Safety Evaluation Report for the Model 9516 Package

Safety Analysis Report for Packaging (R1033-0062-ES, Revision 1, November 30, 2009)

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SUMMARY

By a letter dated June 26, 2008, the Idaho National Laboratory (INL) submitted an application request to amend the U.S. Department of Energy (DOE) Certificate of Compliance (CoC) USA/9516/B(U)F-85 for the Model 9516 package, previously known as the Mound 1 kW package. A Safety Analysis Report for Packaging (SARP) was submitted, along with the application request, to provide documentation that the design, with modifications, satisfies the "-96" requirements per Title 10 of the Code of Federal Regulations (10 CFR) 71.19(e)] and the International Atomic Energy Agency Safety Standards Series No. TS-R-1. Before the submittal of the application request, a pre-application meeting was held at Argonne National Laboratory on September 18, 2008. The meeting was hosted by the Argonne SARP Review Group, on behalf of the DOE Packaging Certification Program (PCP), Office of Packaging and Transportation (EM-45).

On October 14, 2008, the DOE PCP issued twenty-eight (28) Q1 questions on the various chapters in the SARP. The applicant provided written responses to Q1s and proposed changes in Rev. b of the SARP for Chapters 1, 2, 3, and 9 on January 27, 2009. The applicant provided written responses to the remaining Q1s and revised Chapters 5 and 6 in Rev. c of the SARP on March 19, 2009. A conference call was held on May 13, 2009, to discuss all written responses to Q1s and the proposed changes in the Rev. b and c SARP that resulted in a few minor, additional changes. These changes were incorporated into the Rev. 0 SARP, which was submitted on June 18, 2009. The DOE PCP staff has verified that all changes and revisions in the SARP are acceptable.

Subsequent to the submission of the Rev. 0 SARP, it was determined that additional analysis and text change were needed to adequately support inclusion of the ISO cargo container for shipment by sea and land. This configuration is required to support the continuation of international shipments and the approval of Revision 0 was held in lieu of the additional analysis and Revision 1 submittal for approval. The Revision 1 SARP, dated October 2009, included the text that describes the ISO cargo container configuration in Chapter 1, Introduction; additional shielding analysis and supporting text in Chapter 5, Shielding Evaluation; and text to describe loading restrictions for the ISO container in Chapter 7, Package Operations. The DOE PCP staff has verified that all changes and revisions in the Rev. 1 SARP are acceptable

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, as summarized in this Safety Evaluation Report (SER), the design and performance of the 9516 package is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

<u>Reference</u>

Model 9516 Package, Safety Analysis Report for Packaging, prepared by Energy Solutions, Inc., for the Idaho National Laboratory and the DOE, Office of Nuclear Energy, Vols. 1 and 2, R1033-0062-ES, Revision 1, dated October 2009.

1. GENERAL INFORMATION AND DRAWINGS

1.1 Packaging Description

The 9516 packaging consists of a welded, stainless-steel containment vessel placed within a cylindrical stainless-steel cask that is, in turn, housed in a meshed personnel shield. The 9516 packaging is designed for transport of up to 500 W of plutonium dioxide (PuO_2) heat source material in any solid form (e.g., powder, pellets, granules). The containment boundary is provided by the welded containment vessel that is housed within the cask during transport.

The personnel shield (cage) provides protection from heat and radiation from the package contents and serves as an impact limiter for the cask. The overall height is 35.25 inches, including the lid, and the overall base is 30.75 by 30.75 inches. The personnel shield is of welded construction and is fabricated of Type 304 stainless steel, except for the structural tubes at the base, which are constructed of ASTM A-500, Grade B carbon steel. The wire mesh completely encloses the cask during shipment while permitting heat to escape. The top and side cover weldments are removed during cask loading and unloading.

The 304L stainless-steel cask is designed to provide confinement of the containment vessel and contents during Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). The outside diameter and overall height of the cylindrical cask are 9.5 inches and 19.5 inches, respectively; the inner diameter and the cavity height of the cask are 6.5 inches and 16.5 inches, respectively. A 1.5-inches-thick cask lid is attached to the cask body by eight (8) bolts. The base plate of the cask is welded to the cask body and secured to the personnel shield by six (6) bolts. The lid of the cask is sealed with a Helicoflex metal O-ring, and a stainless-steel, shoulder-style eyebolt is used to lift the lid and place the cask into the personnel shield.

1.1.1 Containment Boundary

Containment Vessel

The 304L stainless-steel containment vessel (CV) has dimensions of 6.38 inches (outside diameter), 16.25 inches (height), and 0.12 inch (minimum wall thickness). The base plate and cover plate of the CV are 0.5 inch thick. To assist in loading and unloading of the CVs into the cask, a 3/8-16 threaded hole is tapped into the cover. A 0.06-inch-wide groove is provided on the CV to assist in its opening with a pipe cutter or by other means.

The CV is designed in accordance with American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsection NB. Except for the top-end closure joint location and the final top-end closure welding requirements, the fabrication, examination, and testing of CV meets all of the pertinent requirements of the ASME BPVC, Section III, Division 1, Subsection NB. Strict compliance to Subsection NB cannot be achieved for the top-end closure because of the hazards of the content and the requirement for remote assembly of the CV. To ensure that safety and quality of the top-end closure are equivalent to that provided by the ASME BPVC, equivalent requirements and additional process controls and tests are provided for the closure.

The CV is sealed with a full-penetration weld. All CV welds are radiographed and helium leakage-rate checked to determine their acceptability. Drawings detailing the safety features of the containment system, including welding and inspection requirements for all components, are included in Appendix 1.3.2 of the SARP. The CV materials are in compliance with the ASME BPVC, Section III, Division 1, Subsection NB and the appropriate material specification specified in the ASME BPVC, Section II. A reconciliation analysis will be performed, if the material is purchased to a later edition of the Code to ensure all of the original requirements are met or exceeded. The maximum normal operating pressure (MNOP) for the CV is 37.6 psig.

The maximum gross weight of the 9516 package is 900 lb. The personnel shield weighs approximately 500 lb, and the empty cask weighs approximately 285 lb.

Drawings

The drawings that pertain to the 9516 package are listed in Table 1.1.

Table 1.1 List of Drawings Pertaining to the 9516 Package.

Drawing Title	Drawing No.
9516 Shipping Container	756179, 11 Sheets, Rev. 1
Cylinder, Product Can	756180, 2 Sheets, Rev. 0
Liner, 5.00 High	756181, 3 Sheets, Rev. 1
Liner, 5.75 High	756182, 3 Sheets, Rev. 1
Graphite Filler Block	756183, 1 Sheet, Rev. 0
Graphite Support Block for GPHS Module	756184, 1 Sheet, Rev. 0
Graphite Support Blocks for Product Cans	756185, 1 Sheet, Rev. 0
PuO_2 Powder Can Set	756186, 3 Sheets, Rev. 0
Cylinder, Product Can	756187, 2 Sheets, Rev. 0
Graphite Support Block for Product Cans	756188, 1 Sheet, Rev. 0
Containment Vessel, 16.25 High	756189, 2 Sheets, Rev. 01

1.2 Contents

The contents to be shipped in the 9516 package consist of PuO_2 in any solid form (e.g., powder, pellets, granules). The principal isotope in the PuO_2 is ^{238}Pu, which has an initial composition of 74–90 weight percent (wt.%) of the total plutonium in the mixture. As the initial ^{238}Pu weight percent is increased in a mixture of PuO_2, the ^{239}Pu and ^{241}Pu weight percents are reduced. The initial ^{239}Pu and ^{241}Pu could range between 23.9 and 7.9 wt.% for different mixtures. The plutonium that is ^{241}Pu will be less than 1 wt.% for all mixtures. The fissile isotopes of uranium, ^{235}U and ^{233}U, will only be present in trace amounts.

Almost all of the activity in the mixtures of PuO_2 is from the alpha decay of ^{238}Pu, which is the main decay heat source. The curie content of ^{238}Pu is directly proportional to the decay heat. Limiting the decay heat to 500 W in the package establishes the activity limit to be ≈15,930 Ci. The initial plutonium isotopic limits are shown in Table 1-1 of the SARP. The maximum neutron emission rate for the fueled clad assemblies is 12,000 n/s-g^{238}Pu. The maximum neutron emission rate for the PuO_2 powder is 18,000 n/s-g^{238}Pu. To satisfy the 10 CFR 71.47 dose rate limits for a single package containing fresh fuel, the total neutron emission rate must be kept below 1.58×10^7 n/s, which gives the maximum total mass of ^{238}Pu of 880 grams in a single package.

For a package containing fuel with a neutron emission rate (N_e) greater than 18,000 n/s-g^{238}Pu, but less than 36,000 n/s-g^{238}Pu, the total mass of ^{238}Pu is reduced according to the following formula:

$$\text{Mass of } ^{238}\text{Pu} = (1.58 \times 10^7) / (N_e),$$

as illustrated in Figure 5-7 of the SARP. Administrative controls will be placed on the loading arrangements to ensure that the maximum wattage is not exceeded.

There are six (6) shipping configurations for the contents in the 9516 package. These shipping configurations are briefly described below (detailed descriptions of the six shipping configurations and contents can be found in Section 1.2.2.1 of the SARP). Either one or two 304L stainless-steel liners are used as dunnage for positioning the contents inside the CV. Each liner has a base plate and a cover plate that are welded in place. A graphite support block fills the internal void volume of a liner and positions the payload. Excess end spacing in the CV is filled with graphite filler blocks. The dimensions of the graphite filler blocks are shown in Figure 1-6 of the SARP.

1. General Purpose Heat Source (GPHS) fueled clad assembly (FCA) – one or two PuO_2 fuel pellets encased in an iridium alloy capsule, with one or two FCAs and associated graphite support blocks and a graphite filler block, as necessary, per product can, and up to four product cans with threaded or welded lids are placed in a 5.75-inches-tall liner. Two of the 5.75-inches-tall liners can be placed into the CV with a graphite filler block.

2. GPHS graphite impact shell (GIS) – GPHSs placed within a GIS that is made of fine-weave pierced fabric (FWPF), with one GIS per product can, and a maximum of two product cans in a 5.75-inches-tall liner. Two of the 5.75-inches-tall liners can be placed into the CV with a graphite filler block.

3. GPHS module – a base component in the assembly of a radioisotope thermoelectric generator (RTG), which is placed in 5-inches-tall liners and is held in position by a graphite support block. Two of the 5-inches-tall liners can be placed into the CV with a graphite filler block.

4. Domestic PuO_2 powder – loose PuO_2 powder from domestic sources in a threaded product can, where the PuO_2 powder is placed in up to eight product cans, with a maximum of four product cans per 5.75-inches-tall liner. Two of the 5.75-inches-tall liners can be placed into the CV with a graphite filler block.

5. Russian PuO_2 powder – loose PuO_2 powder from Russian sources in a threaded product can, where the Russian PuO_2 is placed in a threaded ampoule, surrounded by a welded capsule, and then placed on a grade WDF felt cushion inside a Russian (welded) product can and sealed. Up to four Russian product cans, with a graphite support block, may be placed in a 5.75-inches-tall liner. Two of the 5.75-inches-tall liners can be placed into the CV with a graphite filler block.

6. Generic Contents – PuO_2 in any solid form (e.g., powder, pellets, granules) that meets the initial isotopic limits shown in Table 1-1 of the SARP and where the maximum neutron emission rate for a loaded CV does not exceed 1.587×10^7 neutrons/s. The total heat load of the contents must be limited to 500 W, which is ≈1,110 g of a combination of ^{233}U, ^{235}U, ^{238}Pu, ^{239}Pu, and ^{241}Pu isotopes. The generic contents are shipped in powder cans, product cans, or capsules, all of which are held in a liner with the appropriate graphite support block. The liner(s) is contained in the CV, and a graphite filler block(s) is used as a spacer to limit movement.

1.3 Criticality Safety Index

On the basis of the results of the criticality safety analysis presented in Chapter 6 of the SARP, the DOE PCP staff has confirmed that the criticality safety index (CSI) derived on the basis of the procedure in 10 CFR 71.59(b) is CSI = 0, which also satisfies the requirements of 10 CFR 71.59(c) for the exclusive use shipment.

1.4 Radiation Level and Transport Index

The external radiation level and transport index (TI) will be established by measurement at the time of shipment. The external radiation level must meet the 10 CFR 71.47 standards for exclusive use shipment.

1.5 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the general information (and drawings) presented in Chapter 1 of the SARP is acceptable. Evaluations of design and performance of the package for safety and regulatory compliance in structural, thermal, containment, shielding, criticality safety, operating procedures, acceptance tests and maintenance, and quality assurance are given in the remainder sections of this SER.

2. STRUCTURAL

2.1 Discussion

The 9516 packaging consists of a personnel shield, a cask, a CV, and liners for the payload contents. The cask is designed to provide confinement of the CV, which is not attached to the cask. It is important to ensure that the cask will not lose confinement and the CV remains in the cask under normal conditions of transport (NCT) and hypothetical accidents (HAC) specified in 10 CFR 71. All contents are shipped in cylindrical liners that function as dunnage material for positioning the contents inside the CV. Each liner has a 0.25-inches-diameter hole through the sidewall to prevent any pressure buildup inside the liner.

The cask, CV, and the liners of the 9516 packaging are fabricated from Type 304L stainless steel. The room-temperature yield strength and ultimate tensile strength for Type 304 L stainless steel are 34 and 87 ksi, respectively.

The cask lid is closed with eight (8) 1/2-inch 13 UNC bolts and lock washers after the CV is loaded. The bolts are fabricated from ASTM A-193 Grade B6 SS and are preloaded to 1,300–1,450 lb. The room-temperature yield strength and ultimate tensile strength for the bolt material are 85 and 110 ksi, respectively. Failure of the entire bolted closure would result in the loss of confinement for the CV.

2.2 Structural Evaluation

The objective of the structural evaluation is to verify that (1) the structural performance of the 9516 package is adequate under NCT and HAC, and (2) the design of the package meets the requirements and safety standards in 10 CFR 71. The structural evaluation examined the design of the major packaging components, including the cask, the cask closure bolts, the CV, and the payload liners of the 9516 package. The structural performance of the 9516 package is evaluated in the SARP by using both analyses and testing of full-scale, prototype packages. The finite-element analysis code DYNA3D was used to determine the worst orientation for the 30-foot free-drop tests under HAC. Testing of full-scale prototypes under HAC included a 30-foot bottom-down drop, a dynamic crush test, and a puncture test. For the crush and puncture tests, the cask was removed from the personnel shield for maximum damage. The acceptance criteria for the analyses are provided in the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 7.6 and Section III and Section VIII of the ASME BPVC.

The DOE PCP staff's confirmatory structural evaluation of the 9516 package used the general-purpose finite-element code ABAQUS and focused on the structural performance under HAC. (The structural performance of the 9516 package under NCT has been demonstrated in over 90 shipments of its predecessor Mound 1 kW package, since the original CoC-85 was issued in 1999.) For the 30-foot free drop under HAC, eleven (11) drop orientations of the package were analyzed, and the results confirmed the worst orientation being that of the bottom-down drop. In this section, the ABAQUS results are compared to the results of the HAC tests, where applicable, only for the worst-case, 30-foot bottom-down drop test, the dynamic crush test, and the puncture test. The confirmatory analysis results are not compared with the DYNA3D results in the SARP, because the necessary details of the DYNA3D analysis are not available. For example, the DYNA3D results did not identify the filtering frequency used for estimating the rigid-body acceleration, and the results did not include acceleration time history, maximum deformation in the packaging components, and their corresponding locations.

2.2.1 HAC 30-foot Bottom-down Drop Test

The DOE PCP staff's confirmatory analysis results for the 30-foot bottom-down drop test show that the calculated peak acceleration of 1,100 g compares well with the 1,084 g measured during the test and reported in the SARP. The analysis showed that the peak stress intensities are 69 and 60 ksi in the cask and the CV, respectively, which are below the allowable stress intensity for stainless steel under HAC. These stress intensities occurred at locations where the components have experienced localized plastic deformation.

The confirmatory analysis results also show that the peak dynamic load on the eight (8) cask closure bolts is 18,000 lb for the 30-foot bottom-down drop test. The tensile stress area of the 1/2-13 UNC bolts is 0.1419 in^2 (Shigley, Joseph Edward, Mechanical Engineering Design, New York: McGraw-Hill, 1977) so that the peak stress intensity in the bolts is up to 126 ksi, which is higher than the room-temperature tensile strength of 110 ksi. However, the bolts were modeled as elastic connector elements in the ABAQUS analyses and, therefore, do not exhibit the elastic-plastic deformation behavior of the bolt material (ASTM A-193, Grade B6). This bolt material has a yield strength of 85 ksi and an ultimate elongation of 16% (ASME BPVC, 2008a, Section II, Part A, SA-193/SA-193M, "Specifications for Alloy-Steel and Stainless Steel Bolting Materials for High-Temperature or High-Pressure Service and Other Special Purpose Applications," pp. 275–292).

Since the drop test is an energy-controlled event, the actual stress in the closure bolts may be estimated from the area under the nonlinear (elastic-plastic) stress-strain curve for the bolt material. The result shows that the peak stress intensity in the bolts would be reduced to about 86 ksi, which is only slightly above the bolt yield stress and well below the allowable stress limit (110 ksi) for the HAC 30-foot free drop test. Therefore, the bolts are not expected to fail, and it can be concluded that the cask will not lose confinement during the worst case, HAC 30-foot bottom-down drop test.

2.2.2 Crush and Puncture Tests

For the crush test, the DOE PCP staff's confirmatory analysis results show that the peak stress intensities in the cask body, cask lid, and CV are 70, 57, and 25 ksi, respectively, which are below the room-temperature allowable stress limits under the HAC crush test. For the puncture tests, the confirmatory analysis results show that the peak stress intensities in the cask body, cask lid, and CV are 70. 76, and 65 ksi, respectively, which are also below the allowable stress limits under the HAC puncture test.

The DOE PCP staff's confirmatory analysis results for the HAC crush test show that the peak load on the cask closure bolts is 5,000 lb, resulting in a peak stress intensity of 35 ksi in the bolt shanks, which is below the yield stress (85 ksi) of the bolt material. For the HAC puncture test, the peak dynamic load on the closure bolts is 45,000 lb, resulting in a peak stress intensity of 317 ksi, which is much higher than the tensile strength (110 ksi) of the bolt material. However, the puncture test is also an energy-controlled event. Since the plastic deformation of the bolt would absorb significantly more energy than the elastic deformation at the same stress level, the actual stress in the bolts would be lower. Using the area under the nonlinear (elastic-plastic) stress-strain curve for the bolt material shows that the peak stress intensity in the bolts would be about 90 ksi, which is well below the allowable stress limit (110 ksi) for the puncture test. The test results showed that only two of the eight bolts in the cask lid became loose during the tests, which agree with the confirmatory analyses results indicating plastic deformation of the bolts and not the failure of the bolts. Therefore, it can be concluded that the cask will not lose confinement during the HAC crush and puncture tests.

The liners experienced some localized plastic deformation during the puncture test, but the deformation does not affect the geometry significantly and has no adverse effect on the contents. In the DOE PCP staff's confirmatory analyses, the contents in the liners are conservatively modeled as stainless-steel spheres to concentrate the dynamic loads on the liners and the CV. The analyses confirmed that the liners can provide adequate support to the contents under the HAC crush and puncture tests.

The DOE PCP staff's confirmatory analysis results are in general agreement with the HAC test results reported in the SARP. The results show that the peak stress intensities in the cask, the closure bolts, and the CV are all below the allowable stress intensities and, therefore, have met the requirements in the ASME BPVC, Section III.

2.3 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the structural design and performance of the 9516 package presented in Chapter 2 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

3. THERMAL

3.1 Discussion

The 9516 package is designed for transportation of up to 500 W of PuO_2 heat source materials in any solid form. The thermal-related design features of the 9516 package (e.g., thermal properties, maximum temperature limits, maximum temperatures and pressures, and thermal stresses) are described in the SARP. Analyses and tests are used in the SARP to evaluate the packaging component temperatures under NCT and HAC described in 10 CFR 71.71(c) and 71.73(c)(4), respectively.

The 9516 package is designed for exclusive-use shipment only.

3.2 Material Thermal Properties and Temperature Limits

The thermal properties of materials are provided in the SARP for PuO_2 and the various packaging components: Type 304L stainless steel (cask, CV, liners, product cans); iridium (FCA cladding); carbon-based materials (graphite support and filler blocks, graphite felts, CBCF insulating sleeve around GIS assemblies in the GPHS modules, FWPF for the GPHS); and backfilled gases, including argon or helium for the CV at atmospheric pressure, and air, also at atmospheric pressure, for the cask cavity. The SARP has provided the references cited for the data. The listed property values are in agreement with the values found in the published technical reports, standards, test reports, or handbooks.

The maximum operating temperature limit for Type 304L stainless steel is 800°F under NCT. This limit applies to the cask body, CV, liners, the product, and the powder cans. The 800°F limit agrees with the datum (< 800°F) found in the ASME BPVC, Section II: Material Properties, page 324, 1998. Under HAC, the maximum operating temperature limit is 2,500°F, which is the melting point of 304L, for the liner, product can, and powder can. This limit is acceptable since none of these components serve the containment or confinement functions for the 9516 package. For the cask and the CV, Section 2.7.4 of the SARP shows that the peak temperatures of 1,393°F for the cask and 1,192°F for the CV are acceptable for their structural performance under HAC. The minimum allowable temperature for the Type 304L stainless steel is below the regulatory limit of -40°F, which is acceptable since 304L is not classified as a brittle material.

The CV in the 9516 package is of a welded construction, and the containment boundary does not rely on any closure seal that may be sensitive to temperature. The Helicoflex metal O-ring seal used for the cask closure has a design temperature limit of 450°F. This limit is acceptable since the cask acts as a confinement boundary, and the cask lid is held together by eight (8) bolts, even if the closure seal fails during HAC.

3.3 Thermal Evaluation under NCT

The SARP evaluated the thermal performance of the 9516 package during NCT by using the finite-element code SINDA/FLUENT. Details of the finite-element model and the analyses results are described in Section 3.3 and Appendices 3.5.2 and 3.5.3 of the SARP. All shipping configurations of contents in the 9516 package were analyzed by assuming the maximum decay heat load of 500 W. Other thermal loading also included insolation on the outer surfaces of the package per 10 CFR 71 71(c)(1). The primary heat transfer mechanisms considered are conduction and radiation within the package and convection and radiation from the exterior of the package to the ambient environment.

The DOE PCP staff used the ANSYS code (version 10.0) in the confirmatory evaluation of thermal performance of the same shipping configurations in the 9516 package, under the same initial and boundary conditions as those assumed in the SARP. All 500 W of the decay heat load was assumed uniformly deposited on the surface of contents.

3.3.1 Maximum Component Temperatures

Table 3.1 shows the calculated maximum temperatures for the various packaging components obtained for the powder cans under NCT. Because the same 500 W maximum decay heat load and the same insolation conditions are assumed for all shipping configurations regardless of contents, the temperature profiles of the packaging components obtained for powder cans should be representative of those for the other shipping configurations.

Table 3.1. Calculated Maximum Temperatures (°F) under NCT

Components	SARP	DOE PCP Staff*	Allowable
Powder can	977	912	1,940
Liner	703	649	800
CV	462	377	800
Cask	315	363	800
Liner gas**	840	780	N/A
CV gas**	583	513	N/A
Cask gas**	388	370	N/A

* Based on 100°F ambient temperature and 500 W decay heat load.

** Gas temperature is the averaged value of the highest temperatures on the inner and outer bounding surfaces.

There are some differences between the temperatures calculated in the SARP and the DOE PCP staff's confirmatory evaluation under NCT. With the exception of the cask temperature, the SARP temperatures for the product can, liner, and the CV are higher than the staff values; however, both are significantly below the allowable temperatures. Although the calculated maximum cask surface temperature of 315°F in Table 3.1 (and Table 3-1 of the SARP) exceeds the 185°F limit for the accessible surface of the package in an exclusive-use shipment per 10 CFR 71.43(g), the cask surfaces are not accessible during NCT because they are enclosed by the mesh screen of the personnel shield. Full-scale physical testing with a simulated internal heat source of 1,000 W (Section 3.3.3 of the SARP) and analysis (Appendix 3.5.4 of the SARP) showed that the mesh screen surfaces remain at or below 128°F, which is well below 185°F.

3.3.2 Maximum Normal Operating Pressure (MNOP)

On the basis of the maximum cask temperature of 390°F during NCT (Table 3-12 of the SARP), the MNOP in the cask is 23.6 psia, which is lower than the design pressure (314.7 psia) of the cask. The MNOP in the CV is 35.9 psia at the beginning of transport and 52.3 psia after 1 year, on the basis of the maximum cask temperature of 388°F (Table 3-12 of the SARP). These pressures are all lower than 200 psia, which is the ASME design pressure for the CV.

3.3.3 Maximum Thermal Stresses

The calculated maximum temperature for the cask body under NCT is 315°F, with the temperature drops of 5.4, 1.8, and 7.2°F, respectively, across the top (1.5 inches), the bottom plate (1.5 inches), and the side wall (1.5 inches), which translates into a maximum temperature gradient of 4.8°F/inch for the cask body. The 304 stainless-steel CV also has a very small temperature gradient, because of its relatively thin wall construction and high thermal conductivity. The small temperature gradients (ΔT) in the cask and CV are unlikely to result in any significant thermal stresses [$\approx(\alpha\Delta T)$ E] for the cask (see Section 2.6, Chapter 2 in the SARP), where α and E are the corresponding thermal expansion coefficient and Young's modulus, respectively.

3.4 Thermal Evaluation under HAC

The SARP used SINDA/FLUENT in the analyses of thermal performance of the 9516 package under HAC. The personnel shield of the package was assumed lost, as it was assumed in the DOE PCP staff's confirmatory evaluation of the package using the ANSYS code.,

The SARP and the DOE PCP staff's evaluations assumed that the initial conditions for the 9516 package before the HAC fire were obtained with the maximum 500 W decay hear load, 100°F ambient temperature, and no insolation. The surface absorptivities of all external surfaces of the cask were assumed to be 0.8 per 10 CFR 71.73(c)(4). The convective heat transfer coefficients were computed on the basis of forced convection correlations, with gas velocities of 32 foot/s during the 30-min fire at 1,472°F. Heat transfer to the bottom of the cask was treated as an adiabatic surface. Post-fire temperatures of the packaging components were also calculated.

3.4.1 Maximum Component Temperatures

Table 3.2 shows the calculated maximum temperatures for the various packaging components obtained for the powder cans that are also representative of other contents shipping configurations under HAC.

Both the SARP and the DOE PCP staff's calculated maximum component temperatures are below the allowable temperatures for the powder can, the liner, the CV, and the cask. The powder can and the liner do not serve any containment or confinement functions for the 9516 package. For the CV and the cask, Section 2.7.4 of the SARP shows that the peak temperatures of 1,192°F for the CV (containment) and 1,393°F for the cask (confinement) are acceptable for their structural performance under HAC.

Table 3.2. Calculated Maximum Temperatures (°F) under HAC

Components	SARP	DOE PCP Staff	Allowable
Powder can	1,250	1,156	2,500
Liner	1,045	975	2,500
CV	1,186	1,056	1,192
Cask	1,392	1,377	1,393
Liner gas*	1,489	1,065	n/a
CV gas*	1,098	1,015	n/a
Cask gas*	1,282	1,216	n/a

*Gas temperature is the average of the highest temperatures on the inner and outer boundary surfaces. It is conservative since two peak temperatures may not occur at the same time.

3.4.2 Maximum Internal Pressure

The maximum internal pressure in the cask during HAC is 48.4 psia, on the basis of the maximum cask gas temperature of 1,282°F (Table 3-16 of the SARP). This pressure is much lower than 300 psia, which is the cask design pressure limit. The maximum internal pressure in the CV during HAC is 177.5 psia, on the basis of the maximum CV gas temperature of 1,363°F (Table 3-16 of the SARP). This pressure is lower than the CV design pressure limit of 200 psia. The calculations are conservative because the CV was assumed to have reached its maximum allowable pressure of 118.5 psia under NCT, a value that is much higher than the MNOP of 52.3 psia.

3.5 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the thermal design and performance of the 9516 package presented in Chapter 3 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

4. CONTAINMENT

4.1 Discussion

The 9516 package is designed for transportation of up to 500 W of plutonium dioxide heat source material in any solid form. The 9516 package consists of a personnel shield that completely encloses a stainless steel cask. The contents are contained in liners and placed in a welded, Type 304L stainless-steel containment vessel (CV), which is placed in the cask. The CV provides the containment boundary for the package and is designed according to the 2004 ASME BPVC, Section III, Division 1, Subsection NB. The welded CV is to be used for only one shipment and is destroyed after opening. There are no penetrations (closures, valves, or pressure relief devices) through the CV. The radioactive content may need to be outgassed to prevent excessive pressure buildup in the CV during its typical lifecycle (e.g., one year). The SARP states that the need of fuel outgassing is determined by the total helium release from the contents within the CV.

The bottom and top lid of the CV are joined to the cylindrical body by full-penetration gas tungsten arc butt welds, as specified in the ASME BPVC, Section III, Division 1, Subsection NB, NB-4243, welded joint Category C. A definition of Category C welds is provided in the ASME BPVC NB-3351.3, which covers welded joints connecting flat heads to the main shell, such as the bottom and top lid of the CV.

After the contents are loaded and the CV is welded close, the CV is leakage rate tested with a mass spectrograph to the ANSI N14.5-1997, *American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment,* A5.4 (helium) for radioactive materials leakage tests packages for shipment.

4.2 Containment under NCT

The results of the analyses described in Chapters 2 and 3 of the SARP show that the containment boundary will not fail under the NCT. The CV does not experience significant stresses from the mechanical and thermal loads under NCT. Testing of the package under NCT showed minor damage. The CV was leakage rate tested after the NCT tests, as described in Section 4.2.3 of the SARP, and was shown to meet the ANSI N14.5 leak-tight criterion. The 10 CFR 71.51 regulatory limit of 10^{-6} A_2/h for the release of radioactive material during NCT is met by demonstrating that the leakage rate from the CV is $\leq 10^{-7}$ std cm^3/s, which is the ANSI N14.5 leak-tight criterion.

The SARP determined that the maximum allowable pressure within the CV under NCT is 118.5 psia. The SARP indicates that this pressure can be attained in the CV after it is sealed from 61 to 405 months (Table 3-13 of the SARP), depending on the radioactive contents. The restriction on the wattage of the heat source contents ($\leq$500 W) provides assurance that the maximum allowable pressure is not exceeded.

4.3 Containment under HAC

The SARP and the DOE PCP staff's confirmatory structural evaluation (see Section 2 of this SER) showed that even for the worst-case, 30-foot bottom-end free drop, crush, and puncture tests in HAC, the CV maintains its containment boundary. The 10 CFR 71.51 regulatory limit of A_2/week for the release of radioactive material after HAC is met by demonstrating that the leakage rate from the CV is $\leq 10^{-7}$ std cm^3/s, which is the ANSI N14.5 leak-tight criterion.

The SARP determined that the maximum allowable pressure within the CV under HAC is 162.8 psig (177.5 psia). The restriction on the wattage of the heat source contents ($\leq$500 W) provides assurance that the maximum allowable pressure is not exceeded.

4.4 10 CFR 71.61 Requirement

Table 4-1 of the SARP shows that the contents payload may contain up to 5.59×10^5 A_2. 10 CFR 71.61 stipulates that "*A Type B package containing more than 10^5 A_2 must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or inleakage of water.*" Section 2.7.7 of SARP showed that this requirement is satisfied by the CV of the 9516 Package.

4.5 Leakage Rate Tests for Type B Packages

ANSI N14.5-1997 requires fabrication, maintenance, and periodic and pre-shipment leakage rate tests for Type B packages. The SARP provides the basis for a single leakage rate test of the CV after weld closure. The welded CV is not opened before shipment; therefore, no pre-shipment leakage rate testing is needed. The CV is not reused; therefore, no maintenance and periodic leakage rate testing is needed.

The acceptance criterion for the fabrication and pre-shipment leakage rate testing is leak-tight, as defined by ANSI N14.5-1997, or demonstration that the leakage rate from the package is $\leq 1 \times 10^{-7}$ ref-cm^3/s of air at an upstream pressure of 1 atmosphere absolute and a downstream pressure of 0.01 atmosphere absolute. Helium mass spectrometry using the evacuated envelope method per ANSI N14.5-1997 is used to verify a leakage rate $\leq 1 \times 10^{-7}$ ref-cm^3/s.

4.6 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the containment design and performance of the 9516 package presented in Chapter 4 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

5. SHIELDING

5.1 Discussion

Chapter 5 of the SARP describes the shielding design, radiation sources, models, and calculation results of the 9516 package. The source terms and shielding analyses presented in Chapter 5 of the SARP are bounding cases for the PuO_2 heat sources up to 500 W. The dose rates calculated for the two types of payloads in the GPHS modules and the eight powder cans bound those for all shipping configurations described in Table 1-2 of the SARP. The highest dose rates were calculated for the eight-powder-can cases in NCT and HAC. The 9516 package is shipped exclusive use, and up to six (6) packages are allowed in a conveyance.

5.1 Description of Shielding Design

The 9516 packaging does not contain material specifically for shielding, although the stainless-steel cask body, the CV, and the graphite packing materials provide some radiation attenuation. Restricting the amount of source material, maintaining the distance between the source and the external surface of the package, and maintaining proper spacing for multiple packages on the transport vehicle are the means by which the radiation dose rates are kept below the regulatory limits.

5.2 Source Specification

For the contents of the 9516 package, the source of photons is due to the combination of decays of the plutonium isotopes and fuel impurities and photons due to fission and the decay of fission products. Direct decay of the plutonium isotopes is the dominating source terms for photons, and ^{236}Pu and ^{238}Pu are the main contributors. The decay of ^{236}Pu leads to ^{208}Tl, which produces high-energy gammas, and the ^{236}Pu concentration is limited to ≤ 2.0 ppm (see Tables 5-3 and 1-1 in the SARP). Tables 5-11 and 5-12 of the SARP show the ORIGEN-S calculated photons per second for the eight powder cans versus percent enrichment (Table 5-11) and after 17.5-yr decay and 10-day decay (Table 5-12), both for 74 wt. % ^{238}Pu and 2.0 ppm ^{236}Pu. A key observation in Table 5-12 is that the photon source strengths in the higher energy range (0.25–2.75 MeV), which contribute more heavily to the personnel dose, are considerably higher for the 17.5-yr decay than for the 10-day decay for 74 wt. % ^{238}Pu and 2.0 ppm ^{236}Pu.

For the contents of the 9516 package, the source of neutrons is due to the combination of (1) alpha-neutron (α, n) reactions, of which the major contributor is ^{238}Pu (99%); (2) spontaneous fissions, of which the major contributor is ^{238}Pu (99%); and (3) neutron-induced fissions.

The number of neutrons from (α, n) reactions and spontaneous fissions is determined by the total mass of ^{238}Pu. The number of neutrons from neutron-induced fissions is determined by the total mass of ^{239}Pu. An enrichment of 74 wt.% ^{238}Pu yields the largest neutron source due to the higher induced fission neutron source from the larger quantity of ^{239}Pu (23.63%). Since the neutron dose rate is proportional to the source, the 74 wt.% ^{238}Pu mixture yields the largest dose rate for the 9516 package.

The photon and neutron source spectra are calculated in the SARP by using ORIGEN-S, and the results are shown in Table 5-12 and Table 5-14, respectively, for the eight-powder-cans case with 74 wt.% ^{238}Pu and 2 ppm ^{236}Pu. The DOE PCP staff used ORIGEN-ARP 5.1.01 in the confirmatory evaluation. The neutrons from neutron-induced fissions are not included in the ORIGEN-S source terms, but they are included in the neutron transport calculations.

The total neutron source strength based on the specific neutron emission rate (SER) of 18,000 n/s-g-^{238}Pu corresponds to 1.583×10^7 n/s for the eight (8) powder cans at 74 wt. % ^{238}Pu (=18,000 n/s-g-$^{238}P \times 879.5$ g-^{238}Pu). This neutron source term exceeds those calculated by the ORIGEN-S for the 17.5-yr decay and is used as the limiting neutron source term in the MCNP calculations of dose rates.

5.3 Shielding Model

There are six shipping configurations listed in the SARP. The two GPHS modules configurations and the eight (8) domestic powder cans configurations are considered to be the bounding cases for the shielding calculations. For NCT, the surface of the personnel shield is considered to be the package surface where dose rates are calculated. For HAC and on the basis of structural evaluation, the SARP conservatively assumed total loss of personnel shield and that the containment vessel remains inside the cask. Thus, the surface of the cask is considered as the package surface in HAC dose rate calculations. The DOE PCP staff has confirmed the dimension of the shielding model and verified the assumption used in the SARP for the dose rate calculations.

For the shipment of multiple packages, the SARP calculated the dose rates for single packages at appropriate distances and added them together. The DOE PCP staff used a more realistic model that included the multiple packages and the vehicle as a whole. The ground scattering effect was included in the DOE PCP staff's model, whereas the SARP included a 6-inches-thick concrete slab placed at 48 inches below the trailer to account for radiation scattering off the ground.

5.4 Shielding Evaluation

The MCNP 5 was used for shielding evaluation in the SARP. The DOE PCP staff used MCNP 5, version 5.1.4, for the confirmatory evaluation. The cross section library used in the evaluations was based on ENDF VI. ANSI/ANS-6.1.1-1977 *Neutron and Gamma-Ray Flux-to-Dose-Rate Factors* was used to calculate personnel doses. The comparison of the calculated dose rates is shown in Table 5.1 and Table 5.2 for single and multiple 9516 packages, respectively.

Table 5.1 Maximum Dose Rates of a Single 9516 Package

	Dose Location	SARP (mrem/h)	DOE PCP Staff (mrem/h)	10 CFR 71 Limits* (mrem/h)
NCT	Top surface of the personnel shield	58.3	59.9	1,000
	Side surface of the personnel shield	149.2	155.9	1,000
	Bottom surface of the personnel shield	252.9	239.3	1,000
HAC	1 m from the top surface of the cask	9.0	6.5	1,000
	1 m from the side surface of the cask	19.4	19.3	1,000
	1 m from the bottom surface of the cask	13.5	10.3	1,000

* For exclusive use shipment

Table 5.2 Maximum Dose Rates of a Shipment of Six (6) 9516 Packages

Dose Location	SARP (mrem/h)	DOE PCP Staff (mrem/h)	10 CFR 71 Limits (mrem/h)
Bottom surface of the vehicle	189 6	197.2	200
2 m from the side surface of the vehicle	8.3	9.5	10
Normally occupied position in vehicle	3.6	3.9	2*

*Private carriers may exceed this limit, provided that the exposed personnel under their control wear radiation dosimeters in conformance with 10 CFR 20.

For a single 9516 package, the dose rates calculated for NCT and HAC in the SARP and by the DOE PCP staff are all significantly below the regulatory limits for the exclusive use shipment. For six 9516 packages, the dose rates calculated at the bottom surface of the vehicle and at 2 m from the side surface of the vehicle are reasonably close between the SARP and the DOE PCP staff confirmatory evaluation, and both are below the regulatory limits in 10 CFR 71.47. The higher dose rates calculated by the DOE PCP staff are the results of the ground scattering effect included in the staff's MCNP 5 model.

The calculated dose rate in the normally occupied space (i.e., the truck cab) exceeds the 2-mrem/h (0.02-mSv/h) limit for a non-private carrier. This limit does not apply to private carriers, if exposed personnel under their control wear a radiation dosimeter in conformance with 10 CFR 20.1502.

For the ISO cargo container with three (3) 9516 packages placed in accordance with the loading restrictions in Step 11, Section 7.1.1 of the Rev. 1 SARP, the DOE PCP staff has confirmed that the calculated maximum dose rates are bounded by those obtained for the six (6) 9516 packages shown in Table 5.2 of this SER.

5.5 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the shielding design and performance presented in Chapter 5 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, DOE Order 460.1B have been met.

6. CRITICALITY

6.1 Discussion

The 9516 package is designed to transport plutonium heat sources, including assembled general purpose heat source (GPHS) modules, GPHS fueled clad assemblies, and the fuel for the heat sources. The fuel for the heat sources consists of plutonium dioxide containing 74–90 wt.% ^{238}Pu in the plutonium; the balance of the plutonium is predominantly ^{239}Pu and ^{241}Pu. Weight percent and atom percent are essentially equivalent for the plutonium in the heat sources for criticality safety considerations. The differences in the calculated effective neutron multiplication factor (k_{eff}) and the subcritical margin for 74 –90 weight percent (wt.%) ^{238}Pu and 74–90 atomic percent (at.%) ^{238}Pu are negligible. The criticality safety analysis in Chapter 6 of the SARP focused on the GPHS modules, each containing four (4) PuO_2 fuel pellets, and the PuO_2 powder as the bounding contents for the 9516 package.

A GPHS module is placed in a graphite support block for protection during transport, and the graphite support block is placed in a stainless-steel liner. Loaded liners are placed in a stainless-steel CV, which provides the containment boundary for the package. Two liners containing two (2) GPHS modules are loaded into the CV; graphite filler blocks are placed in the CV to eliminate free space and restrict axial motion of the liner(s). The CV is inserted into the stainless-steel cask to complete the 9516 package. The SARP used three (3) GPHS modules containing a total of twelve (12) PuO_2 pellets in the criticality safety evaluation, which is conservative since only two (2) GPHS modules are allowed in the exclusive use shipment.

The PuO_2 powder is contained in a stainless-steel powder can, which is placed in a stainless-steel product can. Loaded product cans are placed in a graphite support block, which is, in turn, placed in a stainless-steel liner. Loaded liners are placed in the CV. Graphite filler blocks are used to separate individual liners and to occupy any free space in the CV. The CV is inserted into the stainless-steel cask to complete the 9516 package. A maximum of four product cans fit into a graphite support block, and a maximum of two liners containing product cans fit into the CV. For criticality safety evaluation, the SARP analyzed the shipping configuration of four product cans in each graphite support block and two liners containing product cans in the CV.

6.2 Evaluation of Criticality Models and Assumptions

Three-dimensional KENO V.a models were constructed for the 9516 packages containing GPHS modules and PuO_2 powder in product cans, as described in the SARP. Figures 6-1 through 6-4 of the SARP show the KENO V.a model for a 9516 package with three GPHS modules in graphite support blocks. Figures 6-5 through 6-8 of the SARP show the KENO V.a model for a 9516 package that has two liners containing graphite support blocks with four product cans per graphite support block. Each product can contains a powder can filled with PuO_2 powder. The SARP determined the most reactive NCT and HAC configurations for each payload category.

6.2.1 GPHS Modules

The SARP model for the GPHS modules is based on (1) the maximum allowable fuel pellet diameter and height; (2) neglect of the rounded corners of the fuel pellet; (3) stoichiometric PuO_2 containing 33 at.% plutonium; and (4) a fuel density of 10.89 g/cm^3, which is 95% of the theoretical density of PuO_2. The actual densities of the fuel pellets range from 84% to 86% of the theoretical density of PuO_2. All of these factors maximize the plutonium content of the fuel pellets and result in a model that exceeds the maximum plutonium content of the GPHS modules allowed in the 9516 package.

The SARP geometric model neglects the personnel shield and reduces the length and width of the base plate to the outer diameter of the cask. These features of the model maximize the interaction of adjacent casks in the array configurations. The cask was assumed to be surrounded by 30 cm of water for the single-unit calculations. These conservative assumptions resulted in identical single-unit configurations for the 9516 package under NCT and HAC.

For the single-unit NCT calculations of an undamaged package containing three GPHS modules, the effects of the allowable range of ^{238}Pu enrichments and the consequences of water leakage into the CV on k_{eff} are shown in Tables 6-8 and 6-9 of the SARP. No calculations were performed for arrays of undamaged packages under NCT because the NCT array is bounded by the calculations for an infinite array of damaged packages in the HAC configuration.

The HAC array calculations are based on an infinite array of damaged packages. For the HAC array cases, the personnel shield was neglected, and the length and width of the cask base plate were reduced to the diameter of the cask. This modeling approximation maximizes interaction between adjacent casks in the array. The HAC calculations for an infinite array of damaged 9516 packages, each containing three GPHS modules, are shown in Tables 6-12 (varying ^{238}Pu enrichment, no water inleakage) and 6-13 (varying inleakage water density and 74 at. % ^{238}Pu) of the SARP.

For the NCT calculations of an undamaged package containing three GPHS modules, the highest $k_{eff} + 2\sigma$ is 0.30496. For the HAC calculations of an infinite array of damaged packages, the highest $k_{eff} + 2\sigma$ is 0.53450. These maximum reactivities under NCT and HAC are all well below the USL of 0.700, with substantial safety margins determined in Section 6.8 of the SARP.

Table 6.1 summarizes the results from the SARP and the DOE PCP staff's confirmatory evaluation of criticality safety for transport of three (3) GPHS modules in a 9516 package. The calculated values of $k_{eff} + 2\sigma$ generally agree within 0.01 after rounding off the calculated value. Given the large margin to the USL of 0.700, this difference is not significant.

Table 6.1 Maximum Reactivity for the 9516 Package with 3 GPHS Modules

		NCT		HAC	
		Single Unit		Infinite Array	
Results		SARP	DOE PCP Staff	SARP	DOE PCP Staff
Maximum Reactivity	k_{eff}	0.30312	0.30304	0.53250	0.54286
	σ	0.00092	0.00063	0.00100	0.00070
	$k_{eff}+2\sigma$	0.30496	0.30430	0.53450	0.54426
Safe Limit	USL	0.700		0.700	

6.2.2 Plutonium Dioxide in Product Cans

The SARP model for product cans containing plutonium dioxide powder has two liners, each containing four product cans, in the CV. The SARP model for the product cans containing powder cans is based on (1) 262.5 g of PuO_2 powder per powder can; (2) one powder can in each product can; (3) four product cans per liner; (4) stoichiometric PuO_2 containing 33 atomic percent plutonium; and (5) a PuO_2 powder density of 5.73 g/cm^3, which is 50% of the theoretical density of PuO_2.

The SARP geometric model neglects the personnel shield and reduces the length and width of the base plate to the outer diameter of the cask. These features of the model maximize the interaction of adjacent casks in the array configurations. The cask was assumed to be surrounded by 30 cm of water for the single-unit calculations. These conservative assumptions resulted in identical single-unit configurations for the 9516 package under NCT and HAC.

For the single-unit NCT calculations of an undamaged package containing powder in eight (8) product cans, the effects of the allowable range of ^{238}Pu enrichments and the consequences of water leakage into the CV on k_{eff} are shown in Tables 6-10 and Table 6-11 of the SARP. No calculations were performed for arrays of undamaged packages under NCT because the NCT array is bounded by the calculations for an infinite array of damaged packages in the HAC configuration.

The HAC array calculations are based on an infinite array of damaged packages. For the HAC array cases, the personnel shield was neglected, and the length and width of the cask base plate were reduced to the diameter of the cask. This modeling approximation maximizes the interaction between adjacent casks in the array. The HAC calculations for an infinite array of damaged 9516 packages, each containing PuO_2 powder in 8 product cans, are shown in Tables 6-14 (varying ^{238}Pu enrichment, no water inleakage) and 6-15 (varying inleakage water density and 74 atomic percent ^{238}Pu) of the SARP.

For the NCT calculations of an undamaged package containing PuO_2 powder in eight (8) product cans, the highest $k_{eff} + 2\sigma$ is 0.24640. For the HAC calculations of an infinite array of damaged packages, the highest $k_{eff} + 2\sigma$ is 0.50160. These maximum reactivities under NCT and HAC are all well below the USL of 0.700 with substantial safety margins.

Table 6.2 summarizes the results from the SARP and the DOE PCP staff's confirmatory evaluation of criticality safety for transport of PuO_2 powder in eight (8) product cans in a 9516 package. The calculated values of $k_{eff} + 2\sigma$ agree within 0.02 after rounding off the calculated value. Given the large margin to the USL, this difference is not significant.

Table 6.2 Maximum Reactivity for the 9516 Package with PuO_2 Powder in Eight (8) Product Cans

Results		NCT Single Unit SARP	NCT Single Unit DOE PCP Staff	HAC Infinite Array SARP	HAC Infinite Array DOE PCP Staff
Maximum Reactivity	k_{eff}	0.24450	0.24540	0.49940	0.51776
	σ	0.00095	0.00060	0.00110	0.00066
	$k_{eff}+2\sigma$	0.24640	0.24660	0.50160	0.51908
Safe Limit	USL	0.700		0.700	

The results in Tables 6.1 and 6.2 demonstrate that the 9516 package meets the fissile material criticality safety requirements in 10 CFR 71.55 and 71.59 under NCT and HAC.

The Criticality Safety Index (CSI), on the basis of the most reactive infinite array of 9516 packages, is zero (0) per 10 CFR 71.59(b), which also meets the requirement in 10 CFR 71.59(c) for the exclusive use shipment.

6.3 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the criticality safety design and performance of the 9516 package presented in Chapter 6 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 73, and DOE Order 460.1B are met.

7. OPERATING PROCEDURES

7.1 Discussion

Chapter 7 of the SARP describes the procedural requirements for loading, shipping, and receiving the 9516 package and summarizes the requirements imposed on all package users. These procedural requirements must be implemented by site-specific procedures to ensure that the package is used in accordance with the CoC and the SARP for the 9516 package. As custodian of the 9516 package, INL provides all other package users with generic procedures and written instructions for all operations that need to be performed. The package must be loaded and handled in accordance with "As Low as is Reasonably Achievable" (ALARA) principles contained in 10 CFR Part 20. As discussed in Chapter 9 of the SARP, all package operations must be performed by qualified personnel using calibrated and controlled gages, instruments, measuring devices, and testing equipment. Oversight organizations, such as Quality Assurance or Quality Control, must monitor activities of users by performing audits, surveillances, and inspections.

7.2 Package Loading

Section 7.1 of the SARP discusses package loading requirements for the 9516 package. Before each shipment, the user must have site-specific procedures that comply with the requirements of Chapter 7 of the SARP and the "Routine Determinations" discussed in 10 CFR 71.87. Before any packaging operations are begun, the payloads to be shipped must be fully characterized with respect to the chemical and physical forms, and the specific radiation, heat output, and age must be determined to ensure that radiation, decay heat, and age limits are not exceeded.

All radioactive material shipped in the 9516 package must be placed in a liner that is then placed in a CV. Up to two liners can be placed in the CV, along with any required graphite filler blocks to restrict movement during transport. Before the CV is loaded into the cask, the CV must be visually inspected, leak tested, and smear tested, and the CV closure weld must be radiographed. A calorimetric measurement must be taken to make sure the decay heat load does not exceed the 500-W limit.

Section 7.1.1 of the SARP discusses preparations for loading of the package. This section of the SARP contains eighteen (18) elements that must be completed before loading of the package. These elements for the preparation of loading include (1) having written procedures that comply with the requirements of Chapter 7 and 10 CFR 71.87; (2) complying with ALARA principles, as stipulated in 10 CFR 20.1101(b); (3) ensuring the 500-W limit is not exceeded; (4) visually inspecting the outer surface of the cask and personnel shield; and (5) having all components surveyed by health physics.

Section 7.1.2 of the SARP discusses loading of the contents into the package for shipment, which is a three-step process: (1) loading the radioactive material into the liner, (2) loading the liner into the CV, and (3) loading the CV into the cask. This section of the SARP sets forth twenty-six (26) elements that must be completed during loading of the contents into the package. These elements for loading include (1) conducting loading of the contents in accordance with written procedures; (2) verifying that the heat generation and radioactive material content are within required limits; (3) loading and welding the liner and CV in an inert gas chamber; (4) visually examining, radiographing, and testing the top CV closure weld; and (5) loading the CV into the cask and the cask into the personnel shield.

Section 7.1.3 of the SARP discusses preparation for transport. This section of the SARP sets forth thirteen (13) elements that must be completed in preparing the package for transport. These preparations for transport elements include (1) performing and documenting the radiological survey of the package; (2) verifying that the surface temperature of the package is less than 180°F at the external, accessible surfaces of the personnel shield; (3) verifying that the gross package weight is 900 lb or less; (4) attaching radiation markings and labels, as required by 49 CFR 172; and (5) verifying that the maximum age of the plutonium dioxide content at the end of the shipment will not exceed the values listed in Table 3-13 of the SARP.

7.3 Package Unloading

Section 7.2 of the SARP discusses package unloading, which includes procedural elements, facility requirements and ALARA principles.

Section 7.2.1 of the SARP discusses the receipt of the package from the carrier. This section of the SARP contains sixteen (16) elements that must be completed when receiving the package from the carrier. Steps 1 through 3 address radiation control and survey elements. Steps 4 through 9 address receiving inspection elements and practices. Steps 10 through 16 address removal of the CV from the Personnel Shield.

Section 7.2.2 of the SARP discusses removal of the contents from the package. Essentially, the CV is transferred to a controlled facility for opening, where the CV is cut open at the cutting groove. The liners are then removed from the CV and surveyed for contamination. The liners are then transferred to a controlled facility and opened in the same manner as the CV to remove the radioactive material contents.

7.4 Preparation of an Empty Package for Transport

Section 7.3 of the SARP discusses preparation of an empty package for transport. This section of the SARP sets forth seven (7) elements that must be completed when preparing an empty package for transport. The accessible surfaces of all items to be shipped are surveyed to make sure they do not exceed contamination limits specified in 49 CFR 173.443. Any internal components to be shipped that are not contaminated are placed in the cask cavity, the metal O-ring from the last shipment is installed, and the cask lid is installed. An empty non-contaminated package is shipped in accordance with 49 CFR 173.428. If the package is internally contaminated, it is shipped in accordance with 49 CFR 173.421.

7.5 Other Operations

Section 7.4 of the SARP discusses other operations. This section of the SARP specifically addresses the age of the authorized contents of the shipments. The age of the authorized contents at the end of a shipment must not exceed the values specified in Table 3-13 of the SARP. The start time of the restriction begins when the PuO_2 powder or FCAs are processed, and the shipment must be completed before the time duration in Table 3-13 is exceeded for the specific shipping configuration being transported.

7.6 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the operating procedure requirements presented in Chapter 7 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

8. ACCEPTANCE TESTS AND MAINTENANCE

8.1 Discussion

Chapter 8 of the SARP describes the acceptance tests and maintenance program for the 9516 packaging. The personnel shield and cask are reusable, while the CV is a one-time-use component. The 9516 packaging components may be purchased from qualified fabricators or vendors or fabricated by DOE facilities (or any combination thereof). The fabricator must provide test reports and certifications for the materials, processes, and the qualification of personnel performing quality activities. The requirements for quality activities are contained in the design drawings, specifications, quality program, and procurement documents. All documentation must be reviewed by qualified personnel before acceptance of the packaging. Nonconforming items are handled in accordance with the Quality Assurance (QA) program described in Chapter 9 of the SARP.

8.2 Acceptance Tests

Section 8.1 of the SARP describes the packaging acceptance tests and inspections for the personnel shield, cask, CV, and internal components. These inspections and tests are performed in accordance with 10 CFR 71.85, "Preliminary determinations," the design drawings in Chapter 1 of the SARP, and other quality documents.

Section 8.1.1 of the SARP addresses visual inspections and measurements. The packaging components are manufactured from standard materials by using standard industrial practices. The fabricator performs visual observations of function, fit, and finish during the manufacturing process. Qualified personnel perform visual inspections and measurements to verify the quality of the manufacturing processes and the finished packaging components. Visual inspections for the personnel shield, cask, internal components, and CV are covered in Sections 8.1.1.1 through 8.1.1.4 and in Appendix 8.3.2 of the SARP.

Section 8.1.2 of the SARP addresses weld examinations. The weld examinations for the personnel shield, cask, and liner are addressed in Section 8.1.2.1 and on the design drawings in Chapter 1 of the SARP. Nonconforming welds are handled in accordance with the QA program described in Chapter 9 of the SARP.

The weld examinations for the CV bottom lid are addressed in Section 8.1.2.2 and on the design drawings in Chapter 1 of the SARP. These examinations are done using visual, radiography and liquid penetrant testing, in accordance with the ASME BPVC.

The weld examinations for the CV top lid are addressed in Section 8.1.2.3 and on the design drawings in Chapter 1 of the SARP. The CV top lid closure weld is not examined in strict compliance with the ASME BPVC. The CV top lid is examined visually and by radiography in accordance with the ASME BPVC. The CV top lid closure weld cannot be examined by liquid penetrant test because of the high operating temperature ($\approx 400^{\circ}$F). In lieu of the liquid penetrant test, the top lid closure weld is subjected to a mass spectrometer helium leakage rate test in accordance with Section 8.1.4 of the SARP. The DOE PCP finds the substitution acceptable.

Section 8.1.2.3.1 of the SARP addresses changes to the CV top lid welding apparatus. If this welding apparatus is changed to a different configuration, the new configuration must be re-qualified, and the operators must be re-qualified in the same manner as the original qualification. The overpressure tests must also be redone.

Section 8.1.2.3.2 of the SARP addresses CV top lid qualification welds. The top lid closure welds do not strictly comply with the ASME BPVC, because the top lid closure welds are made remotely inside a contamination-controlled zone in an inert atmosphere. Specimen welds are made to qualify equipment parameters and configuration and are identical to the CV design. Each welding operator must produce two specimens before making production welds on the CV. The specimen welds are visually inspected, helium leakage rate tested, radiographically examined, metallographic examined, bend tested, and proof tested in accordance with Section 8.1.3.2 of the SARP. All examinations must be conducted with personnel qualified in accordance with the ASME B&PVC. The DOE PCP finds the qualification requirements described in the SARP acceptable.

Section 8.1.2.3.3 of the SARP addresses repair of defective CV top lid welds. To vent the CV, a hole is drilled near the area where the weld repair is to be performed, and then the CV is placed in a contamination control enclosure, evacuated, and backfilled with an inert gas at least three times. After the final evacuation, the weld is repaired by performing a fusion weld pass over the defective area or by performing a manual gas tungsten arc weld in the defective area. After the repair, the visual and radiographic examinations, and the leakage rate testing are repeated.

Section 8.1.2.4 of the SARP addresses use of weld flux. The bottom and top lid welds may be performed with or without the use of weld flux. This is at the discretion of the welding engineer. The composition of the flux is described in the SARP.

Section 8.1.3 of the SARP addresses structural and pressure tests. Section 8.1.3.1 of the SARP discusses these tests for the cask. An external hydrostatic pressure test is performed on each cask before acceptance. This test is conducted at the HAC 50-foot (21.7 psi) water immersion test per 10 CFR 71.73(c)(6) for eight hours. The acceptance standard is that there must be no evidence of water in-leakage. The cask is designed as a pressure vessel and has a MNOP of 8.9 psig and a maximum HAC of 33.7 psig per Table 3-2 of the SARP. The cask was originally designed to 300 psi and is estimated to be able to withstand an internal pressure of ≈408 psi and an external pressure of ≈1,543 psi, according to Chapter 2 of the SARP. The cask lid gasket has a working pressure of 300 psi. These high-pressure capabilities result from the robustness of the cask's design for other HAC regulatory tests, such as the free drop, crush, puncture, and thermal tests. The DOE PCP finds the external hydrostatic pressure test, conducted at ≈21.7 psig, an acceptable alternative to an ASME hydrostatic test.,

Section 8.1.3.2 of the SARP discusses the structural and pressure tests for the CV. 10 CFR 71.85(b) requires an initial pressure test of at least 50% higher than the MNOP. All CV bottom-end to shell welds are hydrostatically tested to 1.5 times MNOP. Because the final assembly of the CV requires that it must be welded shut, sample CVs are prepared for pressure test purposes. These sample CVs are provided with a test pressure connection to demonstrate the adequacy of the design and fabrication process. Two CVs are prepared by each certified welder as samples and tested. Testing must be completed before the first production run. ASME BPVC NB-6220 requires a hydrostatic test pressure of 1.25 times the design pressure (1.25×200 psig = 250 psig). The hydrostatic testing is done at 250 psig at room temperature. 10 CFR 71.85 requires a hydrostatic test pressure of 1.5 times MNOP. This hydrostatic test pressure was established from the 37.6 psig in Chapter 3 (1.5×37.6 psig = 56.4 psig) and 65 psia or 50.3 psig for calculation purposes in Chapter 2 (1.5×50.3 psig = 74.45 psig). Therefore, the NB-6220 hydrostatic test pressure of 250 psig also satisfies the hydrostatic test pressure requirement in 10 CFR 71.85. After the initial production run, one CV must be tested from every 20th CV produced by each welder.

Section 8.1.4 of the SARP addresses leakage tests. Two helium mass spectrometer leakage tests are performed on each CV. The first is done to demonstrate that the leakage of the bottom-end assembly is $\leq 1 \times 10^{-7}$ ref-cm^3/s of air according to ANSI N14.5-1997. The second is done after final assembly, after closure welding, and before shipment to demonstrate that the leakage of the final closure is $\leq 1 \times 10^{-7}$ ref-cm^3/s of air, according to ANSI N14.5-1997.

Section 8.1.5 of the SARP addresses component and material tests. The vendors supplying material for the personnel shield (Section 8.1.5.1) will be qualified vendors and provide material certifications stating that the material meets the purchase order requirements. The cask materials (Section 8.1.5.2) will meet the material certification requirements in the ASME BPVC, Section VIII, Division 1, and the design drawings in Chapter 1 of the SARP. The materials used in the CV (Section 8.1.5.3) will be supplied by qualified vendors and have certifications stating that the material meets all specifications stated on the design drawings in Chapter 1 of the SARP. All material and component examinations will be performed in accordance with the requirements of the ASME BPVC, Section III, Division 1. The vendor supplying the gasket (Section 8.1.5.4) will be a qualified vendor and will provide certifications stating that the cask seals meet the purchase order requirements.

Section 8.1.6 discusses shielding tests. Meeting the required radiation levels outside the package is achieved by a combination of the shielding provided by the packaging components and the distance from the radioactive contents provided by the personnel shield. No special shielding tests are required, other than the pre-shipment radiation surveys discussed in Chapter 7 of the SARP.

Per Section 8.1.7 of the SARP, thermal acceptance tests are not required. The personnel shield provides the necessary standoff distance to ensure that maximum accessible temperatures are not exceeded. No miscellaneous tests are required, as stated in Section 8.1.8 of the SARP.

8.3 Maintenance Program

Section 8.2 of the SARP addresses the maintenance program. The 9516 packaging is visually examined during loading and unloading operations, and any defective packaging is tagged for further examination prior to use.

Structural and pressure tests are conducted in accordance with Section 8.2.1 of the SARP. Before reuse, each cask and personnel shield are examined for surface damage and defective parts. The personnel shield frame cannot have a deflection of greater than 0.25 inch over the full length of the member. Deflections of greater than 0.25 inch are tagged and repaired. The cask O-ring sealing surfaces are examined during each reuse inspection. Any damage is reworked, and the cask is hydrostatically retested to the 50-foot immersion criterion before the cask is placed back into operation.

A new CV is used for each shipment. The new CVs are tested as discussed in Section 8.1.4 of the SARP; therefore, no maintenance leakage tests are required per Section 8.2.2 of the SARP.

Section 8.2.3 of the SARP discusses component and material tests. The seal for the cask closure is new and is visually inspected for surface defects, such as dents and scratches, before installation. Seals found to be defective during this inspection are rejected and replaced. Seals are not reused for shipment of radioactive materials. There are no valves, rupture discs, or gaskets on the CV that require maintenance.

Section 8.2.4 addresses thermal tests. The personnel shield provides protection from the hot surfaces of the cask. No special thermal tests are required for the personnel shield other than the inspections discussed in Section 8.2.1 of the SARP.

Miscellaneous tests are discussed in Section 8.2.5 of the SARP. All miscellaneous items (such as bolts, nuts, screws, and washers) are examined for damage and replaced if necessary. If any damage is found in the welded joints of the personnel shield, the damaged portions are ground out and re-welded. All repaired and replaced items are inspected by qualified personnel.

8.4 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the acceptance tests and maintenance program requirements presented in Chapter 8 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B have been met.

9. QUALITY ASSURANCE

9.1 Discussion

The requirements for a QA Program presented in Chapter 9 of the SARP have been reviewed and found to satisfy the QA requirements of 10 CFR Part 71, Subpart H. These QA requirements provide sufficient control over all items and quality-affecting activities that are important to safety as applied to the design, fabrication, assembly, inspection, testing, operation, maintenance, modification, and repair of the 9516 packaging. The 9516 package has a maximum gross weight of 900 lb and consists of a cylindrical cask that is housed in a 30.75-inches-square by 35.25-inches-mesh personnel shield. The containment boundary is a welded vessel that is housed within the cask during transport. The 9516 package is designed to ship plutonium heat source material in accordance with the requirements of 10 CFR 71 and 49 CFR 173. The QA requirements are based on a graded approach, as described in 10 CFR 71.101.

9.2 Evaluation of Applicant's QA Program

9.2.1 QA Program

The INL *Quality Assurance Program Description* (PDD-13000) provides implementing procedures that demonstrate compliance with each of the 18 QA requirements in 10 CFR 71, Subpart H. The INL Quality Assurance Program Description (QAPD) provides a cross-referencing matrix that links each of the 18 QA requirements in 10 CFR 71, Subpart H, to the QAPD.

9.2.2 Graded Approach

The graded approach in the QA Chapter of the SARP includes an important-to-safety Q-list for each significant item and activity and is graded on the basis of the design function of the item relative to the safety and performance requirements for the complete packaging. The quality assurance categories for each component are listed in Table 9-1 of the SARP with the relationship between NRC Regulatory Guide 7.10 quality categories and the INL quality categories of the SARP. The Q-list uses three QA categories with associated definitions for each. The QA level of each important-to-safety item is based on specific criteria. The QA requirements ensure that the packaging components are designed, fabricated, tested, and operated in accordance with the drawings identified in the SARP. In addition, the QA Chapter requires the user to invoke the same level of QA requirements for the use, maintenance, and repair of the packaging components, as is required for the procurement, fabrication, and acceptance testing of the original packaging components. The QA categories for important-to-safety items and activities and non-safety related items are based on the following definitions in Section 9.2.1 of the SARP:

1. Category A (QL-1) – Items that are critical to safety operation. Category A items could be structures, components, and systems whose failure or malfunction could result directly in a condition adversely affecting the public health and safety. This would include such conditions as loss of primary containment with subsequent release of radioactive material, loss of shielding, or an unsafe geometry compromising criticality control.
2. Category B (QL-2) – Items that have a major impact on safety. Category B items could be structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. An unsafe condition involving category B type items could only occur if a primary failure occurs in conjunction with another failure.
3. Category C (QL-3) – Items that have a minor impact on safety. Category C items could be components and systems whose failure would not significantly reduce packaging functional requirements and would not create a condition that would adversely affect public health and safety.

9.2.3 Level of QA Effort

After determining the applicable QA category, the appropriate level of quality assurance effort for design, procurement, fabrication, testing, operations, maintenance, modification, and repair activities is determined from the eighteen (18) QA requirements identified in 10 CFR Part 71, Subpart H. Specific QA requirements (Level of QA Effort) from Subpart H of 10 CFR 71 relative to packaging activities are categorized in Table 9-3 of the SARP. The eighteen (18) requirements identified in the SARP are as follows: organization; quality assurance program; design control; procurement document control; instructions, procedures, and drawings; document control; control of purchased material, equipment, and services; identification and control of material, parts, and components; control of special processes; inspection control; test control; control of measuring and test equipment; handling, shipping, and storage control; inspection, test, and operating status; control of nonconforming materials, parts, or components;

corrective action; and QA records and QA audits. Each of the eighteen (18) requirements has assigned QA requirements on the basis of the Quality Category, A, B, or C.

9.2.4 Independent Verification

The QA Chapter of the SARP includes independent verification of fabrication, operational, and maintenance activities considered to be critical in satisfying the regulatory requirements, as identified in 10 CFR Part 71. Verification of critical activities is contained in Sections 9.10.1 and 9.10.2 and Tables 9-6 and 9-7 of Chapter 9 of the SARP, which includes inspection criteria for acceptance of the fabricated 9516 packaging components, assembly operations, leak testing, maintenance activities, and package loading.

9.2.5 Application of the ASME Code

Appendix 2.12.12 of the SARP specifies the materials, design, fabrication, testing, and examination requirements for the package CV that comply to the requirements of Section III, Division 1, Subsection NB of the ASME BPVC. Table 2.12.12-1 of the SARP identifies and compares the applicable requirements for Class 1 CV fabrication to the current design of the CV.

9.2.6 Records

Table 9-8 of the SARP specifies which documents are considered to be lifetime records (e.g., the SARP, design drawings, audit reports, and nonconformance reports [and resolutions]). The record retention program specifies that the design authority must retain records for three (3) years beyond the date when the package was last used in a particular activity that is documented by the prescribed records.

9.3 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, the QA program and requirements in Chapter 9 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71, 49 CFR Part 173, and DOE Order 460.1B are met.

Note: Section 9.19, Appendix 9.19.1 of the SARP lists the documents, papers, and reports that are referenced in the SARP for the 9516 Package. The list includes those references added in response to the Q1 questions and the source verification.