SAFETY EVALUATION REPORT

Addendum to T-3 Safety Analysis Report for Packaging
for Shipment of FSP-1R Fuel Assemblies

Docket No. 91-36-9132

April 14, 1992

U.S. Department of Energy
Division of Transportation and Packaging Safety, EH-33.3

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1.0 GENERAL

1.1 Area of Review

The Addendum to the T-3 SARP for the shipment of FSP-1R Fuel assemblies in the T-3 Cask (Ref. 1) seeks authorization to ship six Space Reactor Test Program Fuel assemblies in the T-3 shipping cask as Fissile Class III. The fuel assemblies, from an experiment called FSP-1R, support the Space Reactor Test Program (SP-100) at Hanford. Composition of the FSP-1R payload is defined in Section 1.2.3 (Contents of Packaging) of the Addendum. The fuel assemblies contain columns of uranium nitride pellets clad with Nb-1Zr or rhenium. The encapsulated uranium nitride is surrounded by lithium, which is doubly encapsulated. Lithium is the reactor heat transfer medium under investigation.

Fuel pins containing uranium nitride are an approved payload in the T-3, as described under Payload 9. The FSP-1R fuel assemblies will be contained in the Ident 69 pin container, which is also an approved T-3 fuel pin container.

The payload weight and decay heat are lower in comparison to currently licensed payloads. The FSP-1R payload contains lithium which has not been previously addressed in the T-3 Safety Analysis Report for Packaging (SARP) (NUPAC, 1986).

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 six FSP-1R fuel assemblies shipped within the Ident 69 pin container of the T-3 Spent Fuel Cask. Each of the assemblies is placed in one of the six outer radial compartments of the container. The compartment is not completely filled by the fuel assembly and the assembly is free to move within the compartment. The fuel assembly has a diameter of 0.831 in., and the Ident 69 compartment has an average width of approximately 1.8 in. in the radial and circumferential directions.

Each of the fuel assemblies is made of two circular cylindrical outer capsules connected in series by welding. Each outer capsule is made of 20% cold-worked 314 stainless steel and has an outer diameter of 0.831 in. and a thickness of 0.025 in. Each of the capsules contains a diameter of 0.75 in. and a thickness of 0.025 in. The inner capsule is filled with lithium with a clad fuel pin of 0.3 in. O.D. imbedded along the capsule centerline. The lithium in each capsule weighs 0.073 lb. (33 g.).

The maximum weight of the fuel assembly is 5.83 lb., and the estimated weight of the Ident 69 container is 130 lb. Thus, the total proposed payload weighs 166 lbs., which is only 24% of the maximum allowable pay load of 700 lb. for the T-3 shipping cask.

2.2 Findings and Conclusions

Since the weight of the proposed payload is considerably below the maximum allowable payload weight for the T-3 cask, there is little doubt that the cask can safely carry the proposed payload. The only remaining concern is whether or not the fuel assembly outer shell can confine the lithium for all of the normal and accident transportation conditions. The staff determines that it can, based on the analysis results and margins of safety presented in the T3-SARP Addendum.

The T3-SARP addendum has analyzed all potential failure modes of the outer capsule and has shown that the fuel assembly has sufficient margin of safety to prevent permanent deformation of the outer capsule under the worst free drop conditions. The staff has also determined that the margins of safety demonstrated by the analyses are sufficient to cover all uncertainties about the analyses including two major uncertainties which the SARP omitted to address: 1) that the local buckling stress of the outer capsule can be degraded by geometric imperfections and inelastic deformations of the capsule and 2) that the impact acceleration of the fuel assembly can be raised by a secondary impact between the fuel assembly and its container (Ident 69).

Based on the preceding considerations, 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.

The contents of the package consist of 12 FSP-1R fuel pins in an Ident 69 pin container. The decay heat from each of the uranium nitride fuel pins after 180 days is about 4.5 W. The total decay heat load in a package is about 53.1 W.

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).

3.3 Review Procedure

The package design, including containment and shield has been approved for a content decay heat load of 300 W. Thus, since the content of this package produce a decay heat load of less than 60 W (which is less than for the approved package), the impact of the thermal loads on the containment and lead shielding should be less than that of the approved package. Therefore, the containment and lead shielding conform to the requirements of 10 CFR 71 (Ref. 3).

To examine the design criteria that the lithium temperature in the fuel assembly does not exceed the melting point of the lithium, a one-dimensional steady state heat transfer analysis was performed for normal conditions. The analysis assumed that the heat from the fuel rod flows radially (but not axially) from the cylindrical surface of the rods. In addition, the gaps within the containment boundary are filled with argon (at one atmosphere). The analysis assumes that the heat is transferred radially across the gaps only by conduction through the gas fill and by thermal radiation. Convection is assumed not to exist within the gaps. The maximum temperature of the lithium is calculated to be 328°F, which is less than the melting point of lithium. Note that this is a very conservative method for estimating the lithium temperature because the actual heat transfer includes convection in the gas filled (argon) gaps and is also 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 FSP-1R payload. Thus, package temperatures as well as pressures and thermal stresses during hypothetical accident thermal test conditions are expected to be less than the allowed limits defined in the T-3 SARP (Ref. 2). The lithium may melt during the hypothetical accident. The staff has evaluated the possible consequences of the highly improbable event wherein the lithium melts during a hypothetical accident and leaks into the containment vessel. Since water cannot access the containment vessel during or after a hypothetical accident, an exothermic chemical reaction involving the lithium within the containment vessel will not occur. Thus, the melting of the lithium during the hypothetical accident condition will not result in the breaching of the containment with a release of radionuclides greater than allowed by 10 CFR 71.51 (a)(2).

3.4 Findings and Conclusions

Based on the review described above, temperatures, pressures, and stresses in the T-3 package are within allowed limits.

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 containment features associated with shipments of the FSP-1R fuel assembly payload remain the same as those presented in the T-3 SARP (Ref. 2) with the cask providing containment under both normal and hypothetical accident conditions. The features are also the same for the Ident 69 pin container, which is an approved T-3 fuel container. Thus, the staff's conclusion of a review of Section 4 (Containment) of the Addendum is that the containment design features will assure compliance with the containment requirements of 10 CFR 71.

5.0 SHIELDING EVALUATION

5.1 Area of Review

10 CFR 71, paragraph 71.47 requires that the dose equivalent rate at any point on the external surface of the package does not exceed 200 mrem per hour.

10 CFR 71, paragraph 71.51(a)(1) requires 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 paragraph 71.71.

10 CFR 71, paragraph 71.51(a)(2) requires 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 per hour under the tests specified in paragraph 71.73.

Approval is requested for the transport of six Space Reactor Test Program fuel assemblies from the FSP-1R experiment in the T-3 Cask using the existing and approved Ident 69 container. Each fuel assembly contains two fuel capsules. Each capsule consists of enriched uranium nitride (UN) pellets clad with Nb-1Zr or rhenium, which is surrounded by a lithium-filled Ti-Zr-Mo alloy capsule, which is further surrounded by a helium-argon filled stainless steel capsule. The six FSP-1R fuel assemblies will be contained within the outer six radial compartments of the Ident 69, with the center compartment remaining vacant. Transport of fuel pins containing UN is currently authorized in the payload Type 9 configuration.

This review focuses on the expected difference between the dose equivalent rates at the surface of a single cask from the transport of six FSP-1R fuel assemblies and the dose equivalent rates at the surface of a single cask associated with the transport of materials in the approved payload configurations of the SARP. In the context of this review, the dose equivalent rates associated with the transport of materials in the approved payload configurations of the SARP are assumed to have been calculated correctly.

5.2 Acceptance Criteria

Cask shielding is deemed acceptable if it can be shown that the expected dose equivalent rate at the surface of a single cask from the transport of six FSP-1R fuel assemblies is less than the surface dose equivalent rates in the approved SARP.

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 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. The two sodiumbonded 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. A gamma emission spectrum is calculated for each fuel source with ISOSHLD 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), 5.2 (Source Specification), and 5.2.1 (Gamma Source) of Revision 0. The fuel source consists of 12 fuel capsules with a 180-day cooling period following irradiation for 669.1 effective full power days at a peak flux of 3.178×10^{15} neutrons/second. The mass of the UN in each fuel capsule is 61.9 g of which 76.72 percent is U-238, 18.42 percent is U-235, and 4.86 percent is N. The mass of the structural material in each fuel capsule is 515.63 g of which 31.66 percent is stainless steel, 48.06 percent is Ti-Zr-Mo alloy, 9.13 percent is Nb-1Zr, 4.75 percent is W, and 6.40 percent is Li. The gamma emission spectrum is calculated for the fuel capsules with ORIGEN2 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. Activation of the structural materials above and below the fuel capsules and their associated gamma emission spectrum is not explicitly calculated; however, it is estimated from the ratio of the mass of the structural materials in the six FSP-1R fuel assemblies to the mass of the structural materials in the 24 carbide fuel pin source of the approved SARP.

5.3.1.2 Neutron Source

The original neutron source for the approved SARP 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 UO_2. 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 primary 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. The two sodiumbonded 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 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), 5.2 (Source Specification), and 5.2.2 (Neutron Source) of Revision 0. The fuel source consists of 12 fuel capsules with a 180-day cooling period following irradiation for 669.1 effective full power days at a peak flux of 3.178×10^{15} neutrons/second. The mass of the UN in each fuel capsule is 61.9 g of which 76.72 percent is U-238, 18.42 percent is U-235 and 4.86 percent is N. Neutron emission is calculated for the UN fuel with ORIGEN2 and is apportioned as 91.4 percent from spontaneous fission and 8.6 percent from alpha-neutron reactions. The primary sources of the spontaneous fission and alpha-neutron reaction neutrons are Cm-244 and Pu-238, respectively. The neutron emission spectrum is tabulated in a 15-group structure of mean neutron energies identical to that from the approved SARP. Bounding limits for each of the 15 energy groups are, once again, not provided.

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. 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. No distinction is made in the radial and axial shielding configuration between normal and accident conditions. The 21 sodium-bonded metal pin and 24 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 1.0 (General Information), 1.2.1 (Packaging), 1.2.3 (Contents of Packaging), 5.3 (Model Specification), and 5.4 (Shielding

Evaluation) of Revision 0. The six FSP-1R fuel assemblies will be retained within the confines of the outer six radial compartments of the Ident 69, with the center compartment, a 1.63-inch inside diameter stainless steel tube, remaining vacant. The confines of each Ident 69 outer radial compartment are the 1.75-inch outside diameter stainless steel inner tube, the inside diameter of the 5-inch schedule 5 stainless steel pipe, and six equally spaced stainless steel plates. The Ident 69 is centered within the T-3 using a liner assembly constructed of 6-inch outside diameter by 0.083-inch aluminum pipe. The positions of the bottoms of the two fuel capsules in each fuel assembly are given as 10.5-inches and 31.9-inches from the bottom of the 93.77-inch long assembly. The axial positions of the bottom of the lower fuel capsule and the top of the upper fuel capsule with respect to the outer surfaces of the cover plates for the push rod insertion port and removable plug port are addressed only through the reference to the Ident 69 and its associated liner.

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. No distinction is made in the source and shield region densities 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 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. No distinction is made in the source region densities between normal and accident conditions. The 21 sodium-bonded metal pin and 24 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 5.4 (Shielding Evaluation) of Revision 0. The homogenized density of each fuel capsule is given as 0.5385 g/cm^3. The homogenized density for the structural materials above and below the fuel capsules is not explicitly calculated; however, it is estimated from the ratio of the mass of the structural materials in the six FSP-1R fuel assemblies to the mass of the structural materials in the 24 carbide fuel pin source of the approved SARP.

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. 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 shielding evaluation for the addendum is addressed in Section 5.4 (Shielding Evaluation) of Revision 0. Detailed shielding analyses of the payload configuration for the six FSP-1R fuel assemblies are not provided. Arguments are made that the expected gamma and neutron dose equivalent rates for a single cask from the shipment of six FSP-1R assemblies will be less than those in the approved SARP. No distinction is made between normal and accident conditions. The energy dependent gamma source for the 12 fuel capsules is, by spectral comparison, less than that for the 24 carbide fuel pin source of the approved SARP. The gamma source from the activation of the structural materials above and below the 12 fuel capsules is estimated, on a mass basis, to be less than that associated with the 24 carbide fuel pin configuration as well. Allowing for the aforementioned differences in source spectra and strength, as well as those in density and axial position, the gamma dose equivalent rate for the six FSP-1R fuel assemblies is estimated to be less than 10 percent of the gamma dose equivalent rate for the 24 carbide fuel pin source of the approved SARP. The energy dependent neutron source for the 12 fuel capsules is, by spectral comparison, less than that for the worst case 217-pin FTR driver assembly source of the approved SARP. Allowing for the differences in source spectra and strength, as well as those in density and axial position, the neutron dose equivalent rate for the six FSP-1R fuel assemblies is estimated to be less than 17 percent of the neutron dose equivalent rate for the 217-pin FTR driver assembly source of the approved SARP.

The staff shielding evaluation of the addendum is limited to a review of the data and information provided and a judgement as to the validity of the arguments.

5.4 Findings and Conclusions

Arguments presented are adjudged sufficient to conclude that the expected surface gamma and neutron dose equivalent rates for a single cask from the transport of six FSP-1R fuel assemblies are less than the surface gamma and neutron dose equivalent rates in the approved SARP for the configurations and regional densities of the 24 carbide-pin gamma and 217-pin FTR driver assembly neutron sources.

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, paragraphs 71.47

and 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 supporting 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, it will be possible to transport radioactive material in the package safely.

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 for Fissile Class III included in the T-3 SARP Addendum (Reference 1) is based on the fact that the limited quantity of fissile material in the FSP-1R payload assures an adequate margin of safety even if the contents were deformed in an accident and optimally moderated and reflected. FSP-1R fuel assemblies are packaged inside an Ident 69 pin container. The Ident 69 is a 5.345-in. ID stainless steel pipe (5.563 in. OD) with 6 pie-shaped compartments, each containing one fuel assembly, around an empty central circular compartment. The Ident 69 has a threaded bayonet-type top-locking closure.

The payload consists of 6 individual FSP-1R fuel assemblies packaged in an Ident 69. The quantity of U-235 is not the same for each assembly, but the maximum quantity of U-235 in the payload will not exceed 120 g. The U-235 enrichment is less than 22 wt.%. The 5.345-in. inside diameter of the Ident 69 pin container is less than 79% of the smallest critical cylinder diameter possible for a homogeneous mixture of U(22) metal in water. Six fuel assemblies represent less than 13% of the minimum critical quantity of U(22)N, assuming a homogeneous sphere with optimum moderation and full water reflection.

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 critical mass data cited by the applicants were confirmed by using the 1986 Revision of "Critical Dimensions of Systems Containing U-235, Pu-239, and U-233" by H. C. Paxton and N. L. Pruvost (LA-10860-MS, July 1987). Tables 9 and 10 give critical data for homogeneous hydrogen-moderated units of uranium enriched in U-235 to various degrees. Critical masses and critical volumes of hydrogen-moderated spheres of U(93), U(30.3), U(4.89), U(3.00), and U(2.00) are displayed in Figures 14 and 15.

We used the 30.3 wt. % data as conservative estimates for the 22 wt. % material in the payload. The minimum critical mass, optimally moderated and reflected, is 1.00 kg U-235, which is more than 8 times greater than the 120 g U-235 in the FSP-1R payload.

6.4 Findings and Conclusions

The critical mass data of "Critical Dimensions of Systems Containing U-235, Pu-239, and U-233" presented in the previous section, indicate that the payload consisting of 6 individual FSP-1R fuel assemblies will remain subcritical under all conditions of reflection and moderation even if were double-batched or set next to a similar package.

On the basis of the comparison between the critical masses for 30.3 wt. % U-235 and the inventories of the fissile material in the proposed FSP-1R payload, the staff concludes that the T-3 cask is designed to maintain its contents in a subcritical state when containing the proposed FSP-1R 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 FSP-1R payload defined in the T-3 SARP Addendum (Ref. 1).

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 programs 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 programs is required.

This section of the T-3 SARP Addendum has been reviewed and the staff concludes that it contains acceptance tests and maintenance programs 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, Westinghouse Hanford Co., Richland, WA 99352, March 23, 1992.

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

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.