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

February 19, 2019

Mr. Troy Hedger
President
Alpha-Omega Services, Inc.
9156 Rose Street
P.O. Box 789
Bellflower, CA 90706

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9316, REVISION NO. 9, FOR THE MODEL NOS. AOS-25A, AOS-50A, AOS-100A, AOS-100B, AND AOS-100A-S PACKAGES

Dear Mr. Hedger:

As requested by your application dated July 20, 2018, supplemented December 10, 2018 and January 25, 2019, enclosed is Certificate of Compliance No. 9316, Revision No. 9, for the Model Nos. AOS-25A, AOS-50A, AOS-100A, AOS-100B, and AOS-100A-S packages. The U.S. Nuclear Regulatory Commission safety evaluation report is also enclosed.

The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of Title 49 of the *Code of Federal Regulations* (49 CFR) 173.471. Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471.

If you have any questions regarding this certificate, please contact Pierre Saverot of my staff at (301) 415-7505.

Sincerely,

/RA/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9316
EPID - L-2018-LLA-0201

Enclosures:
1. Certificate of Compliance No. 9316, Rev. No. 9
2. Safety Evaluation Report
3. Registered Users

Upon removal of Enclosure 3, this document is uncontrolled

cc w/encls. 1&2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy, c/o L. Gelder

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9316, REVISION NO. 9, FOR THE MODEL NOS. AOS-25A, AOS-50A, AOS-100A, AOS-100B, AND AOS-100A-S PACKAGES, DOCUMENT DATE: February 19, 2019

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SAFETY EVALUATION REPORT

Model Nos. AOS-025A, AOS-50A, AOS-100A, AOS-100B, and AOS-100A-S Packages
Certificate of Compliance No. 9316
Revision No. 9

SUMMARY

By application dated July 20, 2018, Alpha-Omega Services, Inc. (AOS) submitted an amendment request for the Model Nos. AOS-25A, AOS-50A, AOS-100A, AOS-100B, and AOS-100A-S packages.

Alpha-Omega Services, Inc., made changes related to the Shipping Cage fastening screws, because of the occurrence of galling, and reduced the torque for these screws from 62.5 ft-lb to 37 ft-lb, which is still appropriate for 1/2"-13 UNC lubricated screws. AOS added an optional material, Nitronic 60 per American Society of Mechanical Engineers (ASME) SA-193, ASTM A193 Grade B8S (UNS S21800) for the Shipping Cage screws, along with an optional shipping cage closure. AOS corrected an error in the description of the package geometry, stating that the actual configuration of the AOS-100A has vertical flanges that straddle the shipping cage with the screws passing horizontally through the flanges.

The applicant proposed the following changes to the package:

- adding content limits for exclusive use shipments with an enclosure (a shipping cage) in the 100A and 100A-S package models,
- including low energy gamma-emitting and low energy 'pure' beta-emitting nuclides to the 100A and 100A-S models' contents,
- removing CoC Condition 15, which requires the package to be shipped as exclusive use when loaded with contents requiring the use of axial shielding and spacer plates,
- allowing in each package model the shipment of mixtures of the nuclides that are approved contents for that package model.

The applicant modified the shielding analysis to address these proposed content changes.

Additional changes were made to the package contents and shielding analysis as a result of this review. These changes include the removal of holmium-166 from the contents and changes in content limits to account for using the correct package surface for surface and 1-meter radiation level calculations, which also allowed for the removal of Condition 15 from the CoC.

Additionally, AOS changed the quality classification of the elastomeric seal used for shipments of Special Form material to a quality classification "C". New drawings 1205E9712 and 105E9719 were also submitted.

Several clarifications were made in the application as supplemented, based on staff's input, to properly address: (i) the applicability of the radiation level limits in 10 CFR 71.47(a) and limits in 49 CFR 173.428 for package being shipped/received as an empty package, (ii) the criteria for determination when a repair or replacement of cask lid and cask lid sealing surface is required, (iii) the leak rate testing for use of a new elastomeric lid seal, (iv) the use of the liner, axial shield plates, or cavity spacer, (v) the requirements regarding verification and removal of security seals both on the shipping cage and impact limiters, (vi) the appropriateness of acceptance criteria for the optional additional test for shielding acceptance.

Based on the statements and representation in the application, as supplemented, and the conditions listed below, the staff concludes that the proposed changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

STRUCTURAL AND MATERIALS EVALUATION

The family of AOS transportation packages consists of the AOS-025A, AOS-050A, AOS-100A, AOS-100B, and AOS-100A-S. The "A" designation refers to a tungsten shield, and the "B" designation refers to a carbon steel shield. The "S" designation on the AOS-100A-S means that the cask is double ended and there is a lid on each end of the cask. All of the packages have the same geometric shape, with the AOS-025 and AOS-050 having all dimensions scaled down to 25% and 50% (respectively) of the dimensions of the AOS-100 package.

The general design of the packages was not modified. AOS proposed a change to the Type 410 stainless steel fastening screws that secure the shipping cage to the transportation pallet on the AOS-100A. The shipping cage is a five-sided metal structure covered with an expandable metal mesh or screen material that keeps unauthorized persons away from the transport package surfaces during transport.

Alpha-Omega Services, Inc., reported having issues with galling on the fastener screws. Thread galling, also known as "cold welding," occurs during installation when pressure and friction cause bolt threads to seize to the threads of a nut or tapped hole. To reduce the occurrence of galling, AOS proposed a reduction of the torque from 62.5 ft-lb to 37 ft-lb for ½"-13 UNC lubricated screws. In addition, AOS added an optional material for the fasteners, Nitronic 60 per ASME SA-193, American Society for Testing and Materials (ASTM) A193 Grade B8s (UNS S21800). Section 2.5.3.1.4 of the application was modified to evaluate this optional material.

The staff notes that the reduction on torque from dry to lubricated screws is around 41 percent, which the staff finds to be a representative range for reduction of torque due to lubrication. The staff agrees that lubrication is an acceptable and recommended method to mitigate galling. The staff notes that the drawing does not specify lubricant material, such as lubricants containing graphite, and speed of installation of the bolt (slower revolutions per minute), which are two factors that could affect galling mitigation. The staff finds the proposed change of torque specification from 62.5 ft-lb to 37 ft-lb for ½"-13 UNC lubricated screws acceptable because the functionality of the fasteners is not impacted by the change in specification.

The staff reviewed the material specifications for the screws and compared the materials specifications used in the analysis of the new screws to ensure they are consistent. In addition, the staff reviewed the analysis of the loads for the new fastener material; the analysis calculates the overturning moment of the shipping cage mass using an acceleration of 10 times the acceleration due to gravity and the resultant overturning moment is converted to a force that it is assumed to be supported through shear forces by four fasteners. The staff finds that the assumptions used are conservative and that the results show a good margin of safety.

Drawing 105E9711, Revision K, provides additional information as part of the requested changes. The applicant added an optional fastener material, Nitronic 60, ASME SA-193/ASTM A193, Grade B8S (UNS S21800) stainless steel for the previously approved method of fastening the shipping cage structural components of the packaging. In addition, the applicant added an optional shipping cage closure design that replaces the Keenserts with a flanged nut, hex socket screws, hex nut, lock washer and modified hex bolt constructed of ASME and ASTM material Type 304, 316, 410 or Nitronic 60 stainless steel. The staff notes that Nitronic 60 is an austenitic stainless steel requested as an option to ferritic steel fasteners used to secure the shipping cage to the pallet. The shipping cage is a five-sided metal structure, with the pallet creating a base for tying down the package during transport. Each side is covered with an expandable metal mesh or screen material that keeps unauthorized persons away from the transport package surfaces.

The staff reviewed ASME Section II, Part D properties, associated ASTM specifications and various independent literature. The staff found that Nitronic 60/ASTM A193, Grade B8S, is considered anti-galling and wear resistant. The staff compared various material properties such as tensile, yield, elongation, hardness and found the Nitronic 60 to be comparable if not exceeding the Type 410 ferritic steel. Corrosion resistance of Nitronic 60 falls between that of Type 304 and 316 austenitic stainless steel. In addition, the staff reviewed the material specifications for optional shipping cage closure design and notes that these standards have been previously approved and are used in various other transportation packages. The material components are used as part of the shipping cage structural assembly and are not part of the transportation package containment boundary.

Based on the above discussion, the staff finds the use of the requested Nitronic 60 and optional closure design materials to be acceptable.

SHIELDING EVALUATION

The objective of this review is to verify that the package design, with the proposed modifications, meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

For the shielding review, the staff evaluated the capability of the different AOS package models' shielding features to provide adequate protection against direct radiation from its contents, as modified.

This review includes the staff's evaluation of the proposed changes to the package's allowable contents and the package's shielding features for each package model, including the analyzed radiation levels on the package surfaces and the required distances away from the package during transportation for both normal conditions of transport and hypothetical accident conditions.

1. Description of Shielding Design

1.1 Packaging Design Features

The applicant did not propose to change any of the packaging features that affect shielding. The applicant did propose to add an analysis to show the effect of the cage surrounding the package on the tests for normal conditions of transport and to show the cage survived these tests. The purpose of this evaluation was to support the proposed removal of CoC Condition 15 that the package be shipped as exclusive use for contents that require the use of axial shielding and spacer plates. In other words, the applicant proposed to show the cage survived the

normal conditions tests in order to use the cage to show compliance with the non-exclusive use radiation level limits in 10 CFR 71.47(a).

Until now, the applicant has relied upon the cage being present to allow for the package surface on all package models to be defined at the outermost dimensions of the impact limiters, as deformed by the normal conditions of transport tests. The cage, however, is not a part of the package for any of the package models. CoC Condition 15 had been imposed particularly for contents for which it was considered that radiation levels on the actual package surface where the impact limiters do not cover the cask body would exceed the limits in 10 CFR 71.47(a). This condition affects the AOS-050A and AOS-100A, AOS-100A-S, and AOS-100B package models, for which several inches of the cask surface are exposed. Thus, the package surface in these areas is the exposed cask surface. Additionally, each package model's axial surface changes where there are what the applicant refers to as 'notches,' or recesses, in the impact limiters. These 'notches' are areas of reduced impact limiter thickness that are several inches in diameter and up to several inches deep; the dimensions vary with package model. Thus, the package surface in these areas is actually closer to the cask body than the applicant had previously analyzed.

Since the cage is not a part of the package, it cannot be credited for determining the package surface for the purpose of demonstrating compliance with the radiation level limits in 10 CFR 71.47(a) for non-exclusive use shipments. Therefore, to address the recognition of the proper package surfaces, the applicant made changes to the shielding analyses in the application and other aspects of the application that are affected by the definition of the package surfaces.

Furthermore, the analysis to show the cage survived the normal conditions of transport test became no longer necessary and was therefore not reviewed nor evaluated by the staff. While the cage is always used to transport the package, it is not credited for evaluations for 10 CFR 71.47(a) radiation level limits.

The staff's review of the changes and its findings related to shielding are described in the sections that follow.

1.2 Summary Table of Maximum Radiation Levels

The package contents are defined at quantities that do not exceed the regulatory radiation level limits in 10 CFR 71.47 with a self-imposed 10 percent margin and that also do not exceed the decay heat limit for each package model. Thus, for conditions normally incident to transport, the package's maximum radiation levels are at 90 percent of the regulatory limits in 10 CFR 71.47(a) for non-exclusive use for all package models for the contents quantities specified in Tables 1-2 and 1-2a of the application and for mixtures of radionuclides as set in Appendix 5.5.5 of the application. Some of the contents are more limited by package decay heat limits. For the AOS-100A and 100A-S models, with contents specified in Table 1-2b of the application and for mixtures of radionuclides as set in Appendix 5.5.7 of the application, the maximum radiation levels are 90 percent of the 10 CFR 71.47(b) limits for exclusive use, with application of the 200 mrem/hr surface limit at the surface defined by the impact limiters as they were deformed by the normal conditions of transport tests. The staff found that Tables 5-5 and 5-40 of the application show the maximum radiation levels for hypothetical accident conditions have significant margins to the regulatory limit in 10 CFR 71.51(a)(2).

Table 5-40 of the application also shows the radiation levels for locations at 2 meters from the trailer surface, or the projected planes of the trailer surface, for the trailer sides and the back end of the trailer, for the exclusive use shipments of the AOS-100A and 100A-S models. The package is placed on an 8 foot wide trailer at a position no closer than 4 feet from the back end

of the trailer (measured from the package's axis). The table also shows the radiation levels for the driver cab location, which is set at 20 feet from the package's axis. The staff found the table shows significant margins to the limits for the driver cab and locations 2 meters from the trailer sides and back end.

2. Radiation Source

The applicant defined the allowable contents, including the proposed additions to the allowable contents in Tables 1-2, 1-2a, and 1-2b of the application. The applicant also included, in Section 1.2.2 of the application, low energy gamma-emitting and 'pure' beta-emitting radionuclides as contents for the AOS-100A and 100A-S package models. These latter radionuclides are defined as those for which the energy emissions do not exceed 0.3 MeV. For beta-emitting nuclides, this limit applies to the maximum energy of the beta, not its average energy. This energy limit also applies to any radioactive progeny of these nuclides. Other than the low energy 'pure' beta-emitters, the contents are all gamma sources. While the nuclides listed in Tables 1-2, 1-2a, and 1-2b of the application emit other radiations, such as betas, the applicant ignored these radiations and their secondary particles on the basis that they are not able to penetrate the cask's shielding. Radionuclides that emit neutrons are not allowed.

The staff reviewed the contents descriptions in Section 1.2.2 and Tables 1-2, 1-2a, and 1-2b of the application and found they are consistent with the contents' descriptions and quantities evaluated in the shielding chapter of the application. In addition, for mixtures of nuclides in the package contents, the applicant developed the method for determining the contents of a particular shipment meet regulatory limits in Appendices 5.5.5 and 5.5.7 of the application. Based on its review, the staff found this method is captured in the Package Operations chapter of the application (see Appendix 7.5.1 of the application) and is appropriately referenced in the contents' descriptions in Section 1.2.2 of the application.

Section 5.2 of the application states that charged particles emitted by the nuclides listed in the tables stated above and their secondary particles (bremsstrahlung from betas and electrons) were not considered. The staff reviewed the charged particle emissions from these nuclides and found the applicant's decision to be acceptable for all of the nuclides except holmium-166. The staff's finding is based on:

- the maximum energies for the betas was low, or
- the intensity of the betas was very low, or
- the approximation in Cember's *Introduction to Health Physics*, 3rd edition showed that the gamma source from bremsstrahlung would be a small fraction of the gamma source emitted by the nuclide at gamma energies that were approximately equal to or greater than the maximum beta energies.

For the holmium-166 nuclide, neither of the first two characteristics applied, and it was not clear that third characteristic would apply either. Instead, it seemed that the bremsstrahlung could be a significant contributor to package radiation levels for this nuclide. Therefore, since this contribution to package radiation levels for this nuclide was not evaluated, the nuclide was removed from the allowable contents lists (in the certificate of compliance) for the 025A and 050A package models. The 100A, 100A-S, and 100B models do not currently include this nuclide in their allowable contents.

Tables 1-2 and 1-2a of the application provide quantity limits for compliance with the limits in 10 CFR 71.47(a) for non-exclusive use. Based on the addition of separate quantity limits for exclusive use for the AOS-100A and 100A-S models and the objective of removing CoC

Condition 15, the applicant reduced the quantities of those contents for which it was necessary to ensure compliance with the non-exclusive use limits for the correct package surfaces and at one meter from those surfaces. This affected the maximum quantity of the source labeled cobalt-60-C, which is a cobalt-60 source that requires use of axial shielding and spacer plates, in the AOS-100A and 100A-S models and the iridium-192 content limits for the AOS-050A model. The impact on the iridium-192 limits is a result of the increase in radiation levels from both the iridium-192 source and the iridium-194 source, which may be present (an impurity) in shipments of iridium-192 (see Table 1-2a of the application). The increase resulted from adjusting the surface radiation level calculation to be at the exposed cask surface between the impact limiters for the iridium-194 and to be at the impact limiter surface within the 'notch' area for the iridium-192. With the iridium-194 source quantity kept constant, the iridium-192 source quantity had to be reduced. The changes to the cobalt-60-C quantity limits resulted from adjusting the package surface radiation level calculations to be at the exposed cask surface between the impact limiters. An additional effect was the applicant's modeling the cobalt-60-C source as a point source for the non-exclusive use (i.e., 10 CFR 71.47(a)) calculations instead of as a volumetric source as had been done previously.

The staff reviewed the new values for these quantities and performed a calculation with MicroShield Version 11.22X to confirm the iridium-194 radiation levels. With the calculation, the staff confirmed the radiation levels that were determined for the iridium-194 and that the applicant calculated radiation levels for determining quantity limits at the correct package surface. Thus, the staff has assurance that the new quantity limits for the iridium-192 and the cobalt-60-C contents are appropriate for meeting the non-exclusive use radiation levels in the respective package models. The quantity limits did not change for some of the contents because the bounding location of the source remained in a cavity location away from the exposed cask side and away from the 'notches' in the impact limiters and the maximum radiation levels were also at locations away from the impact limiter 'notches' and exposed cask side or were large enough to remain the maximum radiation level even with the adjustments in the affected areas. With these changes in the contents limits, the staff found that CoC Condition 15 is no longer needed and can be removed from the CoC. Also, the quantity limits in Tables 1-2 and 1-2a of the application are now solely for non-exclusive use shipments.

For determining decay heat of the analyzed nuclides, the applicant used the decay library origen.rev03.decay.data from the ORIGEN code in the SCALE 6.1 code system. The applicant also stated that this library is to be used for determining the decay heat from low energy gamma-emitting nuclides and beta-emitting nuclides shipped in the AOS-100A and 100A-S models. The ORIGEN code has been developed to perform spent fuel irradiation and decay calculations and also has the capabilities to perform decay analyses of specific radionuclides, providing radiation spectra and decay heat results. The decay library used by the applicant is relied upon for these ORIGEN calculations. This code and data are widely used and well validated. Further, the staff used the code Radiological Toolbox, Version 3.0.0, which uses ICRP-107 (International Commission on Radiological Protection, "Nuclear Decay Data for Dosimetric Calculations," ICRP Publication 107, 2008) data to check some of the decay heat values and found them to be consistent. Thus, based on these considerations, the staff found the applicant's selected data source to be acceptable.

3. Shielding Model

The applicant's shielding model did not change in terms of dimensions or materials from the models used for previous revisions of the CoC, except for the source configuration for the cobalt-60-C contents for the non-exclusive use quantity calculations. Other than that, the only thing that changed was the location of surface and 1 meter radiation level detectors for those nuclides for which the applicant determined that the bounding radiation levels would occur

where the cask body was exposed between the impact limiters or in the ‘notch’ areas of the impact limiters (see Appendix 5.5.8 of the application). The detectors were placed on the cask body surface and at 1 meter from that surface to determine the appropriate radiation levels to use to determine the new quantity limits for those nuclides and the radiation levels to use for packages with mixtures of nuclides that included these nuclides.

In previous analyses of the cobalt-60-C content, the applicant used a volumetric source for the allowable quantity and radiation level calculations. In the analyses to support this request to revise the CoC, the applicant changed the analysis approach for non-exclusive use quantity and radiation level calculations, using a point source for the cobalt-60-C content instead of a volumetric source. Thus, the maximum specific activity for this source was removed from the non-exclusive use tables in the application (Tables 1-2, 5-15, and 5-35).

The exclusive use analysis for this content still uses the volumetric source; so, the maximum specific activity limit for the content still applies to and appears in the tables for the exclusive use cobalt-60-C quantity limits and per curie radiation levels (Table 1-2b and the Appendix 5.5.7 analysis and results). The specific activity limit, 350 curies/gram, is based on the source’s activity being at least 19,000 curies and the source volumes used in Appendix 5.5.4 of the application. Since the allowable activity limit for this content in an exclusive use shipment exceeds 19,000 curies, the specific activity limit is appropriate for the cobalt-60-C content under exclusive use.

The staff performed a calculation with MicroShield for the iridium-194 contents. The staff used a source strength of 1 curie in a shielding configuration that represented the AOS-050A package model. For the location of the cask body surface, the staff’s calculation produced a result (mrem/hr/curie) that was consistent with the value used by the applicant. Based on this outcome, the staff has assurance that the applicant used the appropriate detector locations in its models.

For the cobalt-60-C source, for the exclusive use limits in Tables 1-2b, 5-39, and 5-41 of the application, the applicant used the radiation level results (mrem/hr/curie) for the package side that is at the outer surface of the impact limiters instead of the package surface, which is the exposed cask surface, between the impact limiters. The staff found this to be acceptable because the evaluation is for exclusive-use with an enclosure.

For the AOS package, the cage is the enclosure, and the package is always shipped with the enclosure. So, the actual shipping configuration is consistent with the analysis for exclusive use with an enclosure. The applicant also conservatively applied the regulatory limit, less the applicant’s self-imposed 10% margin (i.e., 180 mrem/hr), for the surface of the enclosure to the surface evaluated for the cobalt-60-C contents. The actual regulatory limit for the package surface is 1000 mrem/hr in this case.

The staff found the applicant’s analysis indicates the actual surface radiation level would be below this limit with significant margin. Were the shipping configuration to not include an enclosure, then these tables would need to use the radiation level results on the exposed cask surface and the 180 mrem/hr limit would apply at that surface.

The discussion in the application that relates to Table 5-39 and the notes for Table 5-41 of the application explains that the exclusive use limits in 10 CFR 71.47(b)(1)(i)~(iii) apply to the cobalt-60-C. However, that requirement applies to all of the contents listed in Tables 5-39 and 5-41 of the application. The intent is to explain the basis for the per curie radiation value used for the cobalt-60-C content at the external surface not being the value at the package surface like it is with the other package contents in those tables. That basis being that the package is

shipped with an enclosure, and the per curie radiation value used in these tables is at a location that is consistent with that configuration.

4. Shielding Evaluation

4.1 Radiation Level Calculations

The applicant continued to use the same computer code, MCNP6, and data that the staff reviewed previously and found to be acceptable. This computer code and data are still appropriate for the proposed changes in the current application since the evaluated nuclides are the same and the package models are the same as have been previously reviewed. The MCNP6 code is capable of evaluating individual gamma energy lines. Thus, evaluations of package radiation levels for individual gamma energies using the code are appropriate and acceptable. The applicant continues to use the dose conversion factors from the 1977 version of ANSI/ANS 6.1.1, which is the standard and version that the staff has stated in its review guidance to be acceptable. Thus, the staff found the use of this code and data to be acceptable for this application.

4.2 Evaluation of Low Energy Gamma-Emitters and Beta-Emitters

The staff reviewed the applicant's evaluation in Appendix 5.5.6 of the application for low energy gamma-emitting nuclides and beta-emitting nuclides in the AOS-100A and 100A-S models. These contents are not allowed in the other package models. The evaluation focuses on gamma radiation. The applicant's evaluation demonstrates that for gamma energies that do not exceed 0.3 MeV, the contribution to package radiation levels is negligible for quantities that result in 400 watts decay heat (the package model's decay heat limit). The applicant's evaluation considered only one gamma per decay at the analyzed energies (0.3 MeV and 0.2 MeV).

Many nuclides, however, emit multiple gammas per decay with different energies. Thus, the staff did a separate consideration of multiple gammas at different energies being emitted per decay to verify the applicant's evaluation and conclusions. This evaluation included having both a 0.2 MeV and a 0.3 MeV gamma emitted per decay. The maximum curie content to reach 400 watts decay heat decreased from the case of a single 0.3 MeV gamma per decay; however, the total emitted gammas increased.

However, the data in Table 5-36 of the application indicate that the radiation level contribution from 0.2 MeV gammas is much less than from 0.3 MeV gammas. So, the radiation levels from such a case are still negligible. Based on this separate evaluation, the staff found that the as long as all gamma emissions from a nuclide do not exceed 0.3 MeV, the contributions to radiation levels will be negligible for the analyzed package models.

For beta-emitting nuclides, the beta radiation is not of concern, given the significant shielding provided by the package components (either steel and tungsten or all steel); however, bremsstrahlung can be a concern for sufficiently high beta energies. The bremsstrahlung source may be estimated using simple methods such as is given in Cember's *Introduction to Health Physics*, 3rd Edition (pages 129-131). The staff used this estimation method for strontium-90 together with its progeny yttrium-90 and evaluated the resulting source in MicroShield.

The results indicated that these nuclides would contribute significantly to package radiation levels. The staff found a similar result for the nuclide phosphorus-32. Thus, the applicant also limited these contents to only those that emit betas of maximum energies that do not exceed 0.3

MeV. This energy limit also applies to any radioactive progeny of these nuclides. In this way, the evaluation for low energy gamma-emitting nuclides applies to these nuclides as well and demonstrates that these low energy beta-emitting nuclides will contribute negligibly to package radiation levels and are limited, together with their progeny, if any, by the package decay heat limit. In evaluating and confirming compliance with the package decay heat limit, the contribution from these nuclides' progeny, if any, must also be accounted for.

4.3 Multi-nuclide Contents Evaluation Method

The applicant devised a new method for users of the AOS package models to determine if their contents for a particular shipment meet the regulatory requirements and the conditions of the CoC (e.g., package decay heat limits) when the contents include a mixture of the allowed radionuclides. Appendix 5.5.5 of the application includes the method for non-exclusive use and applies to all AOS package models. Appendix 5.5.7 of the application includes the method for exclusive use and applies only to the AOS-100A and 100A-S models since only those models have been evaluated for exclusive use.

The staff reviewed the method in each appendix. The method uses the radiation level on a per curie basis for each analyzed nuclide at each required location. The method also uses the decay heat, in watts per curie, for each analyzed nuclide. The staff identified that the appropriate external surfaces are used for each appendix's method (i.e., the package surfaces for Appendix 5.5.5 of the application and the deformed impact limiter's surface for Appendix 5.5.7). For most nuclides in the AOS 100A and A-S models, that surface is the same because the package surface where the radiation level is maximized for those nuclides is in a location covered by the impact limiter. For both the non-exclusive use and the exclusive use analyses, the 180 mrem/hr limit is applied to the external surface. The staff notes that for Appendix 5.5.7 of the application, the application of this limit to this surface is conservative since the package is shipped in an enclosure (i.e., the cage) and the enclosure's dimensions in the package drawings show its surface to be at a greater distance from the package body than the impact limiter's surface. It also ensures that there will be significant margin to the actual package surface limit in the regulations for a package shipped exclusive use in an enclosure (1000 mrem/hr).

Since some nuclides require axial shield or spacer plates, mixtures containing these nuclides will also require the use of these shield and spacer plates. The method ensures that the summation of the product of each nuclide's activity and its per curie radiation level for the package model does not exceed 90 percent of the regulatory limit for the applicable locations (e.g., package surface, transport index, 1 meter from cask). The method also ensures the summation of the product of each nuclide's activity and its per curie decay heat does not exceed the decay heat limit of the package model being used. The contents are limited by the most restrictive limit, whether a radiation level limit or the package decay heat limit. Since the method ensures regulatory limits will not be exceeded and includes a 10 percent margin to those limits for radiation levels and the values used in the method are derived from the analyses for the quantity limits in Tables 1-2, 1-2a, and 1-2b of the application, the staff found the method to be acceptable.

For both exclusive use and non-exclusive use for the AOS-100A and 100A-S models, the method accounts for contributions to decay heat from low energy gamma-emitting and beta-emitting nuclides, including their progeny, if any. Based on the evaluation described in Section 5.4.2 of this SER, the staff found that to be acceptable.

4.4 Evaluation of Package Surface Changes

As noted previously, for compliance with the non-exclusive use radiation limits in 10 CFR 71.47(a), the package radiation levels must be evaluated on the package's surface and at one meter from that surface. For the AOS package models, that surface changes, with the 'notches' in each model's impact limiters and the exposed cask body between impact limiters for all but the AOS-025A model. This had not previously been evaluated because of the cage that is used with the package, though it is not itself part of the package, and because CoC Condition 15 required that shipment be made as exclusive use for some contents. However, as previously stated, crediting something that is not part of the package to define the package surface is not appropriate for confirming or demonstrating compliance with non-exclusive use radiation level limits.

The applicant, therefore, evaluated the impacts of correcting the package surface and adjusting the transport index calculation point in relation to the package surface in these areas. The evaluation is in Appendix 5.5.8 of the application. The applicant found that for most nuclides in the 050A, the 100A, 100A-S, and 100B package models the highest radiation levels occurred at surface and 1-meter locations away from the exposed cask surface (e.g., at the top corner of the package and 1 meter from the top corner of the package) and the source located away from the exposed cask surface (e.g., the source in the top corner of the cavity). So, the current analysis and quantity limits were still valid for these nuclides. For the two that did require changes (iridium-194 in the 050A model and cobalt-60-C in the 100A and 100A-S models), the applicant used MCNP6 to reanalyze the nuclides with detectors at the appropriate locations and adjusted the radiation levels (mrem/hr/curie) and maximum quantities for these nuclides as described previously.

The applicant did an analysis to determine how much of a difference the change in package surface would make for radiation levels at the affected areas and came up with a threshold factor that bounded the calculated amount of change. The calculated amount for the exposed cask side was up to a factor of about 2.64 (i.e., the radiation level in that area would be 2.64 times larger than at the surface equal to the impact limiter's outer surface location). Independently, the staff got similar factors. The applicant used a bounding factor of 3 to evaluate when package side radiation levels and maximum quantity limits needed to be reanalyzed for a nuclide.

For the transport index location, the applicant stated it used the radiation levels from the hypothetical accident conditions analysis, which neglects the impact limiters altogether, to evaluate the 1-meter radiation levels for the exposed cask side under normal conditions of transport. The radiation levels for hypothetical accident conditions were calculated at 1 meter from the bare cask, which is the 1-meter, or transport index, location for the exposed cask surface under normal conditions of transport. Thus, the applicant used radiation levels from the hypothetical accident conditions analysis and a factor of 1 to determine whether quantity limits needed to be reanalyzed based on the 1-meter radiation levels. The staff had identified that the radiation levels for the transport index location increased by up to 30 percent when adjusted for the correct package surface.

The staff found that the description of the evaluation is unclear and the values in the related tables in the application, Tables 5-43 through 5-45 of the application, do not seem to clearly agree with the applicant's description of the analysis. Based on Figures 5-11 and 5-12 of the application and the information in Tables 5-43 through 5-45 of the application, the staff determined that the per curie radiation values for the nuclides that are shown in those tables were calculated at the surface that is equivalent to the distance of the impact limiter surface from the actual package surface (i.e., the exposed cask surface) for the side surface values.

The values in the tables for the 1-m transport index values were calculated at 1 meter from the cask surface. The reported values are the maximum values for the detector locations where the cask surface is exposed between the impact limiters on the 050A and the 100 package models.

With consideration of its own evaluation of the estimates in radiation level changes, the staff reviewed the applicant's analysis. The staff notes that only one nuclide in one of the package models had a factor (referred to by the applicant as a safety factor) of less than 2 for radiation levels at 1 meter from the package surface. The same nuclide, the cobalt-60-C content in the 100A and A-S package model, also had a factor for the surface radiation levels that was less than 3. With recalculation of the radiation level per curie of source, the surface radiation level became the limiting condition. There is now significant margin to the transport index limit. Thus, even with the adjustment indicated by the staff's evaluation, the increased radiation level would still have significant margin to the transport index limit. Based on these results, the staff found that the applicant adequately identified those nuclides for which quantity limits and radiation level values (on a per curie basis) needed to be changed to account for the correct package surface on the package's side and the applicant made adequate changes.

The applicant evaluated the effects of the impact limiter 'notches' in a similar way. Based on that approach the applicant identified that the package surface radiation level in the 'notch' area would increase by about 25% to 30% and that the radiation level for the transport index would increase by about 3% to 9%. The amount of increase depended upon the package model; the increase was largest for the 100A, 100A-S, and 100B models. The staff independently calculated increases that were similar to the applicant's results.

For the 025A package model, the location of the maximum radiation levels is on the package side. The maximum radiation levels for each nuclide in the 025A model is larger than the originally calculated radiation levels at the impact limiter's 'notch' area by more than the predicted increase due to correcting the analysis for the package's actual surface at the 'notch.' Thus, no adjustments were necessary for the allowable quantities and calculated radiation levels for the nuclides allowed in this package model.

For the 050A model the location of maximum radiation levels was on the package surface near the impact limiter 'notch.' The increase in 'notch' area radiation levels resulting from the package surface correction resulted in maximum radiation levels now being in the 'notch' area. Thus, the applicant modified the maximum quantities and radiation levels on a per curie basis for these nuclides. The applicant indicated that the effects of the surface location correction on radiation levels at 1 meter from the package surface would not exceed the margin to the transport index limit for the 025A and 050A models (i.e., the margin to the 9 mrem/hr limit imposed by the applicant). So, the applicant did not re-evaluate the radiation levels at 1 meter from the package surface.

While the staff found that the transport index limit would not be exceeded, the staff also found that this means that the location of the maximum 1-meter radiation level (i.e., the maximum estimated transport index) and the location of the maximum radiation level at 1 meter from the package is now not clearly identified in the application. Thus, future revisions to the certificate of compliance that could affect these radiation levels (e.g., contents changes) may need to include an evaluation of the location and values of the maximum radiation level at 1 meter from the package surface.

For the 100A, 100A-S, and 100B models, the most restrictive limit is the transport index limit (i.e., the applicant-imposed 9 mrem/hr limit). The margin to the limit for the package surface radiation levels exceeds the predicted increase in surface radiation levels due to correcting the

package surface location in the impact limiter 'notch' area. Thus, the applicant did not re-evaluate the package surface radiation levels. The staff's findings with regard to the 1-meter radiation levels for the 025A and 050A models therefore apply to the surface radiation levels for the 100A, 100A-S, and 100B models.

For the 100A and 100A-S models the maximum calculated transport index is located above the surfaces near the 'notch.' For the 100B model, the maximum is located above the 'notch.' The predicted increase does not exceed 9%. This is within the applicant's imposed 10% reduction in the limit from the regulatory value (i.e., the applicant has limited the allowable radiation levels to only 90% of the regulatory limit or 9 mrem/hr). However, that margin (from the applicant's limit to the regulatory limit) is currently used to offset other uncertainties in the applicant's analysis. These uncertainties include the effects of using nominal package component dimensions versus dimensions at the tolerances that maximize radiation levels and the impacts of the curvature of the impact limiters. Also, for nuclides that are not limited by decay heat, the allowable quantities have been rounded up from the analysis results. It is not clear how much of the margin is needed to compensate for these uncertainties.

Given these considerations and for consistency with the analyses of the other package models and nuclide mixtures, the applicant used MCNP6 to reanalyze the nuclides affected by the predicted radiation level increase and provide new maximum quantities and 1-meter radiation levels on a per curie basis. Several of the nuclides in the 100A and 100A-S models and all of the nuclides in the 100B model were affected. The staff reviewed the changes, and based on that review, found that the radiation levels and quantity limits are based on the appropriate package surfaces, with the considerations as described above.

The staff also notes that the tests for normal conditions of transport should have been applied to the appropriate package surfaces in the impact limiter ‘notch’ area and the exposed cask side area. The only one of these tests that could impact either of these areas would be the penetration test. The impact of this test was analyzed in the application (see Section 2.6.10) and resulted in a very small deformation and would result in a negligible increase in package radiation levels.

Given the materials of the cask body, any deformation due to the penetration test on the exposed cask surface would be even less than for the impact limiter. Thus, the staff found that the evaluation for compliance with the 10 CFR 71.47 radiation level limits is acceptable. For hypothetical accident conditions tests, the impact limiters are not credited in any way. All radiation levels are calculated at 1 meter from the cask surface. Any impact of these tests is expected to be small (less than 0.25 inches). Given the significant margin to the radiation level limits for these conditions, the staff considers the impact of the cask deformation on radiation levels to be negligible and will not affect compliance with the regulatory limits.

4.5 Exclusive Use Content Limits in the 100A and 100A-S Models

The applicant proposed content limits for exclusive use shipments for only the 100A and 100A-S models. The analysis method for the proposed content limits for shipment by exclusive use is similar to that used for the content limits for non-exclusive use shipments. The differences lie in the radiation level locations for which there are regulatory limits for conditions normally incident to transportation. The package is shipped with a cage and a pallet, which serve as an enclosure. Thus, the limits in 10 CFR 71.47(b) for exclusive use shipment in a closed vehicle apply.

As described in Sections 5.3 and 5.4.3 of this SER, the applicant applied the radiation level limit for the enclosure surface to the surface of the package, or, for the cobalt-60-C contents, the

plane of the impact limiter surface over the exposed cask surface between the impact limiters. As described in those earlier sections of the SER, the staff found that to be acceptable and conservative.

The applicant then added radiation level calculations for locations 2 meters from the projected planes of the trailer sides and rear and for the driver cab location. The distances were based on the vehicle and trailer characteristics described in Section 5.1.2 of this SER. The per cure radiation levels that resulted from this analysis for the different nuclides are used for determining the acceptability of multi-nuclide mixtures, as described in Section 5.4.3 of this SER. Based on the staff's review, described in these earlier SER sections, the consistency of the analyzed locations with the characteristics of the vehicle described earlier, and that the package is shipped in an enclosure, the staff found the applicant's analysis and proposed contents to be acceptable.

5 Evaluation Findings

The staff reviewed the proposed changes to include quantity limits for exclusive use shipments with the 100A and 100A-S package models, allow shipment of mixtures of the approved nuclide contents in all package models, and allow low energy gamma-emitting and low energy 'pure' beta-emitting nuclides as contents in the 100A and 100A-S package models. The staff found these changes to be acceptable, as described in Sections 5.4.5, 5.4.3, and 5.4.2 and 5.4.3 of this SER, respectively.

In addition, the staff found that removal of the holmium-166 contents was appropriate due to the shielding analysis not considering the beta radiation emitted by the nuclide and the resulting bremsstrahlung as described in Section 5.2 above. The staff also found that with the corrections regarding the package and package surface and the adjustments for that in the non-exclusive use content quantity limits, CoC Condition 15 is no longer needed and the structural analysis with the cage that was initially submitted with the revision request was no longer needed, as described in Sections 5.2 and 5.1.1 above, respectively. As noted in Section 5.1.1 above, that structural analysis was not reviewed or evaluated.

Therefore, based on its review of the information and representations provided in the application, and the staff's independent calculations, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements and radiation level limits in 10 CFR Part 71.

OPERATING PROCEDURES EVALUATION

Several changes and clarifications were made to the operating procedures in response to staff's requests. In particular, the applicant clarified that (i) the compliance requirements for the verification of the radiation and external contamination levels, (ii) the transport package's bottom surface is not accessible until the transport package is removed from the pallet, (iii) for the 1-m TI dose rate, the 1-m distance is from the transport package surface, not the shipping cage surface, (iv) the activity limits listed in the CoC represent maximum conditions. Thus, the user shall refer to the guidance in Appendix 7.5.1 of the application for the shipment of multiple isotopes, or isotopes that emit only low-energy gamma/beta emitters (that is, all emissions, including those from their progeny, are ≤ 0.3 MeV).

The staff reviewed the package operations descriptions in Chapter 7 of the application. This included a review of the changes the applicant proposed to account for the additional contents, including the method in the new Appendix 7.5.1 of the application for package users to

determine the contents of their particular shipment meets the CoC for both shipments of individual radionuclides and shipments of multiple radionuclides.

The staff also reviewed the operations descriptions to ensure consistency of the operations descriptions and sequencing with the definition of the package. Since the cage and pallet, used in transporting the package, are not a part of the package, operations such as those to confirm compliance of package radiation levels and contamination levels with regulatory limits required some changes to sequences and descriptions of operations. The applicant made these changes as well as changes to ensure correct regulatory requirements were cited for operations meant to confirm regulatory compliance.

For some operations, this included unique approaches for loaded packages prior to shipment, such as measuring radiation levels at specified distances from the cask surface since some of the package surfaces (e.g., the lower impact limiter surfaces) may not be accessible for radiation measurements at the point in package operations when the package is completely assembled. The specified distances from the cask are set to the minimum thickness of the impact limiter around the cask, including the ‘notch’ area in the impact limiter for the axial measurements. The distance is zero for areas where the cask surface is the package surface. The staff found this approach to be acceptable because it ensures that radiation levels are measured on or at the locations of the actual package surface. The approach is also conservative for determining the package’s surface radiation levels for areas where the impact limiter are present in the package’s transport configuration because it ignores the impact limiters’ contribution to shielding in those areas.

The package operations include completion of radiological surveys when the package is in the transport configuration. The staff expects that the transport index is determined at this time. In the transport configuration, the cage is in place and the package is on the pallet; however, the cage doesn’t really provide any shielding of the package. Given these considerations and that the regulatory requirement is that a determination is made that radiation levels are met, the staff found the applicant’s approach to be acceptable. While radiation measurements are the best way to make this determination, the language of the regulation is flexible enough to allow for other determination methods.

For surface contamination levels, the operations include surveys of the cask surface and the impact limiter surfaces (both inner and outer impact limiter surfaces). The staff found this approach to contamination level verification is appropriate because it ensures all package surfaces including where the package surface is the cask surface for the larger models, will be checked. Plus, the operations include the option of keeping the impact limiter on the pallet or removing it with the rest of the package during package receipt and unloading.

For the verifications of package radiation and contamination levels described in the procedures for receipt of an empty package (Section 7.1.1.2.d of the application) and receipt of a loaded package (Section 7.2.1.g of the application), the package operations for both include a note for the operation to check radiation and contamination levels. Not all package surfaces are accessible for confirming the radiation and contamination levels by measurement at the operations step where the operations descriptions state compliance with the respective regulatory requirements is to be verified. Thus, the note that was added to these steps describes the steps in the operations sequence when measurements for the remaining package surfaces would be done. The staff reviewed the note and found it to be acceptable because, with the note, the operations descriptions enable verification that radiation and contamination levels for all of the package surfaces comply with the regulatory requirements, consistent with the descriptions in those regulatory requirements. The staff found that the package operations are appropriate for ensuring that the radiation levels on the package surface and at the

distances specified in the regulations and the surface contamination limits will be met for all package surfaces.

Based on its review of package operations, the staff found that the applicant has modified the package operations to provide sufficiently clear descriptions of the process for determining acceptability of a particular shipment's contents and perform other needed actions; to ensure the operations involving the package's surfaces are performed on the package's surfaces, consistent with the definition of the package in the application; and to ensure compliance with appropriate regulatory requirements.

ACCEPTANCE TESTS AND MAINTENANCE PROGRAMS EVALUATION

As part of the review, the staff also looked at the descriptions of the acceptance tests and maintenance programs to ensure they were still acceptable and appropriate for the package, as modified in the proposed CoC revision.

Changes were also made to the maintenance procedures regarding the need for a 100% UT examination, as well as dimensional and density checks of the shielding material, the criterion used to evaluate the effect of material defects (such as voids and cracks) i.e., the dose rate cannot exceed 1.5 times (1.5x) the mean measurable dose rate, and the verification of the cask shielding material integrity.

The staff identified concerns with the shielding acceptance tests described Section 8.1.6 of the application. The packaging was deemed acceptable if a dose rate measurement did not exceed 1.5 times the mean measurable dose rate on the packaging surface. This acceptance criterion did not appear to relate to the package design described in the package drawings. The applicant modified the acceptance criteria to better ensure the package shielding conformed to the package drawings. However, the new criterion is somewhat ambiguous as it only states that the results of package dose rate measurements should 'closely match' the values derived from shielding calculations. It is not clear what would be considered to meet that criterion and the tie to the drawings is not explicit.

The staff notes, however, that the shielding materials are fabricated per industry standards listed in the package drawings and other sections of the acceptance tests include visual and dimensional checks to verify conformity with the package drawings.

Furthermore, the tests in Section 8.1.6, if conducted, are in addition to these other acceptance tests. Given these other considerations, the staff found that, since the tests in Section 8.1.6 are not the only acceptance tests for the shielding and that the other tests ensure conformance with the package drawings, the tests and acceptance criterion in Section 8.1.6 are acceptable. However, a modification of the acceptance criterion would be necessary if the tests in Section 8.1.6 were to become either the primary acceptance tests or an alternative to the other acceptance tests.

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

CONDITIONS

The conditions specified in the Certificate of Compliance No. 9316 have been revised as indicated below:

Item No. 3(b) was revised to include the latest Revision H-7 of the application, dated January 25, 2019.

Condition No. 5(a)(2) was revised for editing purposes. A reference to Table 5 of the certificate was added.

Condition No. 5(a)(3) was revised to include the packaging drawings latest revision numbers.

Condition No. 5(b)(2) was revised to (i) prohibit neutron emitting nuclides as authorized contents, (ii) remove the specific activity limit for Co-60-C as no longer needed for the non-exclusive use based content limits, (iii) specify that only Nb-95 resulting from the decay of Zr-95 is allowed, (iv) update Tables 3 and 4 by removing Ho-166 from the authorized contents, (v) add Table 5 specifying activity limits for isotopes when shipped in the Model Nos AOS-100A/A-S as exclusive contents.

Condition No. 7 was revised to include Table 5.

Condition No. 15 on exclusive use shipments for contents utilizing axial shielding and spacer plates was removed since it was no longer needed. Contents that require exclusive use shipments are already specified in Table 5 of the CoC. Contents specified in Tables 3 and 4 can be shipped non-exclusive use regardless of whether axial shielding or spacer plates are required or not.

Condition No. 16, now renumbered 15, now extends the use of revision 6 of the certificate for approximately one year.

The expiration date of the certificate was not modified.

The References section of the certificate was updated to include Revision H-7 of the application.

CONCLUSION

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the design of the Model Nos. AOS-25A, AOS-50A, AOS-100A, AOS-100B, and AOS-100A-S packages has been adequately described and evaluated.

The staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9316, Revision No. 9,
on February 19, 2019