SAFETY EVALUATION REPORT

Addendum to the T-3 Safety Analysis Report for Packaging

Docket 91-6-9132

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Transportation and Packaging Safety Division, EH-33.2
Office of Risk Analysis and Technology
U.S. Department of Energy

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1.0 GENERAL

1.1 Area of Review

Revision 3 of the Addendum (Ref. 1) to the Safety Analysis Report for Packaging (SARP) for the T-3 shipping package for spent fuel (Ref. 2) seeks approval for transport of plutonium oxide powder. The PuO_2 powder is to be retained inside EP-61 vessels which are not currently authorized for transport in the T-3 package. Two EP-61 vessels are to be contained in a Fuel Storage Container (FSC). Two FSCs are to be transported in the T-3 package in the currently authorized payload Type 9, Fissile Class III, configuration. Composition of the PuO_2 payload is defined in Section 1.2.3 (Contents of Packaging) of the Addendum. Maximum bulk density of the PuO_2 powder is to be 11 g/cm^3. Each EP-61 vessel can contain up to 403 g of PuO_2 payload and up to the equivalent of 1 g water. This allows up to 806 g PuO_2 payload per FSC, or a maximum T-3 payload of 1612 g. The contents maximum decay heat is limited to 150 W per EP-61 vessel, which is equivalent to 300 W per FSC and 600 W per T-3 payload.

1.2 Findings and Conclusions

The staff has reviewed the T-3 SARP Addendum (Ref. 1) and concludes that it contains the information required to demonstrate compliance with 10 CFR 71 (Ref. 3) and DOE 5480.3 (Ref. 4). This conclusion is based in part on prior reviews of the T-3 SARP (Ref. 2) and its acceptance for certification.

2.0 STRUCTURAL EVALUATION

2.1 Area of Review

The structural analysis that was presented in the T-3 SARP Addendum (Ref. 1) was reviewed for the proposed alternative payload. The proposed payload consists of two EP-61 vessels braced inside an FSC. Each EP-61 vessel can contain PuO_2 having a maximum bulk density of 11 g/cm^3. Two holders and three spacer assemblies are used to center the two EP-61 vessels axially in the FSC and to separate the vessels within the primary containment vessel (FSC). The weights of the individual components within the FSC are:

EP-61 vessel:	14.89 lb (2)
Holder:	2.00 lb (2)
Spacer:	0.24 lb (3)
Total weight of components in the FSC:	34.5 lb

2.2 Findings and Conclusions

The stress calculations presented in the T-3 SARP Addendum (Ref. 1), for stresses generated by the proposed alternative payload arrangement inside the FSC under the hypothetical accident conditions, have been reviewed. The margin of safety is considered to be adequate for the FSC to contain the proposed alternative payload. Thus, the staff concludes that the T-3 revisions defined in Section 2 of the T-3 SARP Addendum (Ref. 1) are in compliance with the requirements of 10 CFR 71 (Ref. 3).

3.0 THERMAL EVALUATION

3.1 Area of Review

The thermal evaluation presented in the T-3 SARP Addendum (Ref. 1) was reviewed for the proposed payload. The thermal performance of the T-3 package when transporting the proposed payload has been reviewed to assure that the critical components of the package and its contents would not be impaired to the extent that contents would be released during the normal transport and hypothetical accident conditions defined in 10 CFR 71.

Decay heat from the PuO_2 powder payload is limited to 150 W for each EP-61 vessel, 300 W for each FSC, or 600 W for the T-3 package.

3.2 Acceptance Criteria

The requirements of 10 CFR 71.51 can be satisfied if temperatures and pressures in the containment system do not exceed those allowed for containment as defined in the T-3 SARP (Ref. 2). Thermal expansion of components in the FSC must not result in interference stresses in the FSC.

3.3 Review Procedure

The EP-61 vessel assembly and its contents are assumed to be the same as described in the SARP for the Westinghouse Savannah River PuO_2 5320 shipping package (Ref. 5). That is, the EP-61 contains an EP-60 product canister containing PuO_2 powder. The EP-60 canisters are sealed with a threaded cap and metal O-ring. The EP-61 vessels are sealed with a threaded plug and cap and two silicone rubber O-rings. (In the current EP-61 design one of the rubber O-rings is replaced by a metal seal.) The two EP-61 vessels are contained by an FSC assembly which includes inner and outer vessels that are weld sealed.

Data presented in the T-3 SARP indicates that, for 300 W contents heat, the FSC temperature is 410°F. Assuming only one-dimensional radial radiation heat transfer from the EP-61 to the FSC, the EP-61 maximum temperature during normal conditions is 900°F. Corresponding temperatures of the EP-60 and the PuO_2 powder, derived from the 5320 SARP (Ref. 5), are 925°F and 1210°F, respectively. This is a conservative method for estimating these temperatures because heat transfer would actually include convection and also be two-dimensional.

Estimates of maximum temperatures in the package during hypothetical accident thermal test conditions are given in the T-3 SARP (Ref. 2) for contents heat greater than that specified for the proposed PuO_2 payload. Thus, package temperatures during hypothetical accident thermal test conditions are expected to be below allowed limits defined in the T-3 SARP (Ref. 2). Analytical data in the T-3 SARP indicates that the maximum temperature increase of the FSC during hypothetical accident thermal test conditions is approximately 80°F. A conservative method for estimating EP-60 and EP-61 temperatures is to increment them this amount above the normal condition temperatures given in the preceding paragraph.

EP-61 temperatures during hypothetical accident thermal test conditions are high enough to cause decomposition and outgassing of the rubber O-rings in the EP-61. This gas generation will contribute to pressures in the EP-60 and EP-61. A method for estimating these pressures is described in the 5320 SARP (Ref. 5). Based on experimental data, a conservative estimate of the amount of gas generated by the two O-rings is 0.111 moles and the total amount of gas in the EP-61 is 0.197 moles. Since the FSC is the primary containment vessel, it is assumed that gas pressures in the EP-60 and EP-61 vessels are vented into the FSC. For this condition and when maximum temperatures occur during the hypothetical accident thermal test, the pressure contained by the FSC inner vessel is approximately 48 psig. If the gases are vented to the FSC outer vessel, the pressure is approximately 40 psig. Stresses developed in the FSC vessels by these pressures are less than 1,000 lb/in^2, which is well below stress limits (20,000 lb/in^2) for the FSC.

Pressures developed during normal conditions will be less than pressures developed during hypothetical accident conditions. Stresses developed in the FSC during the hypothetical accident impact test are also well below allowed stresses. Pressure stress contributes only a small fraction to the total stress (i.e., <3 percent). The margin of safety for stress will be not less than 25 percent.

3.4 Findings and Conclusions

Based on the review described above, temperatures, pressures, and stresses in the FSC are within allowed limits. Thermal expansion estimates of all components in the FSC do not result in interference stresses.

The staff concludes that the thermal design features described in Section 3 (Thermal Evaluation) of the T-3 SARP Addendum (Ref. 1) will assure compliance with the performance requirements of 10 CFR 71.

4.0 CONTAINMENT EVALUATION

The primary and secondary containments are the FSCs and T-3 cask. These containments are described in the T-3 SARP (Ref. 2) and were previously reviewed and certified for the Type 9 payloads as being in compliance with the requirements of 10 CFR 71. These containments are applicable to the PuO_2 powder payload defined in the T-3 SARP Addendum (Ref. 1). Thus, the staff's conclusion of a review of Section 4 (Containment) of the Addendum is the containment design features will assure compliance with the containment requirements of 10 CFR 71.

5.0 SHIELDING EVALUATION

5.1 Area of Review

The requirement specified in 10 CFR 71.47 (Ref. 3) is that the dose equivalent rate at any point on the external surface of the package does not exceed 200 mrem/hr.

Requirements specified in 10 CFR 71.51(a)(1) are that under normal conditions of transport, there be no significant increase in external radiation levels and no substantial reduction in the effectiveness of the packaging under the tests specified in 10 CFR 71.71.

The requirement specified in 10 CFR 71.51(a)(2) is that under hypothetical accident conditions, the dose equivalent rate at any point 1 meter from the external surface of a Type B package does not exceed 1000 mrem/hr under the tests specified in 10 CFR 71.73.

Approval is requested for the transport of PuO_2 powder in the T-3 Cask using the existing and approved payload Type 9 configuration. The powder is to be retained within EP-61 vessels, and two such vessels may be braced and centered axially within a single fuel storage container. Transport of one or two fuel storage containers is currently authorized.

With the approved SARP (Ref. 2) showing the side surface of the cask as the location of maximum dose equivalent rate, the staff's review focussed on the relative difference between the dose equivalent rates at the side surface of a single cask from the transport of PuO_2 powder and the dose equivalent rates at the side surface of a single cask associated with the transport of materials in the approved payload configurations of the SARP (Ref. 2). In the context of this review, the dose equivalent rates associated with the transport of materials in the approved payload configurations of the SARP (Ref. 2) are assumed to have been calculated correctly.

5.2 Acceptance Criteria

Cask shielding is deemed acceptable if it can be shown that the dose equivalent rate at the side surface of a single cask from the transport of PuO_2 powder is less than the side surface dose equivalent rates in the approved SARP (Ref. 2).

5.3 Review Procedure

The review is divided into three main parts: (1) source specification, (2) model specification, and (3) shielding evaluation.

5.3.1 Source Specification

The review is divided into two parts: (1) gamma source and (2) neutron source.

5.3.1.1 Gamma Source

The original gamma source for the approved SARP (Ref. 2) is addressed in Sections 5.2 (Source Specification) and 5.2.1 (Gamma Source). It is a payload of 21 carbide/nitride pins with a 90-day cooling period following irradiation to a maximum fuel burnup of 80 MWd/kg. The source consists of 9.5 kg of fuel material with a composition of 20 percent PuC or PuN and 80 percent UC or UN. The pins are 0.315 inches in diameter, with a 36.00-inch active fuel length. The gamma emission spectrum for the fuel is calculated with RIBD-II and tabulated in a 16-group structure of mean gamma energies. Bounding limits for each of the 16 energy groups are not provided. Activation of the fuel pin support structure hardware and its associated gamma emission spectrum is not discussed.

Sodium-bonded metal and carbide fuel pins are also evaluated. Their gamma sources are addressed in Appendix 5.5.4 (Shielding Evaluation of Sodium-Bonded Metal Fuel Pins and Carbide Fuel Pins) of the approved SARP (Ref. 2). The two sodium-bonded metal sources consists of 21 pins with a 90-day cooling period following irradiation to a maximum fuel burnup of 200 MWd/kg. Each metal source consists of 5.326 kg of fuel material with one composed of 20.00 percent Pu, 60.00 percent U-238, and 20.00 percent U-235 and the other composed of 60.00 percent U-238 and 40.00 percent U-235. The sodium-bonded carbide source consists of 24 pins with a 90-day cooling period following irradiation to a maximum fuel burnup of 70 MWd/kg. The carbide source consists of 12.343 kg of fuel material with a composition of 23.00 percent Pu, 76.45 percent U-238, and 0.55 percent U-235. The active fuel length is 36.00 inches for both source types. A gamma emission spectrum is calculated for each fuel source using the fission product curie inventories from ORIGEN2. Each spectrum is tabulated in a 16-group structure of mean gamma energies identical to that from RIBD-II. Again, bounding limits for each of the 16 energy groups are not provided. Activation of the fuel pin support structure hardware and its associated gamma emission spectrum is described as negligible but is not tabulated.

The gamma source for the Addendum is addressed in Sections 5.1 (Discussion and Results) of Revision 2 and Sections 5.2 (Source Specification) and 5.2.1 (Gamma Source) of Revision 3 (Ref. 1). The gamma emission spectrum is based on 1.612 kg of PuO_2 powder. It is derived from the original source data and is tabulated in a 16-group structure of mean gamma energies identical to that from RIBD-II. Bounding limits for each of the 16 energy groups are, once again, not provided. In addition, major decay-daughter contributors to the gamma emission spectrum are not discussed.

5.3.1.2 Neutron Source

The original neutron source for the approved SARP (Ref. 2) is addressed in Sections 5.2 (Source Specification) and 5.2.2 (Neutron Source). It is a

217-pin FTR driver assembly with a 350-day cooling period following irradiation to a maximum fuel burnup of 80 MWd/kg. The source consists of 38 kg of fuel material with a composition of 25 percent PuO_2 and 75 percent uranium oxide. The pins are 0.23 inches in diameter, with a 36.00-inch active fuel length. Neutron emission is apportioned as 90 percent from spontaneous fission and 10 percent from alpha-neutron reactions between fission alphas and oxygen in the fuel. The primarily source of the spontaneous fission neutrons is Cm-242. The neutron emission spectrum is tabulated in a 15-group structure of mean neutron energies. Bounding limits for each of the 15 energy groups are not provided.

Sodium-bonded metal and carbide fuel pins are also evaluated. Their neutron sources are addressed in Appendix 5.5.4 (Shielding Evaluation of Sodium-Bonded Metal Fuel Pins and Carbide Fuel Pins) of the approved SARP (Ref. 2). The two sodium-bonded metal sources consist of 21 pins with a 90-day cooling period following irradiation to a maximum fuel burnup of 200 MWd/kg. Each metal source consists of 5.326 kg of fuel material with one composed of 20.00 percent Pu, 60.00 percent U-238, and 20.00 percent U-235 and the other composed of 60.00 percent U-238 and 40.00 percent U-235. The carbide source consists of 24 pins with a 90-day cooling period following irradiation to a maximum fuel burnup of 70 MWd/kg. The carbide source consists of 12.343 kg of fuel material with a composition of 23.00 percent Pu, 76.45 percent U-238, and 0.55 percent U-235. The active fuel length is 36.00 inches for both source types. Only total source strengths are provided for both source types. Neutron emission spectra and the apportionment of the neutron emission spectrum between spontaneous fission and alpha-neutron reactions are not tabulated or discussed.

The neutron source for the Addendum is addressed in Sections 5.1 (Discussion and Results) and 5.2.2 (Neutron Source) of Revision 2 and Section 5.2 (Source Specification) of Revision 3 (Ref. 1). The neutron emission spectrum is based on 1.612 kg of PuO_2 powder. It is derived from the original source data and is tabulated in a 15-group structure of mean neutron energies identical to that from the approved SARP (Ref. 2). Bounding limits for each of the 15 energy groups are, once again, not provided. Apportionment of the neutron emission spectrum between spontaneous fission and alpha-neutron reactions is also not discussed.

5.3.2 Model Specification

The review is divided into two parts: (1) description of the radial and axial shielding configuration and (2) shield regional densities.

5.3.2.1 Description of the Radial and Axial Shielding Configuration

Radial and axial shielding configurations used in analyses of the original gamma and neutron sources are addressed in Section 5.3.1 (Description of Radial and Axial Shielding Configuration) of the approved SARP (Ref. 2). No distinction is made in the radial and axial shielding configuration between normal and accident conditions. The 21 carbide/nitride-pin gamma source and 217-pin FTR driver assembly neutron source are modeled as homogenized cylinders of equal length and 7.00 inches and 4.00 inches diameter, respectively.

The length of each cylinder is initially set at 36.00 inches and the source regions are centered, radially and axially, in the cask cavity, 54.00 inches from each end. In subsequent analyses, the length for each cylinder is adjusted to 96.00 inches with the tops and bottoms of the source regions positioned 13.50 inches from the lower surface of the shielded plug and 40.50 inches from top surface of the pusher plug, respectively.

Radial and axial shielding configurations used in analyses of the sodium-bonded metal and carbide fuel pins are addressed in Appendix 5.5.4 (Shielding Evaluation of Sodium-Bonded Metal Fuel Pins and Carbide Fuel Pins) of the approved SARP (Ref. 2). No distinction is made in the radial and axial shielding configuration between normal and accident conditions. The 21 sodium-bonded metal pin and 24 sodium-bonded carbide pin gamma sources are modeled as homogenized cylinders of 36.00 inches length and 5.00 inches diameter. Homogenized models of a 5.75-inch long lower axial reflector, a 1.40-inch long bottom end cap region, and a 1.33-inch long top end cap region are also included. The top and bottom of the fuel region are positioned 69.45 inches below the lower surface of the shielded plug and 41.67 inches above the top surface of the pusher plug, respectively. This particular location corresponds to a shipment within an Ident 1578 container.

The source region radial and axial shielding configuration for the Addendum is addressed in Sections 1.1 (Introduction) and 1.2.1 (Packaging) of Revision 2 (Ref. 1). The PuO_2 powder is retained inside EP-61 vessels, with two such vessels in each of two fuel storage containers. Each fuel storage container is made of 41.00 inch long, 5-inch Schedule 40 stainless steel pipe and is seal welded at both ends. On the end opposite to the closure, a hemispherical 0.25-inch rod is attached to facilitate handling operations. Axial positioning of the two EP-61 vessels within the each fuel storage container is accomplished through the use of two holders and three spacers, each of which is made of aluminum. Positioning of the two fuel storage containers is addressed only through the reference to the payload Type 9 configuration.

5.3.2.2 Shield Regional Densities

Source and shield region densities used in analyses of the original gamma and neutron sources are addressed in Section 5.3.2 (Shield Region Densities) of the approved SARP (Ref. 2). No distinction is made in the source and shield region densities between normal and accident conditions. The 21-pin carbide/nitride gamma source and 217-pin FTR driver assembly neutron source are modeled as homogenized cylinders of fuel and clad and have material densities of 0.46 g/cm^3 and 6.46 g/cm^3, respectively. Masses associated with the fuel pin canister and assembly support structure are conservatively neglected.

Source region densities used in analyses of the sodium-bonded metal and carbide fuel pins are addressed in Appendix 5.5.4 (Shielding Evaluation of Sodium-Bonded Metal Fuel Pins and Carbide Fuel Pins) of the approved SARP (Ref. 2). No distinction is made in the source region densities between normal and accident conditions. The 21 sodium-bonded metal pin and 24 sodium-bonded carbide pin gamma sources are modeled as homogenized cylinders of fuel and clad and have material densities of 0.528 g/cm^3 and 1.202 g/cm^3, respectively.

Source region densities for the Addendum are addressed in Section 1.1 (Introduction) of Revision 2 and Section 1.2.3 (Contents of Packaging) of Revision 3 (Ref. 1). The density of the PuO_2 powder is stated to vary between 0.0 g/cm^3 and 11.00 g/cm^3.

5.3.3 Shielding Evaluation

The shielding evaluation for the original gamma and neutron sources is addressed in Section 5.4 (Shielding Evaluation) of the approved SARP (Ref. 2). No distinction is made in the shielding evaluation between normal and accident conditions. Gamma dose equivalent rates are calculated with the three-dimensional point kernel code QAC (a derivative of QAD). Buildup factor coefficients used in these QAC analyses of the cask wall and ends shielding are the Taylor coefficients for lead and steel, respectively. Neutron and secondary gamma dose equivalent rates are calculated with the one-dimensional discrete ordinates code ANISN.

The shielding evaluation for the sodium-bonded metal and carbide fuel pins is addressed in Appendix 5.5.4 (Shielding Evaluation of Sodium-Bonded Metal Fuel Pins and Carbide Fuel Pins) of the approved SARP. No distinction is made in the shielding evaluation between normal and accident conditions. Gamma dose equivalent rates are calculated by a point kernel version of the three-dimensional Monte Carlo code MCNP. Neutron dose equivalent rates are estimated from the product of the source strengths ratio (sodium-bonded fuel/217-pin FTR driver assembly fuel) and the 217-pin FTR driver assembly dose equivalent rates of the approved SARP.

The staff shielding evaluation is performed in two parts. The first part involves a comparison of the side surface gamma and neutron dose equivalent rates from the PuO_2 gamma and neutron sources in the Addendum (Ref. 1) with side surface gamma and neutron dose equivalent rates from the original 21-pin carbide/nitride gamma and 217-pin FTR driver assembly neutron sources of the approved SARP (Ref. 2). The second part involves an evaluation of the effect of source region density variations on the side surface gamma and neutron dose equivalent rates. No distinction is made in the shielding evaluation between normal and accident conditions.

Computer codes used in the staff shielding evaluation are MicroShield and COG. MicroShield is a microcomputer adaptation of ISOSHLD, a point kernel code used for the evaluation of gamma dose equivalent rates. COG employs the Monte Carlo method for transport and is used to evaluate the neutron dose equivalent rates. With MicroShield, the staff used the cylindrical source from side-cylindrical shields geometry. The staff determined the gamma dose equivalent rates at the side surface of the cask on the source midplane. Buildup factor coefficients employed in these MicroShield analyses are the Taylor coefficients for lead. With COG, the staff used a three-dimensional finite cylinder analysis. The staff determined the average neutron dose equivalent rates at the side surface of the cask. Areas used in the determination of these average neutron dose equivalent rates are over the length of the active fuel source.

Assumptions made in the first part of the staff shielding evaluation are: (1) the 1.612 kg mass of PuO_2 powder is not distributed but concentrated into a single source region; (2) the mass of the EP-61 vessels is conservatively neglected; (3) the radial and axial shielding configuration and associated source and shield region densities for the 21-pin carbide/nitride source is common for the comparison of the side surface gamma dose equivalent rates from the PuO_2 powder gamma source in the Addendum (Ref. 1) with the side surface gamma dose equivalent rates from the original 21-pin carbide/nitride gamma source of the approved SARP (Ref. 2); and (4) the radial and axial shielding configuration and associated source and shield region densities for the 217-pin FTR driver assembly source is common for the comparison of the side surface neutron dose equivalent rates from the PuO_2 powder neutron source in the Addendum (Ref. 1) with the side surface neutron dose equivalent rates from the original 217-pin FTR driver assembly neutron source of the approved SARP (Ref. 2). To establish a baseline, side surface gamma and neutron dose equivalent rates are first calculated for the original 21-pin carbide-nitride gamma and 217-pin FTR driver assembly neutron sources. Next, the gamma and neutron source terms for the PuO_2 powder are substituted into the models and the side surface dose equivalent rates recalculated. Results are then compared.

Assumptions made in the second part of the staff shielding evaluation are: (1) the gamma and neutron source terms are those of the PuO_2 powder; (2) the 1.612 kg mass of PuO_2 powder is not distributed but concentrated into a single source region; (3) the mass of the EP-61 vessels is conservatively neglected; (4) the radial and axial shielding configuration and associated shield region densities for the 21-pin carbide/nitride source is common for the evaluation of the side surface gamma dose equivalent rates from the PuO_2 powder gamma source; (5) the gamma source region densities are 0.046 g/cm^3, 4.6 g/cm^3, and 11.0 g/cm^3; (6) the radial and axial shielding configuration and associated shield region densities for the 217-pin FTR driver assembly source is common for the evaluation of the side surface neutron dose equivalent rates from the PuO_2 powder neutron source; and (7) the neutron source region densities are 0.0646 g/cm^3, 0.646 g/cm^3, and 11.0 g/cm^3. Side surface gamma and neutron dose equivalent rates are calculated at each source region density and the results evaluated.

5.4 Findings and Conclusions

Side surface gamma and neutron dose equivalent rates from a single cask from the transport of PuO_2 powder are less than the side surface gamma and neutron dose equivalent rates in the approved SARP (Ref. 2) for the configurations and regional densities of the 21-pin carbide/nitride gamma and 217-pin FTR driver assembly neutron sources. For the gamma and neutron source region densities reviewed, the effect of source region density variability is one of inverse proportionality on the side surface gamma dose equivalent rates and a trend of direct proportionality on the side surface neutron dose equivalent rates. In those instances where the source region density variability leads to a relative increase in the side surface dose equivalent rate, the magnitude of this increase is less than the side surface dose equivalent rate in the approved SARP (Ref. 2).

This section of the applicant's SARP has been reviewed to determine that the shielding design features have been designed in a manner that will assure compliance with the performance requirements of 10 CFR 71.47 and 10 CFR 71.51 for a general package under normal conditions of transport and hypothetical accident conditions. The scope of the review covers the shielding design features of the package, the source and model specifications, the shielding evaluation, and supportive information or documentation.

Basis for acceptance in the review has been conformance with established guidelines and criteria. The evaluation of the shielding design provides reasonable assurance that, under normal conditions of transport and hypothetical accident conditions, radioactive material can be safely transported in the package.

The staff concludes that the protective features provided in the design of the package conform to applicable Regulations, Regulatory Guides, and industry standards, and are acceptable.

6.0 CRITICALITY EVALUATION

6.1 Area of Review

The criticality evaluation included in the T-3 SARP Addendum (Ref. 1) is based upon the statement "The criticality evaluation provided in the T-3 SARP is for a greater fissile loading in an unrestricted array [payload 9]. The evaluation is valid for the proposed payload. No further evaluation is necessary." This review focuses upon verifying the fissile loading and verifying the subcriticality of the inventory and geometry of fissile material within the EP-61 vessel which is inside the fuel storage container when loaded in the T-3 spent fuel shipping cask.

The proposed payload consists of PuO_2 in the form of powder, with a bulk density between 0 and 11 g/cm^3, loaded into an EP-61 vessel. The maximum loading in an EP-61 vessel is 403 g. Two EP-61 vessels are loaded into a single fuel storage container within a fuel assembly holder to assure that the centerlines of the EP-61 and the fuel storage container are coincident. Three spacers are placed around two EP-61 vessels and their holders to maintain longitudinal positioning of the EP-61 vessels. The EP-61 vessel is a right circular cylinder (cavity dimensions: 1.730+0.060-0.000 in. diameter, 14.60±0.11 in. long) with an enlarged end that incorporates two threaded joints with O-rings which is seal welded after loading and assembly. Two fuel storage containers are loaded into the cavity of the T-3 spent fuel cask so there are a total of four EP-61 vessels in a fully loaded T-3 spent fuel cask.

The description of the contents of the packaging in the T-3 SARP Addendum (Ref. 1) identifies the bounding fractional composition of the contents.

6.2 Acceptance Criteria

The requirements of 10 CFR 71.61 can be met if it is demonstrated that the T-3 cask remains subcritical for all conceivable configurations and environments. This is based upon a single cask in each shipment.

6.3 Review Procedure

The inventory of fissile and non-fissile nuclides within the EP-61 vessel is specified by the bound on the contents and the upper bounds of the fractional composition of the contents of the EP-61 vessel as indicated in Section 1.2.3 (Contents of Packaging) of the T-3 SARP Addendum (Ref. 1). The specific values of the inventories of each of the isotopes in the payload are presented in Table 6.1 where the fractional compositions of the fissile isotopes are at their upper bounds. The fissile inventory is 42.3 g for an EP-61 vessel which is less than the fissile inventory represented by payload 9 (4 kg) in the fuel storage container.

ANSI/ANS-8.15-1981 presents multi-parameter limits for nuclear criticality control of special actinide nuclides. Paragraph 6.1 of this standard states: "In PuO_2-H_2O mixtures, regardless of the H/Pu atomic ratio, a subcritical limit of 8 kg of plutonium is valid provided the plutonium contains at least 67 percent Pu-238, provided the isotopic concentration of Pu-241 is less than that of Pu-240, and provided the surrounding materials, including other nearby fissionable materials, can be shown to increase k-effective no more than enclosing the unit by a contiguous layer of water of unlimited thickness." This encompasses the case where the cavity of the T-3 spent fuel cask is filled with water and the cavity of the EP-61 vessel is filled with a mixture of water and plutonium (and other actinide isotopes) such that optimum moderation is achieved. The maximum mass of all isotopes within the EP-61 vessel is 403 g, which is much less than the 8 kg limit indicated in the ANSI standard. The steel ends of the EP-61 vessel and the water-filled region between vessels approach neutronic isolation so that each EP-61 vessel can be considered as a separate entity; however, the four EP-61 vessels in total are within the 8 kg limit.

ANSI/ANS-8.1-1983 presents single-parameter limits for criticality control of fissile materials outside reactors. Paragraph 5 of this standard states: "A limit may be applied to a mixture of fissile nuclides by considering all components of the mixture to be the one with the most restrictive limit." This is combined with the criticality control limits presented in ANSI/ANS-8.15/1981 to address the actinide isotopes other than Pu-238, Pu-239 and Pu-241 that are part of the proposed payload. Table 6.2 summarizes the subcritical mass limits for these actinide nuclides.

Table 6.1 Inventories of Fissile and Non-Fissile Isotopes in EP-61 Vessel

Isotope	% Pu	g(metal)	g(oxide)
Pu-236	≤0.0002	<0.0007	0.0008
Pu-238	≤89.0	<316.2	<358.7
Pu-239	≤24	<85.3	<96.7
Pu-240	≤3.2	<11.4	<12.9
Pu-241	≤0.6	<2.13	<2.41
Pu-242	≤0.2	<0.71	<0.80
Am-241	≤0.00028	<0.001	<0.0011
U-234	≤0.00028	<0.001	<0.0011
Th-232	≤0.00023	<0.0008	<0.0009
Np-237	≤0.00022	<0.0008	<0.0009
Total Inventory:		355.3	≤403.0
Total Plutonium:		355.2	≤403.0
Total Fissile Inventory:		87.4	≤99.1

Table 6.2 Comparison of Criticality Mass Limits and Contents

Isotope	Limit for Sub-Criticality	EP-61 Contents	T-3 Contents
$(Pu\text{-}240)O_2$	70 kg	0.403 kg*	1.612 kg*
Pu-242	60 kg	0.355 kg**	1.420 kg**
$(Am\text{-}241)O_2$	40 kg	0.403 kg*	1.612 kg*
$(Np\text{-}237)O_2$	140 kg	0.403 kg*	1.612 kg*
Pu-239	0.450 kg	0.087 kg***	0.348 kg***
Pu-241	0.200 kg	0.002 kg***	0.008 kg***

* Mass of oxides of all isotopes in vessel
** Mass of heavy metal of all isotopes in vessel
*** Mass of heavy metal of Pu-239 and Pu-241

6.4 Findings and Conclusions

The most restrictive mass limits for criticality control presented in ANSI/-ANS-8.15/1981 presented in the previous section, indicate that the inventory of fissile and non-fissile isotopes within the EP-61 vessel will remain subcritical when inserted into a Spent Fuel Container which is then inserted in the cavity of the T-3 spent fuel shipping cask. The conservatism of the values presented in the standards is assured by the peer review that the standards received prior to publication and adoption within the nuclear industry.

On the basis of the comparisons between the mass limits for criticality control in the ANSI standards and the inventories of fissile isotopes in the proposed payload, the staff concludes that the T-3 cask is designed to maintain its contents in a subcritical state when containing the proposed payload and only one cask in a shipment in compliance with 10 CFR 71.61 during transportation and storage.

7.0 OPERATING PROCEDURES

The operating procedures described in the T-3 SARP (Ref. 2) are applicable to the plutonium payload defined in the T-3 SARP Addendum (Ref. 1). Additional operating procedures specified in the Addendum will verify that prior to loading the EP-61 vessels into the FSCs, the PuO_2 payload contained in the EP-61 vessels is within the bounds for mass and isotopic mix as specified in the T-3 SARP Addendum.

This section of the T-3 SARP Addendum has been reviewed and determined to contain operating procedures that have been defined in a manner that will assure compliance with requirements of 10 CFR 71 (Ref. 3) and DOE 5480.3 (Ref. 4).

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The acceptance tests and maintenance program presented in the T-3 SARP (Ref. 2) are for the Type 9 payloads. The payloads described in the T-3 Addendum (Ref. 1) do not increase the containment pressure, package structure temperatures, or impact loads above allowed limits. Thus, no revision of the acceptance tests or maintenance program is required.

This section of the T-3 SARP Addendum has been reviewed and the staff concludes that it contains acceptance tests and a maintenance program that have been defined in a manner that will assure compliance with the requirements of 10 CFR 71 (Ref. 3).

9.0 REFERENCES

1. Addendum to the Consolidated Safety Analysis Report for the T-3 Spent Fuel Shipping Cask Demonstrating Compliance to the Requirements of 10 CFR 71, Rev. 3, Westinghouse Hanford Co., Richland, WA 99352, December 1991.

2. Consolidated T-3 Spent Fuel Shipping Cask Safety Analysis Report, Rev. 4, Nuclear Packaging, Inc., Federal Way, WA, 1986 (Revisions 5 & 6 added by Westinghouse Hanford Co., Richland, WA 99352).

3. United States Nuclear Regulatory Commission, Packaging and Transportation of Radioactive Material, Title 10 Code of Federal Regulations, Part 71 (10 CFR 71), Office of the Federal Register, National Archives and Records Administration, Washington, DC.

4. U.S. Department of Energy, DOE 5480.3: Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substance, and Hazardous Wastes, U.S. Department of Energy, Washington, DC, August 9, 1985.

5. Safety Analysis Report - Packages Pu Oxide and Am Oxide Shipping Cask - Draft Report, Rev. 2, Westinghouse Savannah River Company, January 1990.