David, *Radioactive Decay Data Tables, A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments*, DOE/TIC-11026, Office of Scientific and Technical Information, U.S. Department of Energy, 1981.

6.0 CRITICALITY

In the original submission, the applicant sought authorization to transport fissile materials in a stainless steel insert. However, the applicant indicated in their first additional information response that certification for a stainless steel insert would not be pursued. Per the original submission, fissile materials are only transported in the stainless steel insert. Therefore, a criticality review of the package ceased after the first additional information response was received. Consequently, fissile materials will not be listed as allowed contents in the certificate issued with this SER.

7.0 PACKAGE OPERATIONS

The objective of this review is to verify that the operating controls and procedures meet the requirements of 10 CFR Part 71 and are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

7.1 Package Loading

Preparations for loading include radiation surveys to ensure the package is empty and protect personnel as well as contamination surveys of the package exterior. If either survey indicates levels greater than those which are expected or allowed, an appropriate response is taken. After the package is opened, the top cork and the containment vessel are removed. The containment vessel and the containment vessel lid are inspected for damage which occurred during transport. The closure screws, O-rings, cork packing, and keg components are also checked for damage. Additionally, the shielding inserts are inspected for damage and to ensure that the O-ring seals are present and undamaged. Damaged items which cannot be repaired are replaced. Replacement O-rings are helium leak tested in accordance with ANSI N14.5. The staff has reviewed the criteria for these steps and finds they are consistent with the applicant's analyses of the package.

Before loading the contents, the containment vessel is verified to be dry and clean. Special precautions are provided to ensure the contents comply with the CoC and that the contents are chemically compatible with the product containers and the containment boundary. Since solids and gases are all allowed contents, instructions are provided to ensure the correct insert is employed. To ensure adequate insert closure, the inserts are closed tight such that match marks on the insert lid and body are aligned. This closure method, along with the insert inspections already described, ensures the package contents remain within the insert under both normal and hypothetical accident conditions as assumed in the shielding analysis. The staff reviewed the method for loading and sealing the contents in the inserts and finds it is acceptable. The finding is based in part on the tests described in TR 2014/01/01, Issue A, the use of a silicone rubber disc (described below), and the use of product containers to contain the contents within the insert and their expected durability under NCT and HAC.

After the contents are loaded into the insert, the insert closure is verified as described above and the insert is placed inside the containment vessel. A silicone rubber disc is placed on top of the insert, and the containment vessel lid is installed. A shielding screw is also installed to limit any radiation streaming through the center of the vessel lid. A leak test is performed on the containment vessel lid in accordance with ANSI N14.5. Upon successfully completing the leak test, both the containment vessel and cork packing are placed inside the package. The package lid is installed and a security seal is attached to the package. Contamination and radiation surveys are performed to ensure both non-fixed contamination and radiation levels are less than the transport limits. The package is labeled in accordance with 10 CFR Part 49 requirements.

Based on its review of the package loading operations, the staff finds that the package operations are appropriate and adequate for the purposes of ensuring the package is operated in a manner that is consistent with the package analyses in the application.

7.2 Package Unloading

Package unloading operations are performed per written procedures and include receipt of the package and removal of the contents. The package is verified as the package listed on the shipping documentation, and it is inspected for any exterior damage. Security seals are verified intact. Radiation and contamination surveys are also performed. If the results of these inspections and surveys do not meet specified criteria, instructions are provided for handling the package and rectifying the situation.

After the package has been received, the lid is removed. Cork packing and the containment vessel are removed from the package. The containment vessel lid is removed and the contents unloaded.

7.3 Preparation of Empty Package for Transport

A contamination survey of the interior and exterior surfaces of both the containment vessel and the shielding insert is performed, including the silicone rubber disc, to verify contamination levels meet the requirements in 49 CFR 173.428(d). Based upon the survey results, the containment vessel, the rubber disc, and the shielding insert are decontaminated if necessary. The insert and the rubber disc are placed inside the containment vessel and the containment vessel lid is installed. Both the containment vessel and cork packing are fitted into the package and the package is closed. After verifying the contamination levels and the radiation levels on the package exterior meet the requirements of 49 CFR 173.428(a), the package is labeled in accordance with 49 CFR 172.450 and released for transport.

The staff reviewed these package operations and finds that they are appropriate and adequate for the purposes of ensuring the package is operated in a manner consistent with the package analyses in the application and the requirements for transporting empty packages.

7.4 Evaluation Findings

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

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

Section 8 of the application provided information regarding acceptance testing to verify that each packaging is consistent with the package evaluation in Sections 2 through 6, and a maintenance program to assure that the package maintains its ability to meet the regulatory requirements throughout its service life.

8.1 Acceptance Tests

Section 8.1 specifies the following acceptance tests for each packaging:

- Visual inspection to ensure components, including the inserts, are fabricated and assembled per CoC drawings.
- Verification of dimensions, tolerances, and surface finish specified on CoC drawings by measurement.
- Liquid penetrant examination of welds in accordance with applicable ASME requirements.
- Structural and pressure test at 150% of design pressure performed per ASME B&PV code, Subsection NB-6000.
- Leak rate tests per ANSI N14.5 for components, including the inserts, as well as acceptable leak rates and sensitivities.
- A gamma scan of the fabricated containment vessel body and lid to ensure the containment vessel will perform as designed with respect to shielding.
- Tests to confirm the physical, chemical, and mechanical properties of the depleted uranium meet the specification in the CoC drawings.

For the gamma scan, the measured dose rates from the scan are compared to the dose rates calculated for the same source and configuration (source, shielding, and detector) but with the CV's DU and steel components in the calculation at their minimum dimensions and minimum densities and meeting the chemical composition specified in the CoC drawings. The CV is acceptable if the measured dose rates do not exceed the calculated dose rates. Since the scan will be performed on the fabricated CV body and lid, the dose rates measured on the CV surface will be compared with the calculated dose rates for the CV surface.

The staff reviewed these tests and finds the tests and their acceptance criteria to be acceptable since they are tied to the package design specifications and are supported by the package analyses.

8.2 Maintenance Program

Section 8.2 specifies the following maintenance should be performed annually for each package, including:

- Visual inspection of the following items:
  - welds, surfaces and fasteners on the keg outer shell, containment vessel body and lid
  - threads in the closure of the containment vessel and inserts
  - cork packing
  - confinement vessel O-rings
  - insert components and the silicone rubber disc
- Leak tests of the following items:
  - containment vessel

- inserts

In addition, both the containment vessel leak test and the containment vessel O-ring inspection are performed prior to shipment. O-rings on both the containment vessel and the shielding insert are also replaced annually at a minimum.

The staff reviewed the maintenance programs and finds that these maintenance programs provide adequate assurance that the package will continue to meet the regulatory requirements over the course of its service life in a manner that is consistent with the package analyses in the application.

8.3 Evaluation Findings

Based on its review of the statements and representations in the application, the staff has reasonable assurance 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

In addition to the authorized contents listed in Sections 1.2 and the drawings listed in Section 1.4 of this SER, the CoC includes the following conditions of approval:

Condition No. 6: In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7.0 of the application, as supplemented.

(b) The package must meet the Acceptance Tests and Maintenance Program in Section 8.0 of the application, as supplemented.

Condition No. 7: The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

Condition No. 8: Expiration date: March 31, 2019.

CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated, and the Model No. 3977A package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9338, Revision No. 0
on March 31, 2014.