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DOE Packaging Certification Program

Technical Review Report for the Justification for Small Gram Quantity Contents Safety Analysis Report for Packaging Model 9977-96 Addendum 3

S-SARA-G-00006, Revision 4

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OVERVIEW

This Technical Review Report (TRR) summarizes the results of the Staff's review of the Model 9977-96 Package, Addendum 3, Revision 4 for the Type 4 contents. Type 4 contents may include ^{238}Pu, ^{239}Pu, ^{241}Am, ^{244}Cm, ^{252}Cf, ^{90}Sr, ^{226}Ra, ^{137}Cs, ^{60}Co, ^{192}Ir with the Special Actinide Isotopes (^{242m}Am, ^{243}Cm, ^{245}Cm, ^{247}Cm, ^{249}Cf and ^{251}Cf) limited to a total of 1000 ppm. Type 4 pertains primarily to the Off-Site Source Recovery Project (OSRP). These new contents will be authorized for shipment in the Model 9977-96 Package, supplementing existing Revision 2 to the Safety Analysis Report for Packaging (SARP), once Addendum 3, *Justification for Small Gram Quantity (SGQ) Contents*, is accepted by the certifying official, EM-40, and the existing Certificate of Compliance (CoC) is revised, reflecting the added contents. Existing Content Envelopes for the Model 9977-96 Package include heat sources in food-pack cans or in radioisotope thermoelectric generators (RTGs), neptunium metal, a beryllium-reflected plutonium ball, plutonium/uranium metal at 25% and 50% maximum ^{240}Pu, respectively, and uranium metal at limits of 95% and 100% ^{235}U, respectively. The percentages are of total radioactive material mass. The Isentropic Compression Experiment (ICE) apparatus is an additional content of the Model 9977-96 Package SARP. The latest content added to the Model 9977-96 Package is Type 5, AGR fuel compacts.

Addendum 3 was prepared by Savannah River Packaging Technology, Savannah River National Laboratory, Savannah River Nuclear Solutions, LLC, Savannah River Site, in support of work being performed by LANL.

The Model 9977-96 Package is currently certified under two Certificate of Compliance Numbers, i.e., USA/9977/B(M)F-96 (DOE) and USA/9977/B(M)F-96 (DOE-S/T-1), covering transportation and periodic and extended maintenance, respectively. For transportation, the Safety Analysis Report for Packaging is S-SARP-G-0001, Revision 2 (August 2007). Extended maintenance is covered by Addendum 1, *Justification for DNDO Contents*, S-SARA-G-00003, Revision 2 (October 2008).

The new Content Envelope and container configurations will be incorporated into the next revision to the Model 9977-96 Package SARP.

This TRR addresses only Type 4 contents; content Types 1, 2, and 3 will be covered under a separate TRR. Content Type 5 has been dealt with in a previous TRR. Hereafter, in this TRR, the Type 4 content may be referred to as *Sources*.

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Chapter 1: General Information

This Technical Review Report documents the Staff's review of *Justification for Small Gram Quantity Contents, Safety Analysis Report for Packaging, Model 9977, Addendum 3*, S-SARA-G-00006, Revision 4 (March 2010)[1] — the Submittal — prepared for the Department of Energy (DOE) by Savannah River Packaging Technology, Savannah River National Laboratory, Savannah River Nuclear Solutions, LLC, Savannah River Site, to support the shipment of a variety of so-called *Orphan Sources*, using the Model 9977-96 Package. This section of the TRR covers the review of the General Information provided in Chapter 1 of the Submittal.

The Submittal is an Addendum to S-SARP-G-00001, Revision 2 (August 2007),[2] just as S-SARA-G-00003, Revision 2[3] is Addendum 1, *Justification for DNDO Contents*, to the Model 9977-96 Package SARP. Addendum 2 to the Model 9977-96 Package SARP is *Justification for Metal Contents*, S-SARA-G-00005, Revision 1, December 16, 2008.[4] The safety basis, described in the Submittal, addresses specific supplements to the currently approved SARP. The Model 9977-96 Package is currently certified for transportation by the DOE under Revision 4 to the Certificate of Compliance (CoC),[5] and for storage/transportation under Revision 0 to the CoC.[6]

The new Content Envelope, Content Type 4 (i.e., Sources), will assist with the *Off Site Source Recovery Program* for the disposition of radioactive sources. For Sources, various radioactive isotopes have been proposed for shipment, including ^{238}Pu, ^{239}Pu, ^{241}Am, ^{244}Cm, ^{252}Cf, ^{90}Sr, ^{226}Ra, ^{137}Cs, ^{60}Co, ^{192}Ir, with the Special Actinide Isotopes (^{242m}Am, ^{243}Cm, ^{245}Cm, ^{247}Cm, ^{249}Cf and ^{251}Cf) limited to a total of 1000 ppm.

Three different types of Shielded Containers are proposed for use in transporting these Sources. The first is constructed of lead, encapsulated in stainless steel, with a threaded stainless steel closure. The lead provides gamma-radiation shielding. The second is constructed of high density polyethylene (HDPE). The HDPE provides neutron radiation shielding. The third is constructed of tungsten, and is also encapsulated in stainless steel, with a threaded stainless steel closure.

Gamma-sources will be placed in the lead containers (SGQ-SC1) to meet the requirements of 10 CFR 71.47.[7] The decay heat load for the SGQ-SC1 containers is limited to 6 watts. Gamma-sources can also be placed in the tungsten containers (SGQ-SC3), the tungsten containers being an acceptable substitute for the lead containers. Neutron-sources, on the other hand, will be placed in the HDPE containers (SGQ-SC2) to meet the neutron shielding requirements of 10 CFR 71.47. In this case, however, the decay heat load is limited to 3 watts.

An Engineered Container (i.e., SGQ-EC1) is also described in the Submittal, which was intended for use provided the administrative dose rate limits of 180 mrem/hr (on contact) and 9 mrem/hr (at 1-meter) are met. This container, however, has not been evaluated for use in this TRR — see below.

The Submittal addresses nonexclusive use shipments, and the maximum weight of the payload remains at 100 pounds.

Findings

The Staff has found that the Submittal does not follow the standard approach long-prescribed by the DOE's Packaging Review Guide (the PRG),[8] and/or the NRC's Reg. Guide 7.9.[9] The applicant contends that the safety basis for the Submittal need only be based on dose-rate measurements at the surface of the shielded containers, as opposed to the bounding dose-rate calculations for the proposed contents, properly outlined in Chapter 5 of the Submittal. Although most of the dose-rate issues have been resolved — primarily as a result of dose-rate calculations performed by the Staff — the Staff is not in agreement with this approach.

The Staff, therefore, concludes that, based on the review of the statements and representations in the Submittal, specifically the proposed use of the Engineered Container (i.e., SGC-EC1) without specifying the content mass limits for use of such container and the proposed operating procedures, the packaging design has *not* been adequately described to meet the requirements of 49 CFR 173.7(d), 10 CFR 71.31 through 10 CFR 71.35, or 10 CFR 71.47.

Conditions of Approval

The Staff has also concluded that, at a minimum, the following additional conditions of approval need to be added to the existing CoC[5] for the approval of this request:

- The maximum masses/activities for the proposed Type 4 Contents are limited to the masses/activities noted below:

Isotope	Recommended Maximum Mass Limit [g]	Recommended Maximum Activity [Ci]
Co-60	1.0E-04	0.11
Cs-137	1.0E-01	8.70
Ir-192	3.8E-03	35.00
Sr-90/Y-90	1.0E+00	281.80
Ra-226	2.0E-01	0.20
Am-241	1.0e+00	3.43
Cf-252	6.7E-06	0.0036
Cm-244	1.0E+00	80.90
Pu-238	2.0E-01	3.42
Pu-239	6.6E+01	4.09

and

- The Packaging Configuration requirements specified in Addendum Table A.1.2, *SGQ Packaging Configuration*, must also be followed, with the additional notation that the SC3 (Tungsten) Container may be used as a suitable replacement for the SC1 Container for any of the γ-emitters.

It should also be noted that this review specifically excludes any inclusion of the proposed Type 1, Type 2, or Type 3 Contents in the Submittal, and/or the proposed use of the SGC-EC1 Container, for which no shielding calculations or content mass limits have been provided.

Chapter 2: Structural Evaluation

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the assessment of the Structural Evaluation information provided in Chapter 2 of the Submittal.

Details of the items reviewed are noted above in Chapter 1. The results of the structural review are discussed below.

In Chapter 2, the Submittal presents the following information and conclusions concerning the structural requirements and performances of the Model 9977-96 package with the Small Gram Quantity Shielded Containers (SC1, SC2, and SC3) and the Engineered Container (EC1):

- The weight of each of the containers, including the contents, is near the design payload weight of 100 lb. Thus, the overall impact performance of the package is unchanged, and the maximum impact acceleration measured from the certification drop tests are used for the design of the shielded carrier and its supports within the containment vessel cavity (spacers).
- The allowable decay heat rate is 19 watt for the contents. However, the rate is reduced to 6 watts for the contents of SC1, and only 3 watts for those of SC2. These reductions are necessary to maintain adequate strength of the shielding materials. These requirements are supported by the thermal analyses described in Chapter 3 of this Submittal. Chapter 3 also provides estimates of the Maximum Operating Pressures. They are bounded by the design MNOP of the 9977-96 package.
- The SGQ design requires each of the contents carriers/containers, except the EC1, to maintain its structural integrity during all Normal Conditions of Transport. The EC1 is intended for use for contents without significant radiation emissions, and its structural integrity is not a concern. Appendix 2.1 of Chapter 2 of the Submittal[10] has demonstrated by analysis that the weakest carrier (SC1) would survive a 4-ft NCT free drop. Since test results of the package have shown that the Hypothetical Accident Condition free drop produces about the same maximum impact accelerations, the contents carriers are also expected to survive the HAC drops. Thus the design of the carriers has exceeded the regulatory requirements.

Findings

Based on the review of the statements and representations in Rev. 4 of the Submittal, the Staff has concluded that the packaging design has been adequately described to meet the structural requirements of 10 CFR 71. However, there are exceptions:

- The drawings, R-R1-G-00037, -00039, and -00045, appearing at the end of Chapter 1 of the Submittal, do not show the revision number.

Other than these imperfections in reporting, the Staff finds the information and conclusions in Chapter 2 acceptable. The Staff concurs that the contents carriers SC1, SC2, SC3 and EC1 have adequate structural performance to meet the safety requirements of 10 CFR 71.

Conditions of Approval

The Staff has concluded that no additional structurally-related conditions of approval need to be added to the existing transportation CoC for the approval of this request. However, the Staff recommends that the foregoing-mentioned errors in the drawing references are corrected in the submittal prior to issuance of the CoC.

Chapter 3: Thermal Evaluation

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Thermal Evaluation information provided in Chapter 3 of the Submittal and its associated Appendices.

Details of the items reviewed are noted above in Chapter 1. The results of the thermal review are discussed below.

The Model 9977-96 Package is presently authorized for the shipment of contents in RTGs, Food-Pack Cans, DOE-STD-3013 Containers, and Engineered Containers.[5] The maximum decay heat from radioactive contents is limited to 19 watts for the package, which is based on the considerations of materials integrity for the 6CV, and the maximum operational temperature limit of the Viton® O-Rings.[11]

This addendum requests new contents and new package configurations. The newly proposed contents are listed as Type 4 Sources. The decay heat limits per package are 6 Watts for the Lead Shielded Container (i.e., the SGQ-SC1) configuration, and only 3 Watts for HDPE Shielded Container (the SGQ-SC2) configuration. Although the Tungsten Shielded Container (the SGQ-SC3) configuration is actually rated for up to 19 Watts, the Tungsten Shielded Container configuration, in this case, in only allowed as an acceptable substitute for the Lead Shielded Container configuration.

The limiting temperature for the lead is 200 ºF, as required in the structural evaluation.[10] For high density polyethylene (HDPE) and Tungsten, the limiting maximum temperatures are 226 ºF and 6,129 ºF, respectively.

The submittal indicates that the maximum local temperature of the lead is about 247 °F under NCT, including insolation, and less than 199 °F, under NCT, without insolation. For HDPE, the maximum local temperature under NCT with insolation is marginal. However, the shipment is proposed to be in a closed conveyance under non-exclusive use shipment.

There is no calculation for the case of NCT without insolation for the SC2 configuration, but, based on the results of higher decay heat in that configuration, the HDPE will be below the limiting temperature of 226 °F.

The maximum temperatures for the accessible surfaces of the package for all loading configurations are below the required 122 °F for non-exclusive use shipment by about 13%. Under NCT, the maximum temperatures for the CV and the O-Rings are 236 ºF and 228 °F, respectively, which are less than the corresponding 321 °F and 302 ºF of the original 9977-96 SARP results.

Under HAC conditions, the maximum component temperatures for the CV and the O-Rings are less than the values reported in 9977-96 SARP as a result of less thermal loading with the Type 4 contents.

The Maximum Normal Operating Pressure (MNOP) for the 9977-96 Addendum 3 is based on the configuration using the SGQ-SC2. The maximum normal operating pressure (under NCT) is 117.3 psia (102.6 psig), which includes contributions from decomposition gases of plastic bag material, decomposition from 4.7 kg HDPE, and helium generation from the radioactive decay of the contents. The assumptions for the calculations [2, 10] are reasonable, and a safety factor for the MNOP of about 7 is demonstrated. Although the calculation of the containment vessel pressure under hypothetical accident conditions (HAC) reported in the addendum is incomplete because the gas generation from HDPE was not included, the Staff estimated that the pressure in the CV under HAC should be less than 137 psig, which results a safety margin of about 5.8.

Findings
Based on the review of the statements and representations in the Submittal, the Staff has concluded that the packaging design has been adequately described to meet the requirements of 10 CFR 71.

Conditions of Approval
The Staff has concluded that no additional thermally-related conditions of approval need to be added to the existing CoC for the approval of this request.

Chapter 4: Containment

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Containment information provided in Chapter 4 of the Submittal.

Details of the items reviewed are noted above in Chapter 1. The results of the containment review are discussed below.

The proposed addition of the Type 4 Contents to the Model 9977-96 Package SARP does not increase the impact loading on the containment vessel, the temperatures that must be sustained, or the pressure that must be contained. Therefore, package containment *leaktight* performance (in accordance with ANSI Standard N-14.5[12]), as documented in the existing Model 9977-96 Package SARP, is still valid for the Type 4 Contents addition.

Findings
Based on the review of the statements and representations in the Submittal, the Staff has concluded that the packaging design has been adequately described to meet the requirements of 10 CFR 71.

Conditions of Approval
The Staff has concluded that no additional containment-related conditions of approval need to be added to the existing CoC for the approval of this request.

Chapter 5: Shielding Evaluation

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Shielding Evaluation information provided in Chapter 5 of the Submittal.

Details of the items reviewed are noted above in Chapter 1. The results of the shielding review are discussed below.

Shielding Evaluation
This TRR covers the review of Content Type 4, material contents from the LANL Off-Site Source Recovery Project. The applicant has provided external dose rates for the these sources placed inside the lead shielded container (SC-1) for the gamma sources and the high density polyethylene shielded container (SC-2) for the neutron emitters that, in turn, is placed inside the 6-inch containment vessel of the Model 9977-96 packaging. The neutron and gamma source terms for these materials were estimated using the standard accepted codes.

The applicant performed calculations to estimate the external dose rates using 1 gram of each material. The bounding dose rate was determined to be at the surface of the package, which has a regulatory limit of 200 mrem/h. The applicant scaled down the mass such that the bounding dose rate at the surface is 190 mrem/h, thus allowing an additional 5% margin to the regulatory limit. The Staff performed confirmatory calculations, and, for the most part, matched the applicant's estimates of the maximum shippable mass for each isotope. In some cases, the applicant's estimates were lower than the Staff's estimates, while in others, the reverse was true. In the interest of conservatism in these estimates, it was recommended by the Staff that the smaller amount of the two estimates be taken as the bounding mass for each isotope. This was accepted by the applicant, and the set of bounding mass limits were included as the bounding set of mass limits for the Type 4 contents listed in Table A.1.1 of the Addendum. The Table below specifically outlines these limits.

Isotope	Recommended Maximum Mass Limit [g]	Recommended Maximum Activity [Ci]
Co-60	1.0E-04	0.11
Cs-137	1.0E-01	8.70
Ir-192	3.8E-03	35.00
Sr-90/Y-90	1.0E+00	281.80
Ra-226	2.0E-01	0.20
Am-241	1.0e+00	3.43
Cf-252	6.7E-06	0.0036
Cm-244	1.0E+00	80.90
Pu-238	2.0E-01	3.42
Pu-239	6.6E+01	4.09

Findings
Based on the review of the statements and representations in the Submittal, and independent estimates by the Staff, the Staff has concluded that the packaging design has been adequately described to meet the external radiation requirements of 10 CFR 71 for Content Type 4, provided the individual mass limit of each isotope provided in the above Table is not exceeded, and provided that the appropriate shielded containers are used — SC-1 for the gamma sources, and SC-2 for the neutron sources. Combinations of isotopes are permitted provided the individual mass limit of each isotope is not exceeded, and the overall external dose rates meet the regulatory limits.

The applicant has not evaluated, nor has the applicant provided, any mass limits for the Content Type 4 with the Engineered Container (EC-1) alone, without the use of either SC-1 or SC-2.

Conditions of Approval
The CoC must contain the restriction that the Type 4 Contents be bounded by the individual mass limits prescribed in Table A.1.1 of the Addendum, and that the appropriate shielded containers are used, i.e., SC-1 for the gamma sources, and SC-2 for the neutron sources. Combinations are permitted, with the contents placed in the appropriate shielded container, provided the individual mass limits of each isotope is not exceeded and that all external radiation dose rate limits comply with 10 CFR 71 requirements governing non-exclusive shipments.

The values of the individual mass limits prescribed in Table A.1.1 of the Addendum must also be corrected to be equivalent with the values listed above before the revised CoC is issued.

Chapter 6: Criticality Evaluation

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Criticality Evaluation information provided in Chapter 6 of the Submittal.

Details of the items reviewed are listed above in the introduction to Chapter 1. The results of the criticality review are discussed below.

Criticality Evaluation
It is specified in Chapter 1 that the Type 4 Contents are composed of Pu-238, Pu-239, Am-241, Cm-244, a trace amount of Cf-252, and a few other radioactive isotopes, such as Sr-90, Ra-226, Cs-137, Co-60, and Ir-192. The total amount of Type 4 Contents is less than 80 grams. In addition, the Type 4 contents excludes a few fissile materials, e.g., Cf-251,Cf-249, Am-242m, Cm-247, Cm-245, and Cm-243, which are restricted to trace amounts (less than 1000 ppm, or 0.1 gram), because they have critical masses much less than those of uranium and plutonium isotopes.[2] Also worth noting is that 0.1 grams of special actinides can be bounded by 9 grams of equivalent Pu-239 (conservatively assuming all special actinides are composed of Cf-251, which is most reactive — the subcritical mass limit of Cf-251 is 5 grams, and the subcritical mass limit of Pu-239 is 450 grams, so Eq. Pu is 0.1*450/5 = 9 grams).

The criticality evaluation for the Type 4 Contents was performed by using the bounding calculations for Type 2 Content envelope.[13] The Type 2 contents are limited to 100 grams of fissile and non-fissile materials, as is shown in Table A.1.1 of the Submittal. The Type 2 analyses were performed with the most widely used fissile isotopes, e.g., U-235, U-233, Pu-239, and Pu-241. Therefore, the bounding fissile content for Type 4 analyses is less, compared to those for Type 2, and is, therefore, conservative.

Light elements and impurities are limited to 50 grams for the Type 4 content envelope (identical to Type 2). The analyses were performed with a very conservative assumption that all light elements and impurities are beryllium (Be). For criticality analyses, Be will bound any combination of light elements and impurities as a moderator and/or reflector. The use of 100 grams of polyethylene material to represent a few plastic bags for contamination control is also conservative.

Single Package Analysis — These analyses were performed using a sphere of fissile material reflected by Be or Poly inside the 6CV with or without a SGC-SC2 polyethylene shielded container. It is noted that no calculations with lead shielded or tungsten shielded containers in the 6CV were provided, and also the polyethylene shielding container model is quite different from the actual container. A simple independent scoping calculation with Pu-239 enclosed with 12" of poly, lead or tungsten demonstrates that the reactivity with the polyethylene container will bound those from the other two types of shielding container. The polyethylene container model is judged to be conservative. It is also noted that beryllium, as a form of reflector, is more effective compared to mixing with the fissile material. This was demonstrated in the evaluation of the SARP for the Model 9975 Package.[14, 15, 16, 17]

A similar trend is also observed in this evaluation, as is shown in Table A.6.12 of the Submittal. Reactivity differences between four different types of fissile material are significant. However, only one type of fissile material is present in the Type 4 contents. All k_{eff} results are less than 0.3, as is shown in Table A.6.12 of the Submittal, which shows that there is a significant reactivity margin available, as the k_{safe} value is 0.931.

Single package solution calculations were provided without using any type of shielded container, which is conservative. The maximum reactivity ($k_{eff} = 0.57$) was near the Pu-239 concentration of 0.03 g/cc (see Table A.6.13 in the Submittal). The concentration of Pu-238 near the concentration of 0.03 g/cc will produce minimum critical mass,[18] assuming a spherical, homogeneous mixture of water with Pu-239. It is noted that the subcritical mass limit for Pu-239 in solution is 450 grams, as is shown in ANSI/ANS 8.1.[19]

NCT Analyses — The NCT analyses were performed with infinite number of 9977-96 shipping containers. All k_{eff} results are less than 0.3, similar to those for the single package analyses. It shows that there is a significant reactivity margin available for the NCT scenarios, as the k_{safe} value is 0.931. The interaction between shipping containers is minimal, as demonstrated for the Model 9975 Package[14–17] and the Model 9978 Package.[20] The drum for the Model 9977-96 Package has the same dimensions as the drum for the Model 9975 Package, and, therefore, the separation distances among fissile materials are very similar between the two types of drums. It may also be noted that the subcritical mass limit for Pu-239 metal is 5.0 kg.

HAC Analyses — The HAC analyses were performed with infinite number of 9977-96 shipping containers. A very conservative 2-cluster model was used. This model is conservative and very similar to those employed earlier for the Model 9975 and Model 9978 Packages. The HAC model correctly considered the damaged configuration using the fire and drop test data. All k_{eff}

results are less than 0.36. It shows that there is a significant reactivity margin available for HAC scenarios, as the k_{safe} value is 0.931.

Code Verification and Validation — Scale code verification and validation were properly performed and documented. The k_{safe} value of 0.931 is similar to the values employed for the Model 9975 and Model 9978 Packages.

Therefore, it is concluded that the Type 4 Content envelope for 9977-96 shipping container remains subcritical under the Single Package, NCT and HAC scenarios.

Findings
Based on the review of the statements and representations in the Submittal, the Staff has concluded that the packaging design and bounding analyses have been adequately described to meet the requirements of 10 CFR 71.

Conditions of Approval
The Staff has concluded that no additional criticality-related conditions of approval need to be added to the CoC for the approval of this application. The Staff has further concluded that the new Type 4 Content envelope for the 9977-96 Shipping Container can be shipped with a CSI of 0.0.

Chapter 7: Package Operations

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Package Operations information provided in Chapter 7 of the Submittal.

Details of the items reviewed are listed above in the introduction to Chapter 1. The results of the Package Operations review are discussed below.

Findings
Usually, the information provided in the Package Operations Chapter, i.e., Chapter 7 of the Submittal, would be included, automatically, in the CoC, as a Condition of Approval. In this case, however, the Staff has noted that the applicant has elected to make the shipper/operator responsible for all Packaging Operations decisions. As was noted previously in Sections 1 and 5 of this TRR, the Staff is not in agreement with this approach, as it is not in keeping with the long-established guidance set forth in the DOE's Packaging Review Guide[8] (the PRG), and/or the NRC's Reg. Guide 7.9.[9] More importantly, it should also be noted that by turning all Packaging Operation decisions over to the shipper/operator, it is not clear that the Model 9977-96 Package with the Type 4 Contents can be operated in a manner that is both safe and consistent.

The Staff recommends, therefore, that the procedures outlined in Chapter 7 of the current Submittal be rejected in their entirety and be replaced. The revised procedures should be unambiguous, specifying exactly which types of sources go into which type of shielded container. An example of the suggested revisions is provided below.

For the Type 4 Sources that are γ-emitters, the maximum mass/maximum activity limits are as defined in the following Table:

Isotope	Maximum Mass Limit [g]	Maximum Activity Limit [Ci]
Co-60	1.0×10^{-4}	0.11
Cs-137	1.0×10^{-1}	8.70
Ir-192	3.8×10^{-3}	35.00
Sr-90/Y-90	1.0	281.80
Ra-226	2.0×10^{-1}	0.20

The γ-sources listed above will be placed in the Lead Shielded Container, i.e., the SGQ-SC1 Container, which will then, in turn, be placed into the 6CV in accordance with the allowable content configuration, defined explicitly in Table A.1.2 of the Submittal. (Note: The Tungsten Shielded Container, i.e., the SGQ-SC3 Container, may be used as an acceptable substitute for the Lead Shielded Container, i.e., the SGQ-SC1 Container, provided that there are no changes to the maximum mass/maximum activity limits defined above, and provided that the maximum heat output of the contents is still limited to 6 Watts.)

For the Type 4 Sources that are neutron-emitters, the maximum mass/maximum activity limits are as defined in the Table, below:

Isotope	Maximum Mass Limit (g)	Maximum Activity Limit (Ci)
Am-241	1.0	3.43
Cf-252	6.7×10^{-6}	0.0036
Cm-244	1.0	80.90
Pu-238	2.0×10^{-1}	3.42
Pu-239	$6.6 \times 10^{+1}$	4.09

The neutron-sources listed above will be placed in the HDPE Shielded Container, i.e., the SGQ-SC2 Container, which will then, in turn, be placed into the 6CV in accordance with the allowable content configuration defined explicitly in Table A.1.2 of the Submittal. (Note: No substitutions are permitted for the SGQ-SC2 Container Configuration.)

The Staff recommends that the EC1 Container Configuration not be authorized for use at this time due to the absence of shielding calculations and content mass limits for the configuration in the Submittal.

Conditions of Approval

The Staff has concluded that the Package Operations procedures, as they are currently outlined in the Submittal, will need to be revised before they can be incorporated into the current CoC. An example of the suggested revisions is provided above.

Chapter 8: Acceptance Tests and Maintenance Program

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Acceptance Tests and Maintenance Program information provided in Chapter 8 of the Submittal.

Details of the items reviewed are noted above in Chapter 1. The results of the acceptance tests and maintenance review are discussed below.

The addition of the Type 4 Contents does not affect the Acceptance Testing of the packaging, nor does it affect the Maintenance Program requirements. Therefore, the package acceptance testing and basic maintenance program requirements, documented in the existing Model 9977-96 Package SARP, remain valid.

As is also noted by the applicant,

> "The Small Gram Quantity Shielded Containers perform a function integral to the Package Safety performance and its compliance with the Code of Federal Regulations. As such, the Shielded Containers have required Quality ("Q") dimensional inspections listed in Addendum Appendix 8.2, which are documented per SARP Table 9.7, with the documentation issued to the Design Authority for retention."

Findings
Based on the review of the statements and representations in the Submittal, the Staff has concluded that the packaging design has been adequately described to meet the operational requirements specified in 10 CFR 71.

Conditions of Approval
The Staff has concluded that the following additional condition of approval needs to be added to the existing CoC for the approval of this request:

- The documentation packages for the Q items, numbered as 17–30, in Table A.App.8.2.1 must be supplied by the Site directing fabrication, to Savannah River National Laboratory as the Design Authority/Design Agency.

Chapter 9: Quality Assurance

This TRR covers the Staff's findings regarding the review of the Submittal. This section covers the review of the Quality Assurance (QA) program description and packaging-specific QA requirements provided in Chapter 9 of the Submittal.

Details of the items reviewed are noted above in Chapter 1. The results of the quality assurance review are discussed below.

The Submittal describes that the QA Program for the Model 9977-96 Packaging is documented in the *SARP for the Model 9977-96 Packaging*.[2] Chapter 9 of the Submittal contains a revised Q-list adding the two Shielded Containers, one Engineered Container, two Spacers, and Cup. The Staff concurs that the addition of the sources, two Shielded Containers, one Engineered Container, two Spacers, and Cup do not affect the QA program as stated in Chapter 9 of the existing SARP, and that Chapter 9 of the existing SARP contains a reasonably up-to-date description of the applicant's QA program and packaging-specific QA requirements.

Findings
Based on review of the statements and representations in the Submittal, the Staff concludes that the QA program has been adequately described and meets the QA requirements of 10 CFR 71,

Subpart H. Packaging-specific requirements are adequate to assure that the packaging is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation.

Conditions of Approval

The Staff has concluded that no additional QA-related conditions of approval need to be added to the existing CoC for the approval of this request.

References

[1] *Model 9977, Safety Analysis Report For Packaging, Addendum 3, Justification for Small Gram Quantity Contents*, S-SARA-G-00006, Revision 4, Savannah River Packaging Technology, Savannah River National Laboratory, Aiken, South Carolina, March 2010.

[2] *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory, Washington Savannah River Company, Savannah River Site, Aiken, SC (August 2007).

[3] *Safety Analysis Report for Packaging, Model 9977, Addendum, Justification for DNDO Contents*, S-SARA-G-00003, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory, Savannah River Nuclear Solutions, Savannah River Site, Aiken, SC (October 2008).

[4] *Safety Analysis Report for Packaging, Model 9977, Addendum, Justification for Metal Contents*, S-SARA-G-00005, Revision 1, Savannah River Packaging Technology, Savannah River National Laboratory, Savannah River Nuclear Solutions, Savannah River Site, Aiken, SC, December 16, 2008.

[5] USA/9977/B(M)F-96 (DOE), *United States Department of Energy Certificate of Compliance for Radioactive Materials Packages, Model 9977*, Revision 4, United States Department of Energy, Washington, DC, expires October 31, 2012.

[6] USA/9977/B(M)F-96 (DOE-S/T-1), *United States Department of Energy Certificate of Compliance for Radioactive Materials Packages, Model 9977*, Revision 0, United States Department of Energy, Washington, DC, expires December 31, 2013.

[7] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments*; Final Rule, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[8] U.S. Department of Energy, DOE Packaging Certification Program, *Packaging Review Guide for Reviewing Safety Analysis Reports for Packaging*, Lawrence Livermore National Laboratory, UCID-21218, Revision 3, February 2008.

[9] U.S. Nuclear Regulatory Commission, Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*, Revision 2 (March 2005).

[10] McKeel, C.A., *Design and Evaluation of a Shielded Carrier for Use in 6 inch Containment Vessel of the 9977 Package*, M-CLC-A-00371, Revision 1, Savannah River Nuclear Solutions (2009).

[11] Gupta, N.K., *Thermal Analysis of 9977 Package for Small Gram Quantity (SGQ) Transport of Nuclear Materials*, M-COC-A-00368, Revision 1, October 30, 2009.

[12] American National Standards Institute, *American National Standard for Radioactive Materials-Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, New York, New York, 10036.

[13] *Nuclear Criticality Safety Evaluation 9977 Shipping Package Analysis for Small Gram Quantity SARP Addendum*, N-NCS-A-00021, Revision 0, July 2009. (Note: According to the References cited for Chapter 6, this is actually Appendix A.6.1 for the Submittal.)

[14] *Safety Analysis Report for Packaging, Model 9975*, WSRC-SA-2002-00008, Revision 0, Radioactive Materials Packaging Technology, Savannah River Technology Center, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC (December 2003).

[15] *Safety Analysis Report for Packaging, Model 9975, Addendum 1, Justification for Modified Contents Parameters*, S-SARA-G-00001, Revision 0 (April 2005).

[16] *Safety Analysis Report for Packaging, Model 9975, Addendum 2, Justification for U233 Content Envelope*, S-SARA-G-00002, Revision 1, Savannah River Packaging Technology, Savannah River National Laboratory, Washington Savannah River Company, Savannah River Site, Aiken, SC (May 2008).

[17] *Safety Analysis Report for Packaging, Model 9975*, S-SARP-G-00003, Revision 0, Savannah River Packaging Technology, Savannah River National Laboratory, Washington Savannah River Company, Savannah River Site, Aiken, SC (January 2008).

[18] Nuclear Criticality Safety Guide, LA-12808, 1996.

[19] ANSI/ANS-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

[20] *Safety Analysis Report for Packaging (SARP), Model 9978 B(M)F-96*, S-SARP-G-00002, Revision 1, Savannah River Packaging Technology, Savannah River National Laboratory, Washington Savannah River Company, Savannah River Site, Aiken, SC (March 2009).