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

SAFETY EVALUATION REPORT

Docket No. 71-9337
Safkeg-LS Model No. 3979A Package
Certificate of Compliance No. 9337
Revision No. 7

SUMMARY

On August 27, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25241A001 and ML26139A378), as supplemented on February 3, 2026 (ML26034C103), and March 20, 2026 (ML26079A061 and ML26139A379), Croft Associates Limited (Croft, the applicant) submitted a request to revise Certificate of Compliance (CoC) No. 9337 for the Safkeg-LS Model No. 3979A package. The applicant proposed the following changes:

- changes to the Safety Analysis Report for Packaging (SARP, application) to include additional radionuclides in tungsten shielding insert #3983
- combine the application sections into one document and incorporate some minor clarifications and corrections
- add an alternative production method for the lead shielding
- add an alternative containment vessel bolting material

The applicant requested U.S. Nuclear Regulatory Commission (NRC) staff review the changes to the package against the International Atomic Energy Agency (IAEA) Safety Standards Series No. SSR-6, 2018 edition. Because CoC No. 9337 for Model No. 3979A is a domestic certificate, the staff evaluated the proposed changes to the certificate per its regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, which are the applicable regulations for this package (which are not yet aligned with the IAEA SSR-6 2018 edition). Accordingly, compliance with the IAEA SSR-6, 2018 edition regulations was not assessed as part of this review.

The NRC staff reviewed the application using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material" (ML20234A651). Based on the statements and representations in the application, as supplemented, and the conditions listed in the CoC, the staff concludes that the package meets the requirements in 10 CFR Part 71.

EVALUATION

1.0 GENERAL INFORMATION

This SER documents the staff's review of the proposed revision to CoC No. 9337 for the Safkeg-LS Model No. 3979A package. The NRC staff determined that not all disciplines described in NUREG-2216 were affected by the requested revision. This revision only requires evaluations on Confinement, Shielding, and Materials, as described in this SER.

In the August 27, 2025, initial application for proposed revision to CoC No. 9337, the applicant submitted Revision 9 of the application (ML25241A001). The applicant also submitted a Revision History of the application, detailing the changes to Revision 9 of the application (ML25241A008). On March 20, 2026, the applicant subsequently submitted an updated application, Revision 10 (ML26079A061), to respond to NRC staff observations (ML25345A453) and a discussion (ML26072A319). The applicant also submitted a Revision History of the application, detailing the changes to Revision 10 of the application (ML26079A064). This SER evaluates the changes proposed in both Revisions 9 and 10 of the application.

The application included changes to the drawings in the application. The drawings submitted with the application provide a detailed packaging description that can be evaluated for compliance with 10 CFR Part 71 for each of the technical disciplines. The staff evaluated the changes to the drawings in sections 5.0 and 7.0 of this SER.

Evaluation of compliance with the IAEA SSR-6, 2018 edition, regulations was not included in this review.

4.0 CONTAINMENT EVALUATION

4.1 Review Scope

Safkeg-LS, Model No. 3979A, is a Type B(U) package designed to transport CoC-specified radioisotopes. The package was previously reviewed with CoC, Revision 6. The objective of this review was to verify that the revision's containment-related changes met 10 CFR Part 71 regulations.

There were no significant changes in the revision related to the containment boundary or containment boundary testing. Staff notes that American National Standards Institute (ANSI) N14.5, "American Standards for Radioactive Materials – Leakage Tests on Packages for Shipment" (American National Standards Institute, 2022) was referenced in the application, chapter 7, including as it related to the pre-shipment leakage test in section 7.1.3. In addition, ANSI N14.5-2022 was referenced in section 8.1.4 of the application, regarding leakage rate fabrication acceptance tests of the containment boundary (described in figure 4-1 of the application), including the vessel flange, cavity wall, lid top, containment vessel O-rings, and the vessel closure. The acceptance criterion for these tests was listed as $1x10^{-7}$ reference cubic centimeters per second (ref-cm^3/sec). In addition, shielding inserts are to undergo leakage testing to a test sensitivity of 1 $x10^{-3}$ ref-cm^3/sec. Section 8.2 of the application noted that maintenance leakage testing of the containment vessel, including testing associated with the 12-month periodic maintenance, would be in accordance with ANSI N14.5-2022 and have an acceptance criterion of $1x10^{-7}$ ref cm^3/sec; periodic leakage rate testing was also discussed in section 7.1 of the application. The staff notes that there are no significant changes between the ANSI N14.5-2022 standard and the ANSI N14.5-2014 standard, which was endorsed in Regulatory Guide 7.4, "Leakage Tests on Packages for Shipment of Radioactive Material" (ML19240B383).

4.2 Evaluation Finding

Based on review of the statements and representations in the application, staff concludes that the revision has not changed the previously reviewed Safkeg-LS, Model No. 3979A containment design. Therefore, the package has been adequately described and evaluated, and the package design meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The objective of this review is to verify that the Croft Safkeg-LS Model No. 3979A Type B(U) transportation package meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The staff's shielding review evaluated the Safkeg shielding features to provide adequate protection from the radioactive contents within. This review looked at the methods and calculations employed by Croft to determine expected gamma and neutron radiation at locations near the cask surface and at specific distances away from the cask.

The applicant requested the following changes:

1. Additional radionuclides for tungsten shielding insert #3983 (ML25241A012) and a new Monte-Carlo shielding assessment for the #3983 insert (ML25241A010).

2. Alternate production method for the lead machined from solid billet rather than casting. (ML25241A009 drawing 1C-6045)

5.1 Description of the Shielding Design

5.1.1 Design Features

The Model No. 3979A packaging (the Safkeg-LS 3979A), consists of an outer stainless-steel keg enclosing insulating cork packaging, and an inner containment vessel. There are three specific inserts authorized for use in Model No. 3979A, designated as Shielding Insert Design Nos. 3983, 3984, and 3986. 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.

For Revision 9 of the application, the applicant proposed an alternative to casting the lead which involves machining the lead shielding to achieve the same tolerances, and seal welding this component in place. Licensing drawing No. 1C-9012, section 1.3.3, identifies this option and provides bounding tolerances.

5.1.2 Summary Table of Maximum Radiation Levels

5.1.2.1 Inserts 3984 and 3986

Table 5-1 of the application presents the maximum NCT dose rates for packages under exclusive use for inserts 3984 and 3986. The applicant initially calculated the allowable content quantities based on a surface dose rate limit of 2 millisieverts per hour (mSv/hr). However, due to analytical uncertainties, package tolerances, and the calculation methodology, the required surface dose rates were increased in accordance with the CTR 2015/10 report, "Uncertainties Associated with the Proposed Shielding Calculation Method for the SAFKEG-LS 3979A Package." Consequently, the surface dose rates exceeded the limits for non-exclusive use and were therefore evaluated against the criteria for exclusive use.

5.1.2.2 Insert 3983

For insert 3983 with Contents Type CT-2, activity limits for non-exclusive and exclusive use were assessed in Cerberus report C.02086-CROFT-PACK-B02-C01-I1 (ETR 566), “Transport Shielding Assessment for the Croft SAFKEG-LS Package DN 3979 A” (ML25441A010). Limits were based on surface dose rates of 2 mSv/hr for non-exclusive use and 10 mSv/hr for exclusive use, as shown in tables 5-5 and 5-6 of the application. For exclusive use, the applicant calculated the dose rates using a distance-square approach to ensue 0.1 mSv/hr at 2 meters (m) from the package surface. The results are provided in table 5-8 of the application. These limits generally exceed those for surface contact, except for Erbium (Er)-169, though its activity limit surpasses the 200 gram (g) mass cap in PCS 036, “Package Contents Specification for Safkeg-LS - Package Design No 3979A” (ML25241A012). Since calculations assume 2 m from the package surface, there are no restrictions on vehicle type or package placement for exclusive use shipments.

5.2 Source Specification

Both neutron and gamma-emitting sources were evaluated for transport within the Safkeg-LS 3979A container. The maximum allowable activity for each nuclide is specified in tables 3 to 8 in appendix A of PCS 036. A worst-case configuration was established through a series of reference calculations designed to represent the most limiting conditions. These calculations account for solid sources permitted within the container and incorporate conservative assumptions that vary based on the source type and the insert configuration.

5.2.1 Gamma Source

For both solid and gaseous contents, the radiation source (gamma, beta, or alpha, depending on the nuclide) is modeled as a point source for all calculations. Source strength details for the MicroShield shielding calculations are provided in CTR 2009/22, “SAFKEG LS 3979A: Package Activity Limits Based on Shielding” (section 5.6.2 of the application). Source strength details for the Monte Carlo shielding calculations are documented in SERCO/TAS/003191 (ETR 319), “Monte Carlo Modelling of Safkeg LS Container,” and Cerberus report C.02086-CROFT-PACK-B02-C01-I1 (ETR 566), as referenced in section 5.6.2.

The applicant analyzed a bounding configuration of solid and gas sources with a 1 kilocurie (kCi) Iridium (Ir)-192 point source in the containment vessel (CV) cavity. The applicant performed an additional calculation to determine the dose rate at the position of the O-ring seal for material analysis. The staff found that the point-source assumption is conservative under HAC because it gives higher dose rate than other source geometry.

5.2.2 Neutron Source

The only neutron-emitting source material allowed is plutonium, which must be in solid metallic form. The contents are restricted by the CoC to ensure compliance with the general license provisions in 10 CFR Part 71, Subpart C. Because the applicant did not satisfy the additional requirements specified in 10 CFR 71.64 or 10 CFR 71.74, the shipment of plutonium by air is strictly prohibited.

5.3 Shielding Model

5.3.1 Model for Monte Carlo Calculations

For the Monte Carlo calculations, the reference case assumes a 1 kCi Ir-192 point source positioned at multiple locations around the surface of the empty CV cavity, including alignment with the gap between the CV lid and CV body. Additional calculations were performed with the source placed in comparable positions within the two tungsten inserts: LS-12x65-Tu (Design #3984) and LS-31x73-Tu (Design #3983).

5.3.2 Model for MicroShield Calculations

For the MicroShield shielding calculations documented in CTR 2009/22 (section 5.6.2), the applicant modeled the contents as a point source located at the center of the bottom of the insert within the CV. The applicant selected this position because Monte Carlo simulations for the reference case, a 1 kCi Ir-192 point source, indicated that it produces the highest dose rate on the external surface of the package (table 5-2 of the application). In addition, the calculations included point source modeling for the gaseous contents.

5.4 Material Properties

No change in material properties from previous calculations.

5.5 Shielding Evaluation

5.5.1 Source Data

Originally, a set of 76 radionuclides was proposed, but Nickel (Ni)-64 and Gadolinium (Gd)-160 were omitted because they are stable. An additional 23 radionuclides were proposed in this revision, resulting in a total of 97 solid radionuclides selected for modeling, each assigned an activity of 1 kCi. Source spectra for these radionuclides were generated using the ORIGEN module within SCALE. Spectra were calculated at decay times of 0, 0.5, 1, 7, and 14 days to account for daughter nuclides influencing the source term. In most cases, these intervals were sufficient to reach equilibrium due to the short half-lives of the radionuclides. Where equilibrium was not achieved, the decay time was extended to capture the most energetic condition. For some of the additional radionuclides, particularly Neptunium (Np)-237, this extended decay time was potentially unrealistically long. However, the resulting worst-case scenario was significantly more energetic than a more "realistic" decay time within the package. Therefore, adopting this worst-case decay time was considered prudent from a conservative standpoint.

Neutron, gamma, and beta particle fluxes were calculated over an energy distribution per kCi for each radionuclide. Beta flux was used to model bremsstrahlung gamma production in MCNP. Alpha flux was also modeled to simulate neutrons generated by alpha interactions in MCNP. Additionally, neutron flux from an alpha medium of UO_2 was modeled in ORIGEN to provide conservatism in cases where the alpha source might not produce a significant dose. The source spectra generated by ORIGEN is documented in C02086-CROFT-PACK-B02-C01-EX02-I1, "Excel Document of Outputs from ORIGEN Calculations for Assessed Radionuclides."

For this assessment, each radionuclide spectrum was modeled as an isotropic point source, using the decay time corresponding to the highest total particle emission rate for conservatism. This total particle rate was applied as the normalization factor for each radionuclide, producing results expressed in microsieverts per hour per kilocurie (μSv/h/kCi). Point source modeling was used to eliminate self-shielding effects, further increasing conservatism.

For NCT, the most conservative source position is at the center of the bottom of the insert. This assumption is intentionally pessimistic because, under NCT, the radioactive material will be transported within a product container placed inside the insert. Consequently, the actual source will be located some distance above the bottom of the insert. Therefore, modeling the source at the center bottom of the insert for NCT represents a conservative approach.

For HAC, it is assumed that the tungsten insert provides confinement for solid radionuclides and that the bounding case disregards any confinement offered by the product container. As a result, the geometries for NCT and ACT are considered equivalent. To evaluate positional sensitivity, the source location was varied within the tungsten insert. Additional dose rates were calculated for positions at the middle and top of the insert, as well as at points offset radially by 1.5 centimeter (cm) from the center at the bottom, middle, and top. These positions are illustrated in figure 1 of Report C.02086-CROFT-PACK-B02-C01-I1 shows the 3-dimensional sliced view of the model.

5.5.2 Material Data

Material compositions and densities used in this assessment were sourced from the Pacific Northwest National Laboratory (PNNL) materials compendium and a previous evaluation. These values are summarized in table 1 of Report C.02086-CROFT-PACK-B02-C01-I1. The lead alloy specified in the schematic document, "Croft Limited, 005 Att 1 Section 1 Licensing Drawings," consists of 96% lead and 4% antimony, with density taken from standard material datasheets. Minor impurities (approximately 0.5%) were excluded due to the difficulty of identifying credible constituents. Cork was modeled as a low-density wood. Stainless steel 430 properties match those used in the prior assessment and align with standard datasheet values. The tungsten alloy density was selected at the conservative end of the range reported in standard material references.

5.5.3 Flux to Dose Rate Conversion

The flux-to-dose rate conversion data used in the Monte Carlo shielding calculations reported in section 5.6.2 of SERCO/TAS/003191 were taken from International Commission on Radiological Protection's Publication 74 (ICRP 74), "Conversion Coefficients for use in Radiological Protection against External Radiation." For the MicroShield shielding calculations documented in section 5.6.2 of CTR 2009/22, the software applied conversion factors from ICRP 51, "Data for Use in Protection against External Radiation," using Anterior/Posterior values as they yielded the highest surface dose rate. Notably, ICRP 51 produces lower surface dose rates compared to ANSI/ANS 6.6.1-1977, "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR [light-water reactor] Nuclear Power Plants." Therefore, a correction factor was applied to the results as detailed in CTR 2015/10. These corrections now apply only to inserts 3984 and 3986.

For insert 3983, as discussed in Cerberus report C.02086-CROFT-PACK-B02-C01-I1 (ETR 566), the assessed dose rates were normalized to μSv/h/kCi and then converted to mSv/h/kCi. Dose rates were derived from particle energy flux using dose conversion factors—specifically the H*(10) factors for Ambient Dose Equivalent from ICRP 74.

5.6 Monte Carlo Calculation

The applicant used ORIGEN module within the SCALE 6.3.1 code system to calculate the source spectra per kCi for each radionuclide, as well as the corresponding particle emission rates, which served as normalization factors for subsequent analyses. Primary shielding evaluations were conducted using MCNP 6.1 based on a detailed model of the Safkeg package. The results from MCNP were normalized using the factors derived from ORIGEN, and dose rate limits were then divided by these normalized values to establish activity limits in kCi. To enhance confidence in the assessment, the staff utilized the MAVRIC module of SCALE as an independent verification tool, performing equivalent calculations to cross-check the results.

5.7 Results

NCT results:

The NCT results for 97 radionuclides are summarized in tables 2 and 3 of (C.02086-CROFT-PACK-B02-C01-I1). In nearly all cases, the highest contact dose rate occurred at the bottom of the package, except for Thallium (Tl)-201, where the top tally was highest. This anomaly likely results from a complex radiation pathway around the lead shielding in the CV lid. For Tl-201, the low-energy radiation cannot penetrate the shielding effectively, making this indirect route dominant. Consequently, Tl-201 results use a source position near the top, offset radially, which produced the greatest contact and 1 m dose rates.

For non-exclusive use, 1 m dose rates were generally highest from the side of the Keg, though contact dose limits were consistently lower than 1 m limits. Prompt gamma radiation was typically the largest contributor to total dose, except for Yttrium (Y)-90, Lutetium (Lu)-176m, Strontium (Sr)-89, and Sr-90, where bremsstrahlung gamma dominated. Five radionuclides produced only bremsstrahlung gamma dose rates despite being prompt gamma sources, while Lead (Pb)-210 was unique in producing bremsstrahlung gamma, alpha neutrons, and secondary gamma, but no prompt gamma.

Among fission neutron emitters, Curium (Cm)-244 was dominated by neutron radiation, Americium (Am)-241 by alpha neutrons, and Np-237 by prompt gammas. Alpha-induced neutrons were generally minor contributors, except for Gd-152 and Pb-210, where they nearly doubled the dose rate, and Am-241, where they were the largest contributor.

Finally, Iron (Fe)-55, Gd-153, Ni-63, Carbon (C)-14, Cesium (Cs)-131, Iodine (I)-125, and I-129 produced zero gamma and bremsstrahlung gamma dose rates. I-129 was notable for producing zero dose despite being both a prompt and bremsstrahlung gamma source, likely due to emissions being too low in energy to escape shielding.

HAC Results:

The HAC results, shown in table 4 of C.02086-CROFT-PACK-B02-C01-I1, were generally similar to the 1 m dose rates under NCT conditions, though slightly higher in most cases. However, there was no consistent source position or tally location that produced the highest dose rate at 1 m.

5.8 Confirmatory Analysis

The applicant evaluated Gamma, bremsstrahlung gamma, and neutron dose rates per kCi for all 97 radionuclides under NCT and ACT conditions. From these, total dose rates were derived, and activity limits in terabecquerel (TBq) were determined based on dose rate limits. For NCT, limits were calculated for contact dose rates (exclusive and non-exclusive use) and 1 m dose rates (non-exclusive use). In all cases, contact dose-rates were bounding for activity limits. Activity limits were also established for HAC.

The applicant reported the calculated external dose rates in table 5.1 of the application. These values are substantially lower than the regulatory limits specified in 10 CFR Part 71:

- Package surface dose rate limit: 2 mSv/hr (200 mrem/hr) per 10 CFR 71.47(a)
- Transport index limit: 10, corresponding to 0.1 mSv/hr (10 mrem/hr) at 1 meter per 10 CFR 71.47(a)
- Vehicle/conveyance limit: 0.1 mSv/hr (10 mrem/hr) at 2 meters per 10 CFR 71.47(b)(3)

The staff reviewed the applicant's shielding models and determined that the conservative assumptions and analysis methods are appropriate to this shielding design package. The staff performed source term calculation with SCALE 6.3.1 and Monte Carlo calculation with MCNP 6.2 for dose rates and use the ANSI/ANS 6.6.1-1977 for flux to dose rate conversion for a 1 kg Ir-192 point source to compare the relative dose rates from select nuclides and staff results are within reasonable agreement with the applicant's analysis.

5.9 Evaluation Findings

The staff concludes that the design of the shielding system for the Croft's Safkeg-LS 3979A package is in compliance with 10 CFR Part 71 and the applicable design criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the Safkeg-LS 3979A will provide safe transportation of radioactive materials. This finding is based on staff confirmation of relative dose rates, confirmation of the reference source dose rate, as well as considerations of the applicant's conservative analyses and modeling assumptions in the application.

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

7.0 MATERIALS EVALUATION

The objective of the NRC staff's materials evaluation is to ensure that the applicant has adequately evaluated the materials performance of the Model No. 3979A transportation package design changes. Croft requested a change to add an alternative containment vessel bolting material in sections 2.1.1 and 2.2.1 of the application. In application section 2.1.1 and 2.2.1, the applicant stated that L7 alloy bolting steel material may be used as justified in CTN 2025/01, "Comparison Between ASTM A320 Grade L7 and Grade L43," which provides a comparison of the L43 and L7 grades.

The staff reviewed sections 2.1.1 and 2.2.1 of the application and CTN 2025/01 and evaluated the comparison tables provided by the applicant for L43 and L7 grades. The staff found that the mechanical and thermal properties are similar with a minor difference in the modulus of elasticity. As the properties are nearly equivalent to the previously reviewed revision, the staff finds the option to add the L7 grade acceptable.

The applicant described in section 5.1.1 of the application a change to add an alternative lead component manufacturing option (machined rather than cast in place). The applicant proposed an alternative where the lead shielding would be machined with the same fit, dimensions and tolerances and seal welding and that this alternative would be described in application licensing drawing 1C-9012, which provides the bounding tolerances.

The staff reviewed the proposed language in section 5.1.1 of the application and the licensing drawing 1C-9012 and found that as the cast in place components have equivalent thicknesses and dimensions to the previously approved shielding components, and as there is no material change, the staff finds the change acceptable.

The applicant proposed changes in sections 2.1.4 and 8.1.6 of the application to add ultrasonic examination in addition to gamma scintillation to ensure there is no cracking or voids in the lead shielding.

The staff reviewed the proposed language in sections 2.1.4 and 8.1.6 of the application and determined ultrasonic testing to be an acceptable form of non-destructive examination for lead shielding for the thickness utilized in the Model No. 3979A package.

7.1 Evaluation Findings

Based on the statements and representations contained in the application, as supplemented, and the conditions listed below, the staff concludes that the design has been adequately described and evaluated. Thus, the NRC staff finds that the Model No. 3979A package meets the requirements of 10 CFR Part 71.

CONDITIONS

The CoC includes the following conditions of approval:

Condition No. 3.b., "Title and Identification of Report or Application," was changed to identify the title and date of the latest SARP Revision 10.

Condition No. 5.(a)(2), "Description" was changed in two places to note that the lead shielding is either cast or a pre-machined component fitted.

Condition No. 5.(a)(3), “Drawings,” was changed to:

- Reference the updated revisions of the drawings.
- Add “Assy A” to existing drawing titles for the containment vessel to reflect the cast lead shielding variant: 1C-6044, Rev. H, “Containment Vessel Design No. 3980 (Assy A)” (Licensing Drawing); 1C-6045, Rev. F, “Containment Vessel Lid (Assy A)” (Licensing Drawing); and 1C-6046, Rev. F, “Containment Vessel Body (Assy A)” (Licensing Drawing).
- Add “Assy B” to new drawing titles for the containment vessel to reflect the pre-machined lead shielding variant: 1C-9012, Rev. B, “Containment Vessel Design No. 3980 (Assy B)” (Licensing Drawing); 1C-9013, Rev. A, “Containment Vessel Lid (Assy B)” (Licensing Drawing); and 1C-9014, Rev. A, “Containment Vessel Body (Assy B)” (Licensing Drawing).

Condition No. 5(b)(2)(ii), table 2, for the contents described in Condition No. 5(b)(1)(ii) for solid, normal form material within insert Design No. 3983, was revised to include additional radionuclides and their corresponding activity limits. These radionuclides and activity limits are presented in table 1-12 of the application and in table 4 of PCS-036.

Condition No. 5(b)(2)(ii) was also revised to clarify how radionuclides are listed in CoC table 2. The radionuclides and activity limits in CoC table 2 reflect bounding quantities that include contributions from each radionuclide’s daughter products, consistent with the modeled source term, as discussed in SER section 5.5.1. In addition, certain daughter radionuclides are listed separately in CoC table 2, with their own activity limits, because they may be transported independently from their parent radionuclides (ML26140A289). Accordingly, the following statement was added to this CoC condition:

> The content limits in table 2 include bounding quantities of each listed radionuclide’s daughter products. Note that some daughter radionuclides also appear as separate entries in table 2 because they may be transported independently from their parent radionuclides.

Condition No. 9, regarding continued use of Revision No. 5 to the certificate, was changed to (1) remove reference to Revision No. 5, since the date for continued use of Revision No. 5 has passed, and (2) allow the use of the previous certificate, Revision No. 6, for approximately 1 year.

The References section was changed to reflect the title and date of the latest SARP Revision 10. The dates of the previous supplements to previous applications were deleted.

CONCLUSION

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. 3979A package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9337, Revision No. 7.