Department of Energy
Germantown, MD 20874-1290

SAFETY EVALUATION REPORT

for the

9975 Package

Docket 00-26-9975

Approved:

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Office of Safety, Health and Security, EM-5

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TABLES

ACRONYMS

ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PVC	(ASME) Boiler and Pressure Vessel Code
CoC	Certificate of Compliance
DOE	U.S. Department of Energy
HAC	Hypothetical Accident Conditions
IAEA	International Atomic Energy Agency
ID	Inner diameter
LDPE	Low-density Polyethylene
MNOP	Maximum Normal Operating Pressure
NCT	Normal Conditions of Transport
OD	Outer diameter
PCV	Primary Containment Vessel
QA	Quality Assurance
SAR	Safety Analysis Report
SARP	Safety Analysis Report for Packagings
SCV	Secondary Containment Vessel
SER	Safety Evaluation Report
SGT	Safe-Guard Trailer
SNM	Special Nuclear Materials
SST	Safe Secure Trailer
TI	Transport index
TID	Tamper-indicating device
TSD	DOE/AL Transportation Safeguards Division
WSRC	Westinghouse Savannah River Company

PREFACE

This Safety Evaluation Report (SER) summarizes the review findings of the Safety Analysis Report for Packaging (SARP) for the 9975 Package used for transporting plutonium metal and plutonium oxides.

The SER for this package was prepared for a new application. The review presented in this SER was performed using the methods outlined in the Packaging Review Guide for Reviewing Safety Analysis Reports for Packagings.

The 9975 Package is a 35-gallon drum package design that has evolved from a family of packages designed by Department of Energy (DOE) contractors at the Savannah River Site. The 9975 Package design includes two stainless steel pressure vessel containment systems designed and fabricated in accordance with Section III of the ASME Boiler & Pressure Vessel Code. The two pressure vessels in the 9975 design meet the double containment requirement for plutonium shipments. The 9975 Package design also includes a lead shield to lower the package surface dose rate. Other related package designs include the 9972, 9973, and 9974. Each of these package designs is based on a stainless steel outer drum and has the following specifications:

	9972	9973	9974	**9975**
Drum package outer boundary	30-gal	30-gal	55-gal	35-gal
One pressure vessel containment system	X			
Two pressure vessel containment systems		X	X	X
Lead shield			X	X

The SARP submitted by the applicant addresses all four package designs (9972, 9973, 9974 and 9975). The review presented in this document addresses only the 9975 design. The other package designs (9972-9974) have features different from the 9975 and were not reviewed.

Earlier package designs, the 9965, 9966, 9967 and 9968 were originally designed and certified in the 1970s. (Transportation regulations do not allow new packages from the 9965-9968 series to be built.) In the 1990s, updated package designs that incorporated design features consistent with new safety requirements were proposed. The updated packages are the 9972, 9973, 9974 and 9975.

The 9975 SARP includes several content descriptions. The review documented in this SER addresses two content types: plutonium metal and plutonium oxides. Other content types will be addressed in subsequent SERs.

1. GENERAL INFORMATION REVIEW

1.1 Areas of Review

The description and engineering drawings in Chapter 1, General Information Review of the Safety Analysis Report —Packages (SARP) for the 9975 Package were reviewed. The review also addresses plutonium metal and impure oxide contents as described in Tables 1.14 and 1.15 of the Safety Analysis Report—Packages (SARP) for the 9975 Package (WSRC-SA-7 Rev. 12). The General Information review included:

1.1.1 Introduction

- Purpose of Application
- Summary Information

1.1.2 Package Description

- Packaging
- Contents

1.1.3 Compliance with 10 CFR 71

- Statement of Compliance
- Summary of Evaluation

1.1.4 Appendix

- Drawings
- Other Information

1.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the General Information review of the 9975 Package include:

- An application for package approval must be submitted in accordance with Subpart D of 10 CFR 71. [§71.0(d)]
- An application for modification of a previously approved package is subject to the provisions of §71.13 and §71.31(b). All changes in the conditions of package approval must be approved. [§71.13, §71.31(b), §71.107(c)]
- An application for renewal of a previously approved package must be submitted no later than 30 days prior to the expiration date of the approval to assure continued use. [§71.38]
- The maximum activity of radionuclides in a Type A package must not exceed the limits of 10 CFR 71, Table A-1. For a mixture of radionuclides, the provisions of Appendix A, paragraph IV apply, except that for krypton-85, an effective A_2 equal to 10 A_2 may be used. [Appendix A, §71.51(b)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]

- The application must reference or describe the quality assurance program applicable to the package. [§71.31(a)(3), §71.37]
- A fissile material package must be assigned a transport index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59, §71.35(b)]
- A package with a transport index greater than 10 or an accessible external surface temperature greater than 50°C (122°F) must be transported by exclusive-use shipment. [§71.47(a), §71.47(b), §71.59(c), §71.43(g)]
- The application must include a description of the packaging design in sufficient detail to provide an adequate basis for its evaluation. [§71.31(a)(1), §71.33(a)]
- A package for the shipment of plutonium must satisfy the special containment requirements of §71.63(b).
- The smallest overall dimension of the package must not be less than 10 cm (4 inches). [§71.43(a)]
- The outside of the package must incorporate a feature that, while intact, demonstrates evidence that the package has not been opened by unauthorized persons. [§71.43(b)]
- The application must include a description of the contents in sufficient detail to provide an adequate basis for evaluation of the packaging design. [§71.31(a)(1), §71.33(b)]
- Plutonium in excess of 0.74 TBq (20 Ci) must be shipped as a solid. [§71.63(a)]
- The package must be conspicuously and durably marked with its model number, gross weight, and package identification number. [§71.859(c), §71.13]

1.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the General Information chapter of the SARP for the 9975 Package.

1.3.1 Introduction

1.3.1.1 Purpose of Application

The 9975 Package was docketed as a new package. A previously submitted application for this package was reviewed and a Certificate of Compliance (CoC) was issued. Tests of the package indicated possible design deficiencies. The CoC for the package was withdrawn pending package design revision.

The purpose of the application is to document that the 9975 Package satisfies the regulatory requirements of 10 CFR 71, and International Atomic Energy Agency (IAEA) Safety Series No. 6.

The application is complete and contains all information identified in 10 CFR 71 Subpart D.

1.3.1.2 Summary Information

The 9975 Package is designed to transport fissile actinide metals and oxides in excess of Type A quantities. The package is designed for an internal pressure of 63 Mpa (900 psi). The package type and model number, 9975 B(M)F-85 is provided on Drawing R-R2-F-0025, Rev. 0. To comply with DOE Order 5610.14, the package must be shipped by exclusive use, using a Safe Secure Trailer (SST) when the contents contain more than 2 kg of plutonium. A commercial carrier may ship all other material movements. The SARP does not demonstrate that the package meets the requirements for shipment of plutonium by air. Section 2.1.2 states that the package contents are Normal Form, Category I.

Section 1.2.1.4 of the SARP includes a summary of the design criteria for the package. The American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Division 1, Section III, Subsection NB, 1992 Edition will be used to determine package containment system design, fabrication, and inspection requirements. Stainless steel drum bodies are fabricated in accordance with 49 CFR 178, Subpart L. Cane fiberboard insulating and impact-absorbing material must meet ASTM Specification C-208-95. Additional discussion of applicable codes and standards is included in Chapters 2–9 of the SER.

The applicant's QA program is identified in Chapter 9 of the SARP.

The Package has a nuclear criticality control Transportation Index (TI) of 2.0. The TI, based on package surface dose rate, will be determined from measurements made at the time of shipment. Procedures discussed in Section 7 of the SARP limit the maximum surface dose rate on the surface of a package to <200 mrem/hour for all shipments.

1.3.2 Package Description

1.3.2.1 Packaging

The 9975 Package assembly is depicted schematically in Figure 1.1. The packaging outer container is a 35-gallon removable-head drum designed and fabricated in accordance with 49 CFR 178 Subpart L. The drum and its lid are fabricated of 18-gauge (0.048 inches) Type 304L stainless steel. Four ½-inch diameter vent holes are drilled into the drum approximately 90° apart, 1 inch below the drum flange and are covered with a plastic Caplug (fusible plug).

The drum lid is bolted to a 1-¼-inch-wide × 1/8-inch-thick angle flange welded to the top of the drum body using 24, ½-inch high-strength bolts. The lid is recessed 0.55 inches. A 1/8-inch-thick × 1-¼-inch-wide circular ring is welded to the outer section of the lid. The ring serves to reinforce the lid and prevents it from shearing away from the bolts during a Hypothetical Accident Condition (HAC) event. Four ½-inch pins, asymmetrically positioned on the lid bolt circle, function as alignment keys— restricting lid installation to a single orientation. A 1/8-inch diameter hole drilled in the pins is used to install a tamper-indicating device (TID).

<u>Insulation.</u> The material that surrounds the containment vessels is regular-grade wall sheathing cane fiberboard, manufactured per ASTM Specification C-208-95. The cane fiberboard insulation consists of ½-inch-thick sheets bonded together into top and bottom subassemblies with water-based carpenter's glue. The radial thickness of the insulation is 4-¾inches. In the axial direction, the top thickness of cane fiberboard is 3.7 inches and the bottom thickness is 3.4 inches. A stainless steel air shield is placed over and glued to the top fiberboard subassembly. This thin-walled shield inhibits smoldering of the top fiberboard layers when exposed to air in a fire. A length of sash chain welded to the top of the air shield serves as a handle for removing the top subassembly.

A filler pad consisting of a ceramic fiber blanket (Kaowool) encapsulated in stainless steel foil is required between the top insulation subassembly and the drum lid.

<u>Shielding</u>. Radiation shielding is provided by a lead cylinder assembly that surrounds the primary Containment Vessel (PCV)/Secondary Containment Vessel (SCV) double-containment assembly. The shielding assembly consists of an approximately 7-½-inch ID × 20-gauge 304L stainless steel cylinder with a 20-gauge bottom, surrounded by 0.47 to 0.51 inches of lead. An aluminum lid, ½-inch thick, completes the assembly. The lid has four equally spaced bolt holes near the edge for attachment to the cylinder body (¼–20 UNC threaded steel inserts).

<u>Bearing Plates</u>. Two ½-inch thick aluminum bearing plates provide load bearing surfaces against the cane fiberboard insulation.

Primary Containment Vessel (PCV). The PCV consists of a stainless steel pressure vessel designed in accordance with Section III, Subsection NB of the ASME B&PVC, 1992 edition, with a design condition of 900 psig at 300°F. The PCV is fabricated from 5-inch Schedule 40, seamless, Type 304L stainless steel pipe (0.258-inch nominal wall) and has a standard Schedule 40, Type 304L stainless steel pipe cap (0.258-inch nominal wall) at the blind end.

Both vessel body joints are circumferential full-penetration butt welds examined by radiographic and liquid penetrant methods. These welds satisfy ASME B&PVC Section III, Subsection NB requirements.

The PCV closure consists of a male-female cone joint with surfaces that have been machined to identical angles so that they mate with zero clearance. Two grooves for O-rings have been machined into the face of the Type 304L stainless steel male cone. A leak test port is provided between the two O-ring grooves. Two Viton™ GLT fluoroelastomer O-rings (greased with high vacuum silicone grease) are placed in the grooves to form a leaktight seal (less than 10^{-7} ref·cm^3/sec). Zero clearance behind the two O-rings prevents extrusion and loss of sealing ability at design pressures and temperatures. The seal nut, which forces the male cone against the female cone, is threaded into the containment vessel body. Dissimilar materials were selected for the seal nut (Nitronic 60) and the containment vessel body (Type 304L stainless steel) to minimize galling. For oxide contents, the PCV is backfilled with at least 75% carbon dioxide gas prior to closing.

Secondary Containment Vessel (SCV) . The SCV consists of a stainless steel pressure vessel designed in accordance with Section III, Subsection NB of the ASME B&PVC, 1992 edition, with a design condition of 800 psig at 300°F. The SCV is fabricated from 6-inch, Schedule 40, seamless, Type 304L stainless steel pipe (0.280-inch nominal wall) and has a standard Schedule 40, Type 304L stainless steel pipe cap (0.280-inch nominal wall) at the blind end. Both vessel body joints are circumferential full-penetration butt welds examined by radiographic and liquid penetrant methods. These welds satisfy ASME B&PVC Section III, Subsection NB requirements. The SCV closure is identical to that used on the PCV except that SCV is 1 inch larger in diameter.

PCV Bottom Spacer. The PCV bottom spacer is made of aluminum honeycomb and is contoured to fit the curved bottom of the PCV cavity. The spacer is flat which provides a level surface to support the content assemblies in the PCV. The spacer is fabricated from .003-inch-thick (minimum) foil and is rated for an axial compressive strength before deformation of 1500 ± 500 psi.

SCV Impact Absorbers. Aluminum honeycomb impact absorbers are used in the SCV to reduce the impact loads transmitted between the containment vessels. The SCV bottom impact absorber is contoured to fit the curved bottom of the SCV cavity providing a level surface for the PCV to stand on. The SCV top impact absorber is shaped like a thick ring and separates the top of the PCV cone seal nut from the underside of the SCV cone seal. The impact absorbers are fabricated from 0.003-inch-thick (minimum) foil and are rated for an axial compressive strength before deformation of 1500 ± 500 psi.

PCV Sleeve. The PCV is fitted with an aluminum sleeve to fill the space between the contents and the inner wall of the PCV. The PCV sleeve is fabricated from 6061-T6 seamless aluminum tubing. The sleeve is 14.90 inches tall with a 5.00-inch OD. With the PCV sleeve in place, the maximum gap that may be formed, considering tolerances and off-center effects, is 3.0 mm between the outer sleeve wall and the inner wall of the PCV.

3013 Top Spacer. The 3013 top spacer is fabricated from 6061-T6 aluminum tubing and is 5.06 inches tall with a 4.92-inch OD. It is placed on top of the 3013 container to take up the remaining axial space in the PCV cavity. With the 3013 top spacer in place, the maximum gap that may be formed, considering tolerances and off-center effects, is 5.0 mm between the spacer and the inner wall of the PCV. This gap is identical to the gap between the 3013 outer container OD and the PCV ID.

1.3.2.2 Contents

Type B quantities of radioactive material including fissile materials may be shipped in the 9975 Packages. The double containment 9975 Package may be used to ship plutonium metal or compounds in amounts exceeding 20 curies.

Only those plutonium contents explicitly listed in the Content Table 1.14 and 1.15 of the SARP may be shipped. The contents listed in Table 1.14 are further restricted based on information submitted by the applicant by letter dated May 14, 2001. These additional restrictions are included in the conditions of approval in this section of the SER. Shipping requirements and loading restrictions for the contents are listed in Section 1.2.3.1 of the SARP. Additionally, content restrictions/limits are listed as footnotes following the content table. In all cases, the content configuration requirements listed in Section 1.2.3.1 and the specific restrictions/limits listed with each of the content tables shall be met for shipping.

1.3.3 Compliance with 10 CFR 71

1.3.3.1 Statement of Compliance

Section 1.1 of the SARP states that the 9975 Package satisfies the regulatory requirements of 10 CFR 71 and IAEA Safety Series No. 6.

1.3.3.2 Summary of Evaluation

Section 1 of the SARP does not contain a summary of the package evaluations. However, the required summary information is available in other SARP sections as identified below.

Sections 2.6 and 2.7 of the SARP address package structural performance under normal conditions of transport (NCT) and HAC identified in 10 CFR 71.71 and 10 CFR 71.73.

Sections 3.4 and 3.5 of the SARP summarize package thermal performance compliance under NCT and HAC identified in 10 CFR 71.71 and 10 CFR 71.73.

General requirements for all packages identified in 10 CFR 71.43 are addressed in Section 2.4.

Structural requirements for lifting and tie-down devices identified in §71.45 and §71.61 are addressed in Section 2.5.

Section 5.2 of the SARP addresses external radiation requirements identified in §71.47.

Sections 4.2 and 4.3 of the SARP address requirements for Type B packages and special requirements for plutonium packages identified in §71.51 and §71.63, respectively.

Section 6.2 of the SARP addresses the criticality requirements of §71.55 and §71.59. The package is designed to ship fissile material; therefore, the requirements of §71.53 do not apply.

The Section 7.0 Preface, Sections 7.1 and 7.2 address the operating controls and procedures of Subpart G.

Section 9.1.1 summarizes compliance with the requirements of Subpart H.

1.3.4 Appendix

Drawings of the 9975 are provided in the SARP as follows:

R-R2-F-0026 Rev. 0 9975 Shipping Package Drum with Flange Closure Assembly

R-R2-F-0019	Rev. 4	Insulation Subassemblies
R-R2-F-0020	Rev. 4	Shielding
R-R2-F-0025	Rev. 0	Drum with Flange Closure Subassembly and Details
R-R2-F-0018	Rev. 3	Containment Vessel Subassemblies
R-R3-F-0016	Rev. 6	Containment Vessel Weldments
R-R3-F-0015	Rev. 4	Air Shield Weldment
R-R4-F-0054	Rev. 4	Containment Vessel Details
R-R4-F-0055	Rev. 3	9975 Shipping Package PCV Sleeve and 3013 Spacer

1.4 Evaluation Findings

1.4.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the design of the 9975 Package has been adequately described to meet the requirements of 10 CFR 71. This description also demonstrates that the 9975 meets the minimum size limitations and contains an anti-tampering device required by the regulation. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

1.4.2 Conditions of Approval

In addition to a summary package description and specifications of authorized contents, the following other conditions of approval are applicable to the General Information review of the 9975 Package.

- Contents are restricted to plutonium as identified in Table 1.14 and 1.15 of the SARP.
- Minimum transport index (based on criticality safety) is 2.0.
- The plutonium metal and oxide shall be treated in accordance with the requirements in DOE-STD-3013.
- Any individual piece of the metal contents shall have a minimum mass of 50 grams and have a minimum dimension of 0.1 cm.
- Maximum isotope weight percentages are given in Tables 1.14 and 1.15 of the SARP. Impurities must be alloyed with, or physically/chemically incorporated within, the plutonium and/or uranium metal or oxide structure and not be capable of separation by mechanical, chemical, or thermal means during transport or storage. Obvious, readily removable tramp materials such as metal fasteners and other debris shall be removed from the material prior to packaging. For oxide contents, the moisture contents shall be less that 0.5 percent of the total content mass.
- For metal contents, gallium as an alloying constituent is permitted up to a nominal 1 weight percent of total plutonium mass.
- For metal contents, beryllium and carbon impurities are limited to a maximum of 100 grams each.
- For metal contents, carbon impurities may be present only as graphite inclusions smaller than 25 microns (0.001 inches) in diameter.

- For metal contents, total impurities, including beryllium and carbon, are limited to 200 grams or 5 weight percent of the total plutonium mass, whichever is less. The limit on impurities does not include the gallium alloying constituent.
- For oxide contents, the plutonium plus uranium mass may not be less than 30 weight percent of the total content mass.
- Authorized metal contents shall have been produced by metal casting, molten salt extraction, electro-refining, direct oxide reduction, or plutonium fluorides reduction, such that contaminants are evenly distributed throughout the metal.
- Plutonium metal may not be porous, as evidenced by process knowledge or measurement.
- All plutonium metal surfaces shall be easily observable and inspected and brushed prior to packaging in the convenience can.
- In addition to the isotopes listed in Tables 1.14 and 1.15 of the SARP, small concentrations (<1000 ppm) of other actinides, fission products, decay products, and neutron activation products are permitted, subject to the limit on total impurities identified above.
- Aluminum foil packing between the food pack cans and the PCV is limited to 200 grams (foil packing is not permitted in 3013 can configurations). Plastic in the PCV is restricted to Low-Density Polyethylene (LDPE) or nylon bagging and is limited to 100 grams (plastic limit applies to food pack can configurations only). Small quantities of PVC tape, sufficient to seal slip-lid cans, are permitted. The 3013 top spacer is required for 3013 can shipments.
- Crimp seal food pack cans may be used for metal contents, but may not be used for oxide contents. Slip lid or screw top metal convenience cans may be used with metal or oxide contents. Product or food pack cans with organic liners may not be used for any contents.
- All product, food pack, or 3013 cans must be examined for post-sealing bulging or buckling prior to placement inside the PCV. No can that is visibly bulged or buckled may be transported in the package.
- Product, food pack, or 3013 cans shall be inspected upon removal from the PCV after shipment. Any visible bulging, buckling, or evidence of corrosion shall be reported immediately to the DOE Headquarters Certifying Official.
- For oxide contents, the 3013 outer, inner, and convenience containers must be backfilled with an inert gas such that the oxygen content is no more than 5 volume percent in each container upon welded closure of the outer container.
- For oxide contents, the oxygen content in any space within the 3013 can may not exceed 5 volume percent within a period of one year from the time the contents are loaded in the package.
- For oxide contents, the oxygen content in any space within the PCV may not exceed 5.25 volume percent within a period of one year from the time the contents are loaded in the package.

Drawings that define the package design include:

R-R2-F-0026	Rev. 0	9975 Shipping Package Drum with Flange Closure Assembly
R-R2-F-0019	Rev. 4	Insulation Subassemblies
R-R2-F-0020	Rev. 4	Shielding
R-R2-F-0025	Rev. 0	Drum with Flange Closure Subassembly and Details
R-R2-F-0018	Rev. 3	Containment Vessel Subassemblies
R-R3-F-0016	Rev. 6	Containment Vessel Weldments
R-R3-F-0015	Rev. 4	Air Shield Weldment
R-R4-F-0054	Rev. 4	Containment Vessel Details
R-R4-F-0055	Rev. 3	9975 Shipping Package PCV Sleeve and 3013 Spacer.

1.5 References

American Society for Testing and Materials, "Standard Specification for Cellulosic Fiber Insulating Board," ASTM C208, Philadelphia, PA, 1995.

American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, 1998.

International Atomic Energy Agency (IAEA), Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Compatibility with the International Atomic Energy Agency (IAEA)," 60 FR 50248, September 28, 1995, as amended.

Title 49, Code of Federal Regulations, Part 178 (49 CFR 178), "Specifications for Packagings," 55 FR 52716, December 21, 1990, as amended.

U.S. Department of Energy, Albuquerque Operations Office, Supplemental Directive, Transportation Safeguards System Program Operations, DOE AL SD 5610.14, December 15, 1994.

Westinghouse Savannah River Company, Interoffice Memorandum, SR Responses to EM-5 questions on the 9972-9975 SARP(U), SRT-RMPT-2001-00015, May 14, 2001.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

2. STRUCTURAL REVIEW

Chapter 2, Structural Review, of the Safety Analysis Report—Packages (SARP) for the 9975 Package was reviewed to address the structural performance of the package design for the tests specified under NCT and HAC. The review also compares the package design requirements to the structural requirements of 10 CFR 71.

2.1 Areas of Review

The structural design of the package was reviewed. The structural review included the following:

2.1.1 Description of Structural Design

- Design Features
- Codes and Standards

2.1.2 Materials of Construction

- Material Specifications and Properties
- Prevention of Chemical, Galvanic, or Other Reactions
- Effects of Radiation on Materials

2.1.3 Fabrication, Assembly, and Examination

- Fabrication and Assembly
- Examination

2.1.4 General Considerations for Structural Evaluations

- Evaluation by Test
- Evaluation by Analysis

2.1.5 Structural Evaluation for Normal Conditions of Transport

- Heat
- Cold
- Reduced External Pressure
- Increased External Pressure
- Vibration
- Water Spray
- Free Drop
- Corner Drop
- Compression
- Penetration

- Structural Requirements for Fissile Material Packages

2.1.6 Structural Evaluation for Hypothetical Accident Conditions

- Free Drop
- Crush
- Puncture
- Thermal
- Immersion–fissile material
- Immersion–all packages

2.1.7 Structural Evaluation of Lifting and Tie-Down Devices

- Lifting Devices
- Tie-Down Devices

2.1.8 Structural Evaluation for Special Pressure Conditions

- Analysis of Pressure Test

2.1.9 Appendix

2.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Structural review of the 9975 Package include the following.

- The package must be described and evaluated to demonstrate that it meets the structural requirements of 10 CFR 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The package must be made of materials of construction that assure there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water among the packaging components, among package contents, or between the packaging components and the package. The effects of radiation on the materials of construction must be considered. [§71.43(d)]
- The performance of the package must be evaluated under the tests specified in §71.71 for NCT. [§71.41(a)]
- The package must be designed, constructed, and prepared for shipment so there will be no loss or dispersal of contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging under the tests specified in §71.71 for NCT. [§71.43(f), §71.51(a)(1)]
- A package for fissile material must be so designed and constructed and its contents so limited to meet the structural requirements of §71.55(d)(2) through §71.55(d)(4) under the tests specified in §71.71 for NCT.

- The performance of the package must be evaluated under the tests specified in §71.73 for HAC. [§71.41(a)]
- The package design must meet the lifting and tie-down requirements of §71.45.
- The package design must have adequate structural integrity to meet the internal pressure test requirement specified in §71.85(b).

2.3 Review Procedures

The Structural review ensures that the package design has been adequately described and evaluated under the NCT and the HAC to demonstrate sufficient structural integrity to meet the requirements of 10 CFR 71.

The structural review is based in part on the descriptions and evaluations presented in the General Information and the Thermal Evaluation sections of the application. Similarly, results of the structural review are considered in the review of all other sections of the application.

2.3.1 Description of Structural Design

Table 2.1 of the SARP identifies the following critical structural components:

- Stainless steel drum
- Cane fiberboard impact absorbing and insulating material
- Stainless steel air shield
- Secondary containment vessel (stainless steel)
- Primary containment vessel (stainless steel)
- Containment closure nut and seals
- Aluminum honeycomb impact absorbers

Features of each of these components are summarized below.

2.3.1.1 Design Features

The 9975 Package has been designed to provide a containment system that can withstand loading resulting from NCT, as well as those associated with HAC.

Specifically, the 9975 Package is designed to:

- Withstand loads resulting from handling, transportation, and accidents
- Provide double containment under NCT that is leaktight to less than 10^{-7} std cm^3/s air as measured in accordance with ANSI N14.5.
- Provide double containment under HAC. Each containment vessel will remain leaktight after an accident by demonstrating a post-accident leak rate of less than 10^{-7} std cm^3/s air.
- Include a leaktest port on the closure for post-load leakage tests.
- Protect the containment vessels from heat in a hypothetical fire.
- Provide cushioning to prevent mechanical damage to the containment in the event of impact.
- Accept a 4.92-inch-diameter Pu storage container.

Table 2.3.1 provides a data summary of the important components of the 9975 Package.

Table 2.3.1 Data Summary of Components of the 9975 Package

Item	**9975**
Drum Material	SA-240
Drum size, gallons	35
PCV	Yes
SCV	Yes
Lead shield	Yes
Air shield	Yes
Fiberboard insulation	Yes

The 9975 Package has no lifting or tie-down components, but does have a lead shield whose structural integrity depends on the structural performance of the Celotex, aluminum honeycomb, and bearing plates. The lead is only required to reduce radiation dose under NCT. The package can meet the higher dose limit allowed following the HAC without lead shielding.

2.3.1.2 Design Criteria

The criteria for the design of the 9975 Package are in accordance with 10 CFR 71; 49 CFR 173 through 178; IAEA Safety Series No. 6, 1973 Revised Edition; and the intent of Section III, Subsection NB of the ASME B&PV Code, 1992 edition as detailed in Section 9.3. The structural analysis was performed in accordance with the methodology and stress criteria specified in ASME Code Section III, Division I, Subsection NB, and the Regulatory Guides 7.6 and 7.8.

The design criteria from NCT and HAC loadings are from packaging requirements based on content activity levels defined in Figure 2-2 of the Packaging Review Guide. For the packages considered, activity levels of the enriched uranium product exceed 3000 A_2 and, therefore, the packages are classified as Category I. The design criteria of critical components are listed in Table 2.3.2.

Table 2.3.2 Critical Component Design Criteria

Component	Design Criteria
Drums	49 CFR 178, Subpart L
Insulating and impact absorbing material	Cane fiberboard per ASTM Specification C208; Density: Nominally 15 pounds per cubic foot
Primary containment vessel	Section III, Subsection NB of the ASME B&PV Code, 1992 edition, 900 psig at 300°F
Secondary containment vessel	Section III, Subsection NB of the ASME B&PV Code 1992 edition, 800 psig at 300°F
Containment vessel seals	Viton GLT per Parker Compound No. V835-75, greased with high vacuum silicone grease. Static seal for continuous service of temperatures of -40°F to 400°F. Higher temperatures are possible for non-continuous service (Section 2.7.3).
Aluminum impact absorbers	Aluminum honeycomb tube, 0.003-inch minimum foil, pre-crushed, crush strength of 1500 +/- 500 psi

2.3.1.3 Drums

The 9975 drum is constructed from stainless steel with a flange closure, manufactured as shown on the drawing R-R2-F_0025 in Appendix 1.1. The drum design and fabrication satisfy the intent of the requirements of Section III, Subsection NF, of the ASME B&PV Code, 1995 edition . The drum and its lid are fabricated of 18-gauge (0.048 inches) Type 304L stainless steel. Four ½-inch diameter vent holes are drilled into the drum and plugged with a plastic BP Caplug. The plugging device prevents water or moisture from entering the drum through the vent holes under NCT. In the event of a fire, the plugs melt, allowing the drum to vent gases generated from the insulation to prevent rupture of the drum. The drum lid is bolted using 24 ½-inch high-strength bolts to a 1¼-inch-wide × 1/8-inch-thick angle welded to the top of the drum body. The lid is recessed 0.55 inches. A 1/8-inch thick × 1-¼-inch wide circular ring is welded to the outer section of the lid. The ring serves to reinforce the lid and prevents the lid from shearing away any bolts during an HAC event. Nuts are tack welded to the flange underside to ease assembly operations. The bolts are tightened to 30+/-2 ft-lbs of torque. (Note: no specific tightening sequence is required.) Bolts are then re-tightened to ensure none were missed on the first pass. Four ½-inch pins, asymmetrically positioned on the lid bolt circle, function as an alignment key, restricting lid installation to a single orientation. The pins are drilled with a 5/16-inch diameter hole for installation of a tamper indicating device (TID) while the drums are in storage. A 1/8-inch diameter hole is drilled through the shank of each bolt for insertion of a TID during shipping operations. The drum chime includes a non-structural skip weld that serves as a TID to meet the IAEA requirement of demonstrating that the package has not been tampered with during use.

Each package is identified by a stainless steel data plate mounted on the outside of the drum. The plate labeling and mounting requirements are shown on drawings in Appendix 1.1. The 9975 Package is also affixed with a bar coded steel data plate. The drums will have no paint or other markings.

2.3.1.4 Cane Fiberboard

Each drum package is lined with cane fiberboard that complies with ASTM Specification C208 and has a nominal density of 15 lb/ft^3. The cane fiberboard protects the containment vessels during NCT and HAC by providing both impact protection and thermal insulation.

Cane fiberboard discs for the 9975 Package are held together by glue. Cutouts (see fiberboard assembly drawings in Appendix 1.1) are provided in the fiberboard discs at the top and bottom of the SCV to prevent or minimize tearing of fiberboard discs during HAC drops. Cutouts also help in providing softer impacts, which result in lower impact *g* values.

2.3.1.5 Containment vessels

The containment vessels are sealed with dual concentric elastomer O-rings (Parker O-ring compound V-835-75 or equivalent). The containment boundary is comprised of the outermost O-ring and the containment vessel body. An evacuation port is located between the O-rings to facilitate post-load leakage testing. A package assembly verification air leak rate of 10^{-3} std cm^3/s must be demonstrated before the package is released for transport (refer to Chapter 4 of the SARP). This air leak rate assures effective O-ring sealing. After the leak test, the evacuation port is sealed with an approved pressure plug and gland nut and then leak tested.

10 CFR 71.73(c) requires that the containment system be immersed in water such that the external pressure is equivalent to at least a 50-foot head of water, which equates to an external pressure of 21 psig. Immersion tests are described in Section 2.7.4 of the SARP.

To verify the capability of the system to maintain structural integrity, 10 CFR 71.85(b) requires that the containment vessels be tested at an internal pressure at least 50% higher than the Maximum Normal Operating Pressure (MNOP) when MNOP exceeds 5 psig. Pressure tests to meet this requirement are described in Section 2.6.1.1 of the SARP.

The containment vessels of the 9975 Package are fabricated in accordance with ASME B&PV Section III. The design analysis is performed in accordance with ASME Section III, Subsection NB as explained in the opening paragraph of Section 2.3.1.2. The 9975 Package has double containment vessels. The inside diameter of the PCV is sized to accept a 4.92-inch-diameter Pu storage container.

2.3.1.6 Air Shield

A 24-gauge (0.0239-inch) thick stainless steel air shield is provided at the top of the 9975 Package to prevent air from coming into contact with fiberboard above the containment vessel during a fire accident and thus prevent higher temperatures near the closure seal of the containment vessels. The air shield design incorporates a gap of approximately 1/8-inch all around the shield so that combustible gases can flow around and escape through the vents. From a structural standpoint, the shield is thin enough that it does not affect the energy-absorbing capacity of the fiberboard.

2.3.1.7 Weights and Centers of Gravity

The nominal component weights and the maximum content weights of the 9975 Package are provided in Table 2.3.3. The weight of the contents of the actual packages will be less. Packaging drop tests were performed at the approximate gross weight provided in Table 2.3.3.

The Center of Gravity of the 9975 Package is located on the longitudinal centerline, approximately 17½ inches from the bottom end.

Table 2.3.3 9975 Package Component Weights

9975 Components	Weights (lbs)
35-gal drum and insulation (overpack)	127.0
Primary containment vessel	33.9
Secondary containment vessel	54.1
Aluminum honeycomb spacers (impact absorbers)	0.68
Lead shielding material	140.0
Aluminum bearing plates	10.0
Packaging net weight	360.0
Contents (maximum)	44.4
Package gross weight	404.0

2.3.1.8 Conclusions

- The Structural review confirmed that the text and sketches describing the structural design features are consistent with the engineering drawings and the models used in the structural evaluation.
- The criteria for design of the 9975 Package are in accordance with 10 CFR 71; 49 CFR 173 through 178; IAEA Safety Series No. 6, 1973 Revised Edition; and Section III, Subsection NB of the ASME B&PVC, 1992 edition.
- Local buckling for the containment vessels is evaluated to the requirements of the ASME Code. Specifically, Code Case N-284 is used to evaluate the containment vessels for buckling.
- To avoid brittle fracture problems, the selection of all material components was based upon the guidance provided by NRC Regulatory Guide 7.11.
- The staff has confirmed that the application identifies the established codes and standards, which are judged by the staff to be appropriate for the intended purpose and are properly applied.

2.3.2 Materials of Construction

2.3.2.1 Material Specifications and Mechanical Properties

The material specifications for the packaging components are provided in Table 2.3.4. These specifications are also provided in Appendix 1.1 of the SARP. The mechanical properties of the packaging materials are presented in Tables 2.5 through 2.9 of the SARP. Design temperature ranges are listed to establish allowable stresses used in containment vessel design calculations and provided in Appendix 2.1 of the SARP. ASME Section III, Subsection NB allowable stresses for the containment vessels are provided in Table 2.10 of the SARP.

Mechanical properties are obtained from the ASME B&PV Code, Section II, Part D, 1992. These properties are given from -20°F and above. However, the lowest design temperature is -40°F per 10 CFR 71 (see Table 2.5 of the SARP). In general, the mechanical properties, such as yield strength and tensile strength, increase with decreasing operating temperature and, therefore, are not a concern. However, fracture toughness decreases as the operating temperature decreases.

It is indicated in NRC Regulatory Guide 7.11 that the austenitic steels are not susceptible to brittle fracture at transport temperatures and, therefore, failure brittle fracture in containment vessels at -40°F is not a concern.

Table 2.3.4 Packaging Material Specifications

Component	**Specifications**
Drum	18-gauge stainless steel, Type 304L, ASME SA-240
Insulation	Industrial cane fiberboard, 14-16 lb/ft^3, ASTM C208
Containment vessels	Type 304L, ASME SA-312, SA-403, and SA-479
Bottom bearing plate	Aluminum, type 1100, ASTM B209
Top bearing plate	Aluminum, type 1100, ASTM B209
Lead shielding	Lead, ASTM B749, Shielding Material
Drum vent plugs	Caplugs© , model BP-½, Protective Closures, Co., Inc.
Hex cap screw	½-13 UNC-2A × 1.25, ASME SA-320, Grade L7, with 0.19-inch hole
Flange nut	Hex nut, ½-13 UNC-2B, ASME SA-194, Grade 8
Angle	1.25 × 1.25 × 0.125 thick angle, 304 or 304L stainless steel, ASME SA-479, Roll to fit OD of drum
Pin	304 or 304L stainless steel, ASME SA-479
Reinforcing ring	20.85 OD × 1.25 wide × 0.125 thick, 304 or 304L stainless steel, ASME SA-479 bar
Washer	½-inch hardened circular washer, ASTM F436
Lid (cone seal) nut	Nitronic-60 Stainless Steel alloy. UNS-S221800 alloy, ASTM A479, Crucible Specialty Metals, Syracuse, NY
Thread grease	KRYTOX® fluorinated grease by E.I. du Pont, 240 AC
O-rings	Viton GLT per Parker Compound No. V-835-75
Spacers and impact absorbers	Aluminum honeycomb tube. Minimum foil thickness is 3 mil, crush strength 1500 +/- 500 psi, pre-crushed.
PCV sleeve	Aluminum tube, type 6061-T6
3013 spacer	Aluminum tube, type 6061-T6

2.3.2.2 Conclusions

- The material properties are appropriate for the load conditions (e.g. static or dynamic impact loading, hot or cold temperatures, and wet or dry conditions). Because the Celotex® insulating material can degenerate over time under wet conditions, drum overpacks are designed to minimize the infiltration of water under NCT.
- The temperatures at which allowable stress limits are defined are consistent with minimum and maximum service temperatures.
- The force-deformation properties for the Celotex® energy absorbing material are based on appropriate test conditions and temperatures.
- The materials of structural components have sufficient fracture toughness to preclude brittle fracture under NCT and HAC.
- The staff has verified that the materials and coatings of the package will not produce significant chemical or galvanic reactions among packaging components, among packaging contents, or between the packaging components and the package contents.
- The possible generation of hydrogen due to radiolysis of the plastic bags has been addressed in Appendix 3.4 of Chapter 3 of the SARP.

2.3.3 Fabrication, Assembly, and Examination

The staff has confirmed that appropriate fabrication specifications are prescribed by codes or standards, and that the code or standard is identified on the engineering drawings, or in the text of the SARP. For the containment vessel components, the fabrication meets the requirements of the ASME B&PV Code, Section III, Subsection NB. For components for which no fabrication code or standard is specified, control of the fabrication will be maintained by implementation of the Quality Assurance Plan through the procedural methodology described in Chapter 9.

The staff has confirmed that the examination methods and acceptance criteria are dictated by the same code or standard used for the fabrication of a component. For components for which no fabrication code or standard is specified, the examination will be controlled by implementation of the Quality Assurance Plan through procedural methodology described in Chapter 9.

2.3.4 General Considerations for Structural Evaluations

Structural evaluations of the package were performed by full-scale testing of prototype packages. The testing program was supplemented by analysis to extrapolate test conditions to other credible HAC and NCT conditions.

2.3.4.1 Evaluation by Test

- The staff considered the description of the surface (e.g., material, mass, dimensions) used for the free drop and confirmed that it represents an essentially unyielding surface as specified in §71.73(c)(1).
  - The staff considered the description of the steel bar (e.g., material, dimensions, orientation, method of mounting) used for the puncture test and confirmed that it is securely attached to an essentially unyielding surface, has sufficient length to cause maximum damage to the package, and meets the other specifications of §71.73(c)(3).

- The staff verified that the test specimen has been fabricated using the same materials, methods, and quality assurance as specified in the design. The staff identified differences between the materials and evaluated the effects in the application. Substitutes for the contents have the same representative weight as the actual contents.

- The staff verified that the selected drop orientations consider the orientations for which maximum damage is expected, and that the selection is justified.

- The staff verified that all test results are evaluated and their implications interpreted, including both interior and exterior damage of the test article. Unexpected or unexplainable test results indicating possible testing problems or non-reproducible specimen behavior have been discussed and evaluated.
  - The staff evaluated the appropriate videos and/or photos of the tests.
  - The staff verified that the margin of safety of the package design has been adequately evaluated.
  - The staff addressed the criteria for evaluating pass/fail for the test conditions. The test results have been compared with these criteria.

2.3.4.2 Evaluation by Analysis

- The staff verified that a clear description of the calculations, and all assumptions, are included.

- The staff verified that the models and material properties are appropriate for the load combinations considered, that the material properties (e.g. elastic, inelastic) are consistent with the analysis methods, that the application justifies the strain rate at which the properties were determined, and that the analysis considers true stress-strain or engineering stress-strain, as applicable.
  - The staff has confirmed that bounding dynamic analyses were performed and that dynamic amplification of component stresses have been adequately addressed.
  - The staff is satisfied that the most unfavorable drop orientations were chosen for the simulated 30-ft drops.
  - The staff has confirmed that the analyses adequately account for varying impact loading transmission to the contents, resulting in variable test conditions.
  - The staff verified that the computer codes, if applicable, are appropriately used and benchmarked.
  - The staff verified that the response of the package to loads, in terms of stress and strain to components and structural members, is shown, and that the structural stability of individual members, as applicable, is evaluated.
  - The staff examined the summary table of the results of the analyses, compared the results with the acceptance criteria provided, and verified that the acceptance criteria have been met and the criteria are in accordance with appropriate codes and standards.

2.3.5 Structural Evaluation for Normal Conditions of Transport

2.3.5.1 Heat

- If exposed to direct radiation at 100°F ambient temperature, the drum outer surface and containment vessel assembly (with the source) will reach maximum temperatures as shown in Table 3.14 of the SARP. These temperatures are consistent with those in the Thermal Evaluation section.

- The review also verifies that any differential thermal expansions and possible geometric interferences have been considered and the stresses are within the limits for normal condition loads.

- Structural adequacy of the containment vessels for prolonged service under high-temperature environments is demonstrated by comparison with the test results from the tests conducted on containment vessels of packages 9965 and 9968. During the test, the specimens were pressurized to 1000 psig and held at a temperature of 600°F for 16 hours. At the conclusion of the test, helium leakage from the containment vessels was not detectable with a helium detector. The test results show that the O-rings meet the leakage criteria with an internal pressure of 1000 psig (which is far greater than the MNOP) and a temperature of 600°F for 16 hours. Interpolation of the test results indicates that the containment will remain leak tight for approximately 1000 hours at the 500°F design temperature, which is found to be acceptable.

2.3.5.2 Cold

- A regulatory cold test required per 10 CFR 71 was performed on the 9965 Package PCV at -40°F. A helium leakage test was conducted on the PCV per NRC Regulatory Guide 7.8, September 1988, and ANSI N 14.5, 1987, using the bell jar method. The PCV remained leak tight to 10^{-7} std cm^3/sec air for a test period time of 10 minutes. The SCV is nearly identical to the PCV in design, with the exception of a slightly larger diameter and length. The cold test results for the PCV are applicable to the SCV as well. The temperatures under the cold test condition are consistent with the Thermal section. The cane fiberboard assembly properties at -40°F lead to load/deflection data that show a significant stress spike during impact loading. However, the duration of the spike is too short to cause any significant stress amplification in the containment vessels. Therefore, containment vessel response to impact loads at -40°F will be similar to the response at room temperature.

- The packages contain no liquids or other materials that could freeze or otherwise be adversely affected by ambient temperatures of -40°F.

- The staff has verified that no component stress allowables are exceeded by normal condition loading.

2.3.5.3 Reduced External Pressure

- Reducing the external pressure to 3.5 psia combined with maximum internal pressurization could cause increased pressure loading on the containment vessel walls. An analysis of the vessels for an internal pressure differential of 150 psi was conducted. This analysis bounds the possible effects of reduced external pressure.

- For the 9975 Package, the SCV experiences the effect of the reduced external pressure. For these vessels, the maximum pressure differential will be 102.2 psi if external pressure of 3.5 psia is assumed. This pressure differential is enveloped by the internal design pressure differential of 150 psi used in the analysis.

- The drums are protected from reduced external pressure transients by the vent holes covered by Caplugs.

- It is determined that the application adequately evaluates the package design for the effects of reduced external pressure equal to 25 kPa (3.5 psi) absolute and that the application considers the greatest possible pressure difference between the inside and outside of the package.

2.3.5.4 Increased External Pressure

- Increased external pressure to 20 psia combined with minimum internal pressurization will not cause localized buckling of the containment vessel walls. A buckling analysis for the vessels, per ASME Code Case N-284, for an external pressure differential of 20 psi is conducted.

- The drums are protected from an increased external pressure transient by the cane fiberboard which is capable of withstanding an additional load of more than 5.3 psi (=20 psi–14.7 psi).

- It is determined that the application adequately evaluates the package design for the effects of increased external pressure equal to 140 kPa (20 psi) absolute. In the evaluation, the application considers this loading condition in combination with minimum internal pressure, the greatest possible pressure difference between the inside and outside of the package as well as the inside and outside of the containment system, and the possibility of buckling.

2.3.5.5 Vibration

- A random vibration analysis based on power spectral density for SST was performed to demonstrate that vibration and shock loadings are small and would not cause any fatigue concerns. Though the analysis neglects the load transmission characteristics of the vehicle's suspension, the application indicates that similar packaging (DOT Spec 6M) has withstood years of transport with no significant damage occurring from normal vibration. The containment vessels for the application are smaller than the largest vessels permitted by DOT Spec 2R. Therefore, the containment vessels are less susceptible to vibration damage than the DOT Spec 2R containment vessel. The cone closure is tightened to a predetermined torque, which results in the closure joint components fitting metal-to-metal. The compressed O-rings and the metal friction of the closure thread lock the joint, preventing loosening from vibration. Since the containment vessel components have the same coefficient of thermal expansion, no thermal loosening of the cone-seal nut will occur. Use of required torque values over several years of successful operation has verified that vibration caused by NCT will not result in loosening the cone-seal nut.

- Therefore, it is determined that the application adequately evaluates the package design for the effects of vibration normally incident to transport. A fatigue analysis was provided for highly stressed systems, considering the combined stresses due to vibration, temperature, and pressure loads, and closure bolt preload.

2.3.5.6 Water Spray

- Water spray would cause no damage to the outer drum. The stainless steel drum of the 9975 Package is not affected by corrosion. The drum closure is weather sealed, along with the four sealed vent holes. The containment vessels, which are fabricated from austenitic stainless steel, are not affected by water. A corrosion study shows that water-induced corrosion is insignificant in this package.

2.3.5.7 Free Drop

- The application indicates that a free drop through a distance of 4 feet onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected, would not reduce the effectiveness of the packaging. This is indicated by the fact that no drum and containment vessel failures are observed as a result of the 30-ft drop tests performed on the 9975 with the 4-ft drop tests performed prior to the 30-ft drops.

2.3.5.8 Corner Drop

- This test is not required for 9975 because the total weight of the package exceeds 220 lb.

2.3.5.9 Compression

- A compression test using a load of 2061 pounds on the top of the 9975 Package for a minimum of 24 hours yielded no effect on the package.

2.3.5.10 Penetration

Penetration testing was performed on a modified 6M package with a drum overpack similar to the 9975 Package.

- The application indicates that a 13-lb. vertical steel rod, 1-¼ inches in diameter was dropped from a height of 4 feet onto the most vulnerable surface of each of several different sizes of drums with the cane fiberboard in place for the modified 6M package. Maximum deflection of the drum surface was ¼ inch. No rupture of the drum or damage to the insulation occurred.

2.3.5.11 Structural Requirements for Fissile Material Packages

The SARP structural analysis demonstrates that the following conditions are met for fissile material packages:

- The form of the contents is not substantially altered. Note that the SARP criticality evaluation assumes the contents are in the most reactive configuration.
- The containment system precludes the in-leakage of water following NCT and HAC tests.
- The total effective packaging on which nuclear criticality safety is assessed is not reduced following NCT tests.
- The total effective spacing between fissile contents and the outer surface of the package is unchanged following NCT tests.
- The outer surface of the package does not have an opening large enough to pass a 10-cm cube following the HAC test.

2.3.6 Structural Evaluation for Hypothetical Accident Conditions

2.3.6.1 Free Drop

- Structural integrity of the packages against 30-ft drops onto a flat, essentially unyielding horizontal surface was demonstrated by prototype testing. The unyielding impact surface is constructed from a 6.25-inch thick armor plate, a specialty very-high-strength steel used in armored vehicles to resist penetration from high-velocity impacts, approximately 5 feet square. The plate is anchored in a 30-inch-thick reinforced concrete slab that is insulated from the existing building concrete floor. The impact target weighs approximately 15,600 lb, which is nearly forty times the weight of the 9975 Package. The plate is level with the surrounding floor in the test facility. This impact surface has been used for a number of years for drop testing of the nuclear packages. There is no visible evidence of bending, cracking, or movement of the impact surface relative to the surrounding floor.
- For the 9975 Package, three drop tests (a 10° slap down, and two shallow (17.5° and 22.5°) side impacts to the closure end) were conducted at ambient normal, i.e., test facility environment, conditions. The acceptance of the package against 30-ft drop impacts is based on these three tests for ambient normal conditions and finite element analysis for high/low temperature desiccated, normal, and moist (saturated) environmental conditions.
- Earlier testing of prototype packages provided information on pressure vessel and aluminum honeycomb response to HAC tests.
- Extensive dynamic impact tests were performed on the Celotex impact cushioning/insulating material incorporated in the 9975 Package. For use in the 9975 Package, ½-inch Celotex sheets are cut to form by abrasive water-jets and bonded together by wood glue. The test samples, cut from the glued assemblies used in the package, were pre-conditioned to represent the high/low temperature desiccated, normal, and moist (saturated) environmental conditions that were not evaluated by physical drop testing of the 9975 Package. The results of this testing effort were used to benchmark and validate the material models used in the FEA simulated 30-ft drops at high/low temperature desiccated, normal, and moist (saturated) environmental conditions.
- The FEA simulations found that impact loading of the containment vessels during a 30-ft side drop is sensitive to the widths of the glue layers in the bonded Celotex assemblies at high/low temperature desiccated and low temperature moist (saturated) environmental conditions. However, further finite element analysis on a package modified by excluding the outer drum and Celotex was performed. A simulated 55-ft drop was performed and the results show that no buckling occurred and there was no extensive plastic deformation in the closure region. Therefore, based on actual physical testing and finite element analysis, it is demonstrated that the 9975 Package can withstand a 30-ft drop under all environmental conditions required by 10 CFR 71 and maintain acceptable structural integrity with adequate margin.

2.3.6.2 Crush

- A crush test is not applicable to this package. This is due to the package density being greater than 1000kg/m^3.

2.3.6.3 Puncture

- Three puncture tests were performed on 9975. The case judged in situ as being most vulnerable to further damage via puncture bar impact was the case where a local closure buckling occurred in the 30-ft slap-down drop test. A 1-ft angled top-down drop on the 40-inch puncture bar was performed to exploit the lid buckle, and attempt to tear open a gap. The test results demonstrate the acceptance of the 9975 Package as having sufficient margin against failure by puncture.

2.3.6.4 Thermal

- Compliance with the thermal requirements of HAC is demonstrated by analysis and by fire testing on packages 9973 and 9975. When exposed to 1475°F fire, the drum outer surface and the containment vessel assembly (with the source) will reach the maximum temperatures which are well below the design temperature of 500°F. Peak temperatures calculated in the thermal analysis were compared with the temperatures recorded during the fire tests on 9973 and 9975 Packages and were found to be consistently higher than the test temperatures. The calculated temperatures were then used to calculate peak vessel pressures and stresses. The stresses were found to be within the allowables. Peak temperatures during and after the fire test were consistent with temperatures used to determine the limiting stresses.

2.3.6.5 Immersion—Fissile Material

- The construction of the overpack for the 9975 Package is similar to that of the earlier 9966 Package. The water immersion test requirement for these packages is satisfied by the tests done on the 9966 Package.

2.3.6.6 Immersion—All Packages

- The response of a separate, undamaged specimen subjected to water pressure equivalent to immersion under a head of water at least 15 m (50 ft) was evaluated by analysis and found to be acceptable.

2.3.7 Lifting and Tie-Down Standards for All Packages

- This package has no lifting or tie-down devices.

2.3.8 Structural Evaluation of Special Pressure Conditions

- The contents of this package contain no irradiated nuclear fuel.

2.3.8.1 Analysis of Pressure Test

- The response of a separate, undamaged containment system specimen subjected 150% of its MNOP was evaluated by analysis and found to be acceptable.

2.3.9 Appendix

- The appendix includes background calculations and other appropriate supplemental information. In particular Appendices address:
    - Containment system stress and deflection calculations under both NCT and HAC
    - Containment system buckling calculations under both NCT and HAC
    - Containment system fatigue analysis under both NCT and HAC

2.4 Evaluation Findings

2.4.1 Findings

Based on review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated and that the package has adequate structural integrity to meet the requirements of 10 CFR 71. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

2.4.2 Conditions of Approval

- Maximum weight of the package shall not exceed 183kg (404 pounds).
- Maximum weight of the contents shall not exceed 20.1kg (44.4 pounds).

2.5 References

American Society for Mechanical Engineers, "Metal Containment Shell Buckling Design Methods," Code Case N-284, Section III, Div. I, Class MC, August 25, 1980.

American Society for Testing and Materials, "Standard Specification for Cellulosic Fiber Insulating Board," ASTM C208, Philadelphia, PA, 1995.

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," United Engineering Center, 345 East 47th Street, New York, NY, 1992.

Institute of Nuclear Materials Management, American National Standard for Radioactive Materials, "Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York.

International Atomic Energy Agency (IAEA), Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Compatibility with the International Atomic Energy Agency (IAEA)," 60 FR 50248, September 28, 1995, as amended.

Title 49, Code of Federal Regulations, Part 173 (49 CFR 173), "Shippers—General Requirements for Shipments and Packagings," 57 FR 20953, May 15, 1992, as amended.

U.S. Department of Transportation, "Specification 2R; Inside Containment Vessel," 49 CFR 178.360, 55 FR 52716, December 21, 1990.

U.S. Department of Transportation, "Specification 6M; Metal Packaging," 49 CFR 178.354, 55 FR 52716, December 21, 1990.

U.S. Nuclear Regulatory Commission, "Design Criteria for the Structural Analysis of Shipping Package Containment Vessels," Regulatory Guide 7.6.

U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Package Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m), Regulatory Guide 7.11.

U.S. Nuclear Regulatory Commission, "Load Combinations for the Structural Analysis of Shipping Packages for Radioactive Material," Regulatory Guide 7.8.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

3. THERMAL EVALUATION

3.1 Areas of Review

Chapter 3, Thermal Evaluation, of the Safety Analysis Report—Packages (SARP) for the 9975 Package was reviewed for the adequacy of the thermal design features of the 9975 Package with plutonium corresponding to Tables 1.14 and 1.15 of the SARP, including additional conditions related to the Tables given in Chapter 4 of the SER.

Included in the review:

3.1.1 Description of Thermal Design

- Design Features
- Decay Heat of Contents
- Codes and Standards
- Summary Tables of Temperatures
- Summary Table of Maximum Pressures

3.1.2 Material Properties, Thermal Limits, and Component Specifications

- Material Properties
- Temperature Limits
- Component Specifications

3.1.3 General Considerations for Thermal Evaluations

- Evaluation by Analysis
- Evaluation by Test
- Margins of Safety

3.1.4 Thermal Evaluation under Normal Conditions of Transport

- Initial Conditions
- Effects of Tests
- Maximum Normal Operating Pressure
- Maximum Thermal Stresses

3.1.5 Thermal Evaluation under Hypothetical Accident Conditions

- Initial Conditions
- Effects of Thermal Tests
- Maximum Temperatures and Pressures
- Maximum Thermal Stresses

3.1.6 Thermal Evaluation of Maximum Accessible Surface Temperature

3.1.7 Appendix

- Description of Test Facilities and Equipment
- Test Results
- Applicable Supporting Documents or Specifications
- Analyses Details

3.2 Regulatory Requirements

Regulatory requirements of 10 CFR 71 applicable to the thermal evaluation are as follows:

- The package design must be described and evaluated to demonstrate that it satisfies the thermal requirements of 10 CFR 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The package must be made of materials of construction that assure there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water, among the packaging components, among package contents, or between the packaging components and the package. The effects of radiation on the materials of construction must be considered. [§71.43(d)]
- The performance of the package must be evaluated under the tests specified in §71.71 for NCT. [§71.41(a)]
- The package must be designed, constructed, and prepared for shipment so there would be no loss or dispersal of contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging under the tests specified in §71.71 for NCT. [§71.43(f), §71.51(a)(1)]
- The package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade, the accessible surface temperature does not exceed 50°C (122°F) in a nonexclusive-use shipment or 85°C (185°F) in an exclusive-use shipment. [§71.43(g)]
- The performance of the package must be evaluated under the tests specified in §71.73 for HAC. [§71.41(a)]
- The package design must not rely on mechanical cooling systems to meet containment requirements. [§71.51(c)]

3.3 Review Procedure

The 9975 SARP includes the information essential for a thermal evaluation including drawings and the content decay heat. Of particular importance is the response of the containment vessel(s) and associated O-rings, the shielding, and the contents of the 9975 Package to the imposed NCT (10 CFR 71.71) and HAC (10 CFR 71.73).

3.3.1 Description of Thermal Design

3.3.1.1 Design Features

The applicant described the packaging components that control the response of the 9975 Package to the thermal environment. These components, which primarily include the cane fiberboard overpack and the containment vessel(s) are described in sufficient detail in Section 1.2.1 of the 9975 SARP to provide a sufficient basis for the thermal evaluation of the package.

The primary design features intended to protect the containment vessel(s) and O-rings of all the packages as well as the lead shielding of the 9975 Package from structural damage and overheating are:

- A cane fiberboard overpack confined in a steel drum which acts as an impact limiter and insulation during a hypothetical accident
- The stainless steel pressure vessel with cone seal plug and nut which provides the containment system of the package contents during NCT- and HAC-imposed structural loads. The containment system of the 9975 Package utilizes two nested concentric containment vessels. The containment boundary for each containment vessel is completed by the use of two Viton O-rings between the cone seal plug and the vessel.

All contents are packaged in inerted 3013 cans to less than 5% oxygen. For oxide contents, the primary containment vessel is diluted by a minimum of 75% with CO_2.

3.3.1.2 Contents Decay Heat

The maximum contents decay heat rate for the 9975 Package is given in Table 1.2 of the SARP. The maximum contents decay heat rate of 19 watts was used in the review of the thermal evaluation of the 9975 Package. This conforms to about 640 curies of plutonium isotopes with about 5 Mev alpha decay products.

3.3.1.3 Codes and Standards

The structural materials used in the package conform to Section III of the ASME B&PVC. The cane fiberboard used in the overpack conforms to ASTM Specification C208. The cast lead shield material conforms to ASTM B749. The plutonium contents defined in Tables 1.14 and 1.15 in the SARP conform to the DOE-STD-3013.

3.3.1.4 Summary Tables of Temperatures

The maximum temperatures reached in the 9975 Package components during NCT are given in Tables 3.3 and 2.14 of the SARP. These temperatures bound the various content configurations described in Figure 1 and Table 1 of Appendix 3.15.

The minimum temperature is –40°C based on the assumption that the package is without content heat generation in the shade.

For a 100°F environment temperature in the shade, the 9975 Package has the maximum accessible surface temperature below the limit of 122°F allowed for nonexclusive-use shipments.

The applicant presents the maximum temperature in the 9975 packaging components during a hypothetical accident fire in Tables 3.4 and 2.22 of the SARP. The post-fire cool-down did not include insolation. These results are based on tests as well as analysis of an undamaged 9975 Package with a simulated 21-watt content decay heat rate. Table 2 of Appendix 3.18 lists the maximum temperatures of a damaged 9975 Package determined by analysis, with 19-watt content decay heat rate and post-fire insolation, for the lead shield and the secondary containment vessel (including the O-ring). The temperatures of the other components presented in this table have not yet reached their maximum 4 hours following cessation of the fire. However, the temperatures for NCT bound the maximum temperatures of these components.

3.3.1.5 Summary Tables of Maximum Pressures in the Containment System

The MNOP in the PCV and SCV cavities of the 9975 Package for NCT are given in Tables 3.5 and 2.13 of the 9975 SARP. The maximum pressures in the 9975 containment system cavities during a hypothetical accident fire are given in Tables 3.6 and 2.23 of the 9975 SARP.

The pressures in the 9975 containment vessels are lower for the HAC than the MNOP. The initial temperatures prior to the hypothetical accident are based on the absence of insolation while the temperatures for the maximum normal operating condition are based on insolation on the package surface.

The package must be designated as a Type B(M) since the MNOP is greater than 700kPa (100 psig) per 10 CFR 71.4.

3.3.2 Material Properties and Component Specifications

3.3.2.1 Material Thermal Properties

The required thermal properties for all the materials used in the fabricated 9975 packaging were presented in Section 3.2 of the 9975 SARP. A small volume of the cane fiberboard exceeds the allowable temperature limit of 121°C (250°F) during normal operating conditions. A region of the cane fiberboard decomposes during the hypothetical thermal accident resulting in a change of the thermal properties during and following the thermal event. These properties were determined experimentally by the applicant (Hensel and Gromada 1994). The properties were reviewed by the staff and determined to be acceptable in both detail and accuracy.

3.3.2.2 Temperature Limits

The temperature limits of the lead shield, the primary and secondary containment vessels and their O-rings and the fiberboard are given in Table 3.1 of the SARP. The pressure limits of the primary and secondary containment vessels are also given in Table 3.1 of the SARP.

3.3.2.3 Component Specifications

The component specifications for the overpack drum, insulation, and containment vessels are presented in the SARP. Included in the component specifications are the emissivity and absorptivity of the overpack drum, the identification of the ASTM Specification C208 and temperature limits of the 15 lb/ft^3 cane fiberboard insulation, and the temperature limits of the Viton GLT fluoroelastomer O-rings used as closure seals.

3.3.3 General Considerations

3.3.3.1 Evaluation by Analysis

The applicant performed thermal evaluations using the finite element code P/Thermal with the pre- and post-processing software package PATRAN. The axisymmetric models were used for each package. The thermal properties of the packaging materials including the lead (where applicable), the insulation, and the air are appropriate for the thermal analyses of the package. The expressions for the various modes of heat transport at the package boundaries are appropriate. The PATRAN-PLUS and P/Thermal descriptions are given in Appendix 3.1 of the SARP. The material properties, convection coefficients and radiation surface properties, and internal and solar heat source data input to P/Thermal are also given in Appendix 3.1 of the SARP. The benchmarking of P/Thermal against a documented shipping package problem is described in Appendix 3.3.

The analyses of the undamaged 9975 Package for both the NCT and the HAC fire were benchmarked against experiments as discussed in Appendices 3.8 and 3.9 of the SARP. The analysis of the hypothetical accident fire of the damaged package utilized the cane fiberboard thermal properties inferred from experiments.

3.3.3.2 Evaluation by Test

Tests described in Appendix 3.7 of the 9975 SARP were performed on a prototype of the 9975 packaging (described in Appendix 3.8 of the 9975 SARP) not significantly different from the production design with a 22-watt heater to simulate the content decay heat rate. These tests were used to benchmark the analyses of the package. The package was tested for 120 hours in a building with an ambient temperature ranging between 77°F and 80°F. The measured temperatures in the package were used to benchmark the analyses of the 9975 Packages for NCT as described in Appendix 3.8 of the 9975 SARP.

Immediately following the test on the 9975 packaging for NCT, the package was tested in a vertical orientation in a radiant heat facility for greater than 30 minutes as described in Appendix 3.7 of the SARP. The temperature of the 35-gallon drum outer confinement vessel exceeded 1500°F for approximately 45 minutes. The insulation that covered the top and bottom of the facility to prevent heat loss during the heating cycle was removed and the package was allowed to cool 15 hours by radiation and natural convection to the ambient air near 100°F while remaining in the test facility. A member of the SARP review team witnessed this test. The staff has determined that it was appropriate not to furnish excess oxygen to replenish the oxygen depletion during the heating portion of this test. The measured temperatures in the package were used to benchmark the analyses of the 9975 Package under HAC as described in Appendix 3.9 of the 9975 SARP. The drop and puncture tests of the HAC had not been performed on the prototype 9975 packaging tested.

3.3.3.3 Margins of Safety

The temperatures and pressures for both the NCT and HAC are, with the exception of the cane fiberboard, substantially less than the allowable design limits given in Table 3.1 of the SARP. For NCT, the temperature of the cane fiberboard can exceed the allowable design limit by only a few degrees over a small, thin volume of material located near the bottom of the secondary containment vessel of a package that sits on an adiabatic surface. This “excess” temperature of the cane fiberboard will not adversely affect the package components important to containment, subcriticality, or shielding.

3.3.4 Thermal Evaluation for Normal Conditions of Transport

The applicant performed thermal evaluations of the various packages for NCT using analyses benchmarked against the experiment on the 9975 packaging using a 22-watt heater to simulate the content decay heat rate. The use of nominal thermal conductivity properties of the cane fiberboard results in the calculation of higher temperature gradients in the insulation than measured in the experiment. The cane fiberboard properties were not adjusted in the analytical model to duplicate the experimental results because the analytical results are conservative, producing higher values of temperatures in the package components important to safety.

The maximum accessible surface temperatures of the 9975 Package with the 19-watt content decay heat rate were determined without insolation based on the surface heat flow by natural convection and thermal radiation to the environment at an ambient temperature of 100°F. This surface temperature is less than 122°F, which is one condition for allowing the package to be transported as a non-exclusive-use shipment. The staff concurs with this analysis and conclusion. Thus, 10 CFR 71 Section 43(g) is satisfied.

The minimum temperature of -40^{o}C in the package occurs when the content decay heat load is zero in an environment at -40^{o}C. As noted in Section 2.3.6.2 of this SER, the *Cold* condition of -40^{o}C ambient temperature will not result in a degradation of the 9975 Package. The 304L austenitic stainless steels used for the containment vessels and the overpacked drum do not have a ductile-to-brittle transition temperature above -40°C. The secondary stresses from the differential thermal contraction for the *Cold* condition are less than those from the differential thermal expansions for the *Heat* condition.

The applicant performed a thermal evaluation for the 9975 Packages under NCT thermal conditions with insolation applied to the surfaces of the package in 100°F still air. The insolation is based on the appropriate values given in 10 CFR 71, Section 71(c) for a 12-hour time period. The solar absorptivity of the stainless steel drum surface was assumed to be 1.0 while the surface emissivity was assumed to be 0.21. The applicant evaluated two 3013 content configurations for shipping plutonium metal and one 3013 content configuration for shipping plutonium oxides. For each 3013 content configuration, the applicant determined (by analyses) the component temperatures for the package in the shade (steady state) as well as with insolation. The content decay heat rate of 19 watts was used in the analyses of the 9975 Packages. The maximum component temperatures are given in Table 3.3 of the SARP as described in Section 3.3.1.3, above. Confirmatory calculations by the staff of the package surface temperature and the content envelope surface temperature verify that the above results were reasonable and conservative. The steady-state temperatures of the package components during NCT do not compromise the functions of the packaging.

The MNOP in the 9975 containment vessel with oxide contents is due to the increased temperature of the cavity air initially at atmospheric pressure and 70°F temperature, the helium from the decay of the plutonium contents, the decomposition of 25 grams of moisture into hydrogen, and by thermal decomposition of the plastic bags per Appendix 3.4, Rev. 8. The MNOP calculated by the applicant is given for the PCV and the SCV in the Summary Table 3.5 of the SARP given in Section 3.3.1.4, above. This pressure, obtained for the case of oxide in food cans, is an upper bound for the containment vessels with metal oxide contents in a 3013 container. As shown in Chapter 2 of this report, this pressure does not produce stresses in the confinement vessel that exceed the allowable stress limits. A review of the calculations of the MNOPs confirmed that the pressure results were reasonable and conservative.

Pressures were estimated for the deflagration of the hydrogen produced from the decomposition of the 25 grams of moisture in a package with the oxide contents in a 3013 container. The peak pressure in the PCV is less than given in Summary Table 3.5 of the SARP. The peak pressure in the SCV exceeds that given in Summary Table 3.5, but is substantially less than the design pressure of the SCV. A review of the calculations of the deflagration pressures confirmed that the pressure results were reasonable and conservative.

The potential for detonation of the hydrogen produced from the decomposition of the 25 grams of moisture in a package with the oxide contents in a 3013 container was investigated. The use of an inerted 3013 container to less than 5% oxygen, with the primary containment vessel diluted by a minimum of 75% CO_2, is sufficient to prevent detonation within either the PCV or SCV. An independent analysis of the maximum cell size to prevent detonation within the 3013 container, the PCV and the SCV, confirmed that, with the inerted 3013 container and the primary containment vessel diluted by a minimum of 75% CO_2, the maximum cell size is larger than the maximum gaps and free spaces in the PVC and SCV of the 9975 Package with 3013 containers. This is sufficient to prevent detonation within either the PCV or SCV.

The thermal stresses in the 9975 Package due to the differential thermal expansions between the package components are small as shown in Chapter 2.

The staff finds that the containment vessels of the 9975 Package remain fully effective as containment boundaries for the payloads during the NCT or in the event of deflagration of hydrogen gases within the containment vessels. The resultant deformations, if any, of the vessel will not impair the containment, shielding, or criticality functions of the package. The staff finds that detonation of hydrogen gases within the containment vessels will not occur for an inerted 3013 container and the primary containment vessel diluted by a minimum of 75% CO_2. The staff also finds that the NCT do not impair the ability of the 9975 Package to withstand the HAC discussed below.

3.3.5 Thermal Evaluation of Hypothetical Accident Conditions

The thermal evaluations of the HAC [10 CFR 71 Section 73(c)(3)] were performed on the 9975 Package by test and analyses. The analysis was benchmarked against the experiment on the 9975 packaging that used a 22-watt heater to simulate the content decay heat rate. The use of the nominal cane fiberboard thermal conductivity properties results in the calculation of larger temperature gradients in the insulation for the initial conditions than measured in the experiment. For the HAC, the cane fiberboard properties were adjusted in the analytical model to duplicate the experimental results to more accurately produce the temperatures measured in the HAC benchmark test of the 9975 Package.

The undamaged 9975 Package was tested for 120 hours in a building with an ambient temperature ranging between 77°F and 80°F. Immediately following the test on the 9975 packaging, the package was tested in a vertical orientation in a radiant heat facility for greater than 30 minutes. The temperature of the drum surface exceeded 1500°F for approximately 45 minutes. The package was allowed to cool 15 hours by radiation and natural convection to the ambient air near 100°F while remaining in the test facility. An analysis was performed to determine the response of the 9975 Package to the experimental fire test conditions based on the initial conditions determined above. The analyses used the thermal properties of the uncharred and charred cane fiberboard based on the applicant’s high temperature tests specifically designed and performed to develop thermophysical property models.

The measured temperatures in the 9975 Package were used to benchmark the analyses of the 9975 Packages HAC. The calculated internal 9975 Package temperature histories compare well with the measured histories.

The fire test analyses were modeled as an undamaged package and used the thermal properties of the uncharred and charred cane fiberboard based on the applicant’s high temperature tests specifically designed and performed to develop thermophysical property models. The analysis of the drum wall temperature compares well with the experimental measurements, demonstrating that the analytical boundary conditions used in the analyses were appropriate. The calculated secondary containment vessel (SCV) seal and side temperatures were within 20°F greater than the measured temperatures. Because the calculated temperatures overestimated the measured temperatures of the package internals, the analytical models with the appropriate content heat were used to calculate the thermal response of the 9975 Packages to the regulatory HAC of a 30-minute, 1475°F fire. The maximum temperatures experienced by the 9975 Package components during the regulatory HAC are given in Table 3.4 of the 9975 SARP as described in Section 3.3.1.4, above. The temperatures of the package components during a HAC do not compromise the functions of the packaging.

A 9-m (30-ft), low-angle drop test of a 9975 Package resulted in gaps forming between the radial cane fiberboard sheets. A hypothetical accident thermal analysis of a 9975 Package with a separation between the radial cane fiberboard sheet caused by the 9-m, low-angle drop was performed. Table 2 of Appendix 3.18 lists the maximum temperatures of a damaged 9975 Package determined by analysis, with 19-watt content decay heat rate and post-fire insolation, for the lead shield and the secondary containment vessel (including the O-ring). The temperatures of the other components presented in this table have not yet reached their maximum 4 hours following cessation of the fire. However, their temperatures for NCT bound the maximum hypothetical accident temperatures of these components.

The maximum pressure in the containment vessels is due to the increase of the temperature of the cavity air initially at atmospheric pressure and 70°F temperature, the helium from the decay of the plutonium contents, the hydrogen and oxygen produced by radiolysis (per Appendix 3.12, Rev 6) of the moisture associated with the PuO_2 contents, the saturated water vapor, and the hydrogen produced by the radiolysis of the plastic bags used with food cans (per Appendix 3.4, Rev 8). The maximum pressure in a hypothetical accident calculated by the applicant for oxide contents is given for the PVC and the SCV in Table 3.6 of the SARP as described in Section 3.3.1.4, above. These pressures bound the pressures produced in the containment vessels with metal contents. As shown in Chapter 2 of this report, these pressures do not produce stresses in the confinement vessel that exceed the allowable stress limits. A review of the calculations of the pressures produced during a hypothetical accident confirmed that the pressure results were reasonable and conservative. Also, as shown in Chapter 2, the thermal stresses in the 9975 Package due to the differential thermal expansions between the package components are small.

The staff finds that the containment vessels of the 9975 Package remains fully effective as containment boundaries and shielding for the payloads of plutonium metal or oxides during the HAC. The resultant deformations, if any, of the vessel will not impair the containment function of, or allow water leakage into the payload. While the applicant has conservatively assumed the lead shield is absent and that the shielding of the radiation from the metal contents furnished by the containment vessel will satisfy 10 CFR 71.51(2) as given in SER Chapter 5, the staff finds that the lead shielding of the 9975 Package remains fully effective as a shield for the payload source term during the HAC, and that the resultant deformations, if any, of the lead shield will not impair the shielding function of the payload source term. Thus, the functions of the 9975 Package are not affected by the HAC.

3.3.6 Appendices

The evaluations of several thermal properties of the packaging components are presented in the appendices of Chapter 3 of the SARP. These properties include the thermal radiation properties of stainless steel at 400 K (Appendix 3.2) as well as the calculations of the thermal properties of the aluminum honeycomb used as an impact absorber and spacers in and between the containment vessels (Appendix 3.6).

The PATRAN-PLUS and P/Thermal codes used in the analyses of the thermal responses of the 9975 Packages to normal operating conditions and a hypothetical fire are described. Included in the description are the listings of the material properties data file, the file containing the convection correlation parameters and the radiative surface properties. The file contains internal and solar heat source data (Appendix 3.1). The benchmark of the P/Thermal code against a documented shipping package thermal problem is also presented (Appendix 3.3). The results indicate that the analysis code P/Thermal computes the thermal response of the benchmark problem to an acceptable accuracy.

The thermal tests were performed on the 9975 Packages. The package configurations most vulnerable to a fire were selected for testing. The most vulnerable damaged package is the 9973 after an axial drop, while the most vulnerable undamaged package is the 9975 (Appendix 3.10). The thermal tests of the 9973 and 9975 Packages were performed at Sandia National Laboratory. The hypothetical fire was simulated in Sandia's radiant heat facility. The test report, including the test plan and the assembly instructions for the instrumented 9973 and 9975 Packages, is presented (Appendix 3.7). The 9975 Package included the 22-watt heater to simulate the content heat source. The content heat simulator preheated the undamaged package until the package reached normal operating conditions, at which time the package was placed in the radiant heat facility. The damaged 9973 Package, which did not contain a content heat simulator, was placed directly into the radiant heat facility. The test report includes the measured temperature histories of various components.

The analyses of the 9975 Package using P/Thermal were compared to the results obtained from the tests of the package. The analytical model was adjusted to bring the calculated temperatures of the 9975 Package under NCT into near compliance with the measured results (Appendix 3.8), and the calculated temperatures of the 9975 Package under the thermal portion of the HAC into near compliance with measured results (Appendix 3.9). The benchmarked models were then used to perform the analyses of all the 9975 Packages with their content heat sources under HAC (Appendix 3.11). An analysis of the 9975 Package for NCT with 3013 contents was also performed (Appendix 3.15). A hypothetical accident analysis of a 9975 Package with a separation between the radial cane fiberboard sheet caused by a 9-m (30-ft), low-angle drop was also performed (Appendix 3.18).

The pressures in the containment system (primary containment vessel and secondary containment vessel) are due to the fill gas, decomposition of the O-ring seals, the helium from the decay of the plutonium contents, the hydrogen produced by decomposition of the moisture associated with the PuO_2 contents, and the hydrogen produced by the decomposition of the plastic bags used with food cans. The pressure due to the decomposition of the plastic bags was estimated for food pack cans (Appendix 3.4). The total pressure from impure Pu oxide in a 3013 system from the fill gas, helium generation, and hydrogen generated from the decomposition of moisture in the primary containment vessel was analyzed (Appendix 3.5). The 3013 system does not include plastic bags. The pressure in the secondary containment vessel—assuming a leaking primary containment vessel—and 3013 system was calculated (Appendix 3.17).

An analysis of the pressure produced from the deflagration of flammable gas mixtures from the hydrogen produced from the decomposition of the moisture associated with Pu oxides in both the primary and secondary containment vessels was performed for both food pack cans and the 3013 vessel (Appendix 3.16). Food pack cans are not authorized for oxide shipments in the 9975. An analysis was also performed on the effect of (1) inerting the 3013 container to less than 5% oxygen for oxide contents and (2) diluting the primary containment vessel by a minimum of 75% with CO_2 (Appendix 3.19). The pressure rise from the diluted system is less than the system analyzed in Appendix 3.16.

Detonation cell widths in the 9975 Package, with CO_2 diluting the primary containment vessel, were estimated (Appendix 3.20). Stack-up dimensions of the 9975 packaging components in NCT and HAC were determined (Appendix 3.21). The maximum allowable gap sizes were determined and can be compared to the maximum detonation cell sizes.

Shipments of plutonium in either the oxide or metal form are usually made in an SST. To the extent that the trailer is well insulated, and the package contents generate a substantial quantity of heat, the loss of cooling capacity in the trailer could result in an increase in the ambient temperatures in the trailer that exceed regulatory NCT (100°F) with the resulting increase in the temperatures of the package internal components. A transient analysis was performed on the 9975 Packages with an adiabatic boundary condition applied for 36 hours to the drum surface followed by natural convection cooling to 100°F ambient air. The maximum component temperatures and the containment pressures never exceeded their design allowable temperatures (Appendix 3.13).

3.4 Evaluation Findings

3.4.1 Findings

Based on review of the statements and representations in the application, the staff concludes that the thermal design of the 9975 Package with metal and oxide contents has been adequately described and evaluated, and that the thermal performances of the 9975 Package meet the thermal requirements of 10 CFR 71. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

3.4.2 Conditions of Approval

- The conditions of approval for the 9975 Package for the shipment of uranium or plutonium metal that conforms to Table 1.14 and 1.15 of the 9975 SARP and to the DOE-STD-3013 must include a decay heat limit of 19 watts.
- The maximum allowable polyethylene in the package contents of the 9975 Package is limited to a total of 100 grams.
- The 3013 container must be inerted to less than 5% oxygen for oxide contents.
- The primary containment vessel for oxide contents must be diluted by a minimum of 75% with CO_2.

3.5 References

American Society for Testing and Materials, “Standard Specification for Cellulosic Fiber Insulating Board,” ASTM C208, Philadelphia, PA, 1995.

American Society for Testing and Materials, “Standard Specification for Lead and Lead Alloy Strip, Sheet, and Plate Products,” ASTM B749, Philadelphia, PA, 1995.

Hensel, S. J. and R. J. Gromada, “Development of a Simulation (Thermophysical Property) Model for Cane Fiberboard Packages Subjected to a Hypothetical Accident Fire,” DOE Defense Programs Packaging Workshop, Knoxville, TN, 1994.

International Atomic Energy Agency (IAEA), Safety Series No. 6, “Regulations for the Safe Transport of Radioactive Material,” 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), “Compatibility with the International Atomic Energy Agency (IAEA),” 60 FR 50248, September 28, 1995, as amended.

U.S. Department of Energy Standard, “Stabilization, Packaging, and Storage of Plutonium Bearing Materials,” DOE-STD-3013-2000, September 2000, Washington, DC.

Westinghouse Savannah River Company, “Safety Analysis Report–Packages 9972-9975 Packages (U),” Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

4. CONTAINMENT REVIEW

4.1 Areas of Review

Chapter 4, Containment Review, of the Safety Analysis Report —Packages (SARP) for the 9975 Package was reviewed for the adequacy of containment and transport of plutonium metal or oxide contents (as defined in Tables 1.14 and 1.15 of the SARP) that are treated in accordance with the requirements in DOE-STD-3013-2000. The Containment review included the following:

4.1.1 Description of Containment Design

- Design Features
- Codes and Standards
- Special Requirements for Plutonium

4.1.2 General Considerations for Containment Evaluations

- General Containment Considerations for Type B Packages
- Combustible-Gas Generation

4.1.3 Containment Under Normal Conditions of Transport

- Containment Design Criterion
- Demonstration of Compliance with Containment Design Criterion

4.1.4 Containment Under Hypothetical Accident Conditions

- Containment Design Criterion
- Demonstration of Compliance with Containment Design Criterion

4.1.5 Leakage Rate Tests for Type B Packages

4.1.6 Appendix

4.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Containment review of the 9975 package are as follows:

- The package design must be described and evaluated to demonstrate that it meets the containment requirements of 10 CFR 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package. [§71.43(c)]
- The package must be made of materials and constructed to assure that there will be no significant chemical, galvanic, or other reactions, including reactions due to possible inleakage of water, among the packaging components, among package contents, or between the packaging components and the contents. The effects of radiation on the materials of construction must be considered. [§71.43(d)]

- Compliance with the permitted activity release limits for Type B packages may not rely on filters or on a mechanical cooling system. [§71.51(c)]
- The package may not incorporate a feature intended to allow continuous venting during transport. [§71.43(h)]
- Any valve or similar device on the package must be protected against unauthorized operation and, except for a pressure relief valve, must be provided with an enclosure to retain any leakage. [§71.43(e)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- A package containing plutonium in excess of 0.74 TBq (20 Ci) must satisfy the special containment requirements for plutonium. [§71.63]
- The package must be designed, constructed, and prepared for shipment to ensure no loss or dispersal of radioactive contents under the tests specified in §71.71 for NCT. [§71.43(f)]
- A Type B package must meet the containment requirements of §71.51(a)(1) under the tests specified in §71.71 for NCT.
- A Type B package must meet the containment requirements of §71.51(a)(2) under the tests specified in §71.73 for HAC.

4.3 Review Procedures

The following procedures were employed in the review of Chapter 5, Containment, of the SARP. These procedures correspond to the Areas of Review listed in Section 4.1 of this SER.

4.3.1 Description of the Containment Design

4.3.1.1 Design Features

4.3.1.1.1 Containment Boundary

The boundary of the containment system is described in Sections 1.2.1.6 and 4.1 of the SARP. The containment boundary for the 9975 Package consists of the containment vessel body, the male cone seal, the outermost of two O-ring, and the leak test port plug. The closure seal is formed with the O-rings between the female cone-sealing surface on the containment vessel body and the male cone sealing surface. The O-rings are secured by tightening down the cone seal nut against the male cone seal. The leak test port is sealed by tightening the gland nut, which presses the tip of the plug into the port. The seal is formed by the metal-to-metal contact between the conical tip of the plug and the corresponding conical surface of the outer edge of the port. The components of the containment system are shown in the following drawings in the SARP: R-R1-F-0005, Rev. 4; R-R2-F-0018, Rev. 1; R-R3-F-0016, Rev. 2; and R-R4-F-0054, Rev. 1.

Within the containment boundary, the plutonium metal is confined either in product cans(s) (food pack cans) or a 3013 container. The plutonium oxide contents will be packaged in 3013 containers that are inerted to less than 5% oxygen. For oxide contents, the primary containment vessel contains a gas mixture of air and a minimum of 75% CO_2.

4.3.1.1.2. Containment Boundary Penetrations

The 9975 Package has a single containment boundary penetration, i.e., the leak test port described in the previous section. As was noted in the previous section, the leak test port is sealed by tightening the gland nut, which presses the tip of the plug into the port. The seal is formed by the metal-to-metal contact between the conical tip of the plug and the corresponding conical surface of the outer edge of the port.

4.3.1.1.3. Seals and Welds

The seals and welds on the containment boundary are adequately described in Section 4.1.3 of the SARP. Although two O-rings are used to seal the containment vessel the outer O-ring is considered part of the containment boundary. The inner O-ring is used to facilitate leakage testing. To prevent movement, each O-ring is placed in a machined groove on the conical surface of the male cone seal. The seal is formed when the male cone seal is pressed against the female conical surface on the inner wall of the containment vessel body. To meet the design criteria for this application, the O-rings must maintain their seal at internal temperatures of up to 400°F and at an internal pressure of up to 900 psig. The elastomer selected for the O-rings is a Viton GLT fluorocarbon (Parker Compound V835-75 or equivalent). The normal operating range for the Viton GLT O-rings is -40°F to 400°F. Under NCT, the maximum temperature that the O-rings are expected to reach is 272°F in the primary containment vessel and 268°F in the secondary containment vessel. Under HAC, the maximum temperature that the O-rings are expected to reach is 197°F in the primary containment vessel and 192°F in the secondary containment vessel. The review confirmed that the maximum and minimum temperatures of seals, under NCT and HAC, are within the manufacturer's recommended operating ranges. The O-ring lid seals are appropriate for use in the 9975 Package with the plutonium metal and oxide contents if the seal grooves are properly sized.

The leak test port is a ¼-inch steel plug designed for high-pressure service. The leak test port plug forms its seal at the outer edge of the leak test port in the top of the male cone seal.

Each containment vessel has two circumferential, full-penetration butt welds. The top circumferential weld joins the female conical section to the Schedule 40 vessel-body pipe section. The bottom circumferential weld joins the standard weight pipe cap to the Schedule 40 vessel body pipe section. Welding qualifications are established in accordance with Section IX of the ASME B&PVC, 1992 edition. The welds are examined with liquid penetrant and are fully radiographed after completion.

4.3.1.1.4. Containment Closure

The closure of the containment system is adequately described in Section 4.1.4 of the SARP. Closure of the containment boundary is accomplished by forming a leaktight seal with the Viton GLT O-rings between the female conical section of the containment vessel and the male cone plug wall. The female conical surface (20° included angle) is machined into the inner wall of the containment vessel weldment and finished to RMS 32 surface finish. Female threads are cut into the containment vessel wall outboard of the conical surface. A male cone, also with a 20° included angle, forms the removable plug for the seal.

Two O-ring grooves are cut into the conical surface of the male cone. The O-ring and its groove volume are equal. This provides sealing on all four surfaces of each groove and aids in providing very low leakage and permeation rates. The male cone seal is pressed into place by a threaded nut made from a dissimilar material

(Nitronic 60 stainless steel alloy) to prevent galling with the Type 304 or Type 304L stainless steel containment vessel and cone seal.

A shallow circumferential rectangular groove (0.063-inches wide × 0.060-inches deep) between the O-rings is also machined into the male cone seal. The rectangular groove intersects with the leak test port opening at the cone surface between the two O-ring grooves. The rectangular groove provides a channel to ensure that the test gas is applied against the entire inner and outer O-ring sealing surface during leakage testing.

A point of reference is established for tightening the male cone seal by first seating the joint metal-to-metal. This is accomplished by assembling the joint without the two O-rings and tightening the cone seal nut to 25 ft-lb. A radial line is then scribed across both the top of the cone nut and the top of the containment vessel body. When the cone closure is assembled with the two O-rings installed, the two radial lines must line up to within 1-inch when the prescribed torque is applied. With this match a maximum radial clearance of 0.0007-inches exists between the male and female cone components. This clearance is adequate to prevent the O-rings from extruding from the grooves under design conditions. The closure on the primary containment vessel is torqued to 50 ft-lb. The closure on the secondary containment vessel is torqued to 100 ft-lb. It was verified, through coordination with the structural review, that the specified bolt torques provide proper compression for containment seals.

It was verified that the method of closure for the containment boundary penetrations is adequately described and that the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

4.3.1.2 Codes and Standards

The review verified that the codes or standards applicable to the containment design of the package were identified and appropriate, including those for material specifications and fabrication. The review ensured that such codes and standards were consistent with those specified in the General Information, Structural, and Thermal Evaluation chapters of the SARP. The review determined that these codes or standards specify temperature limits for materials, that the temperatures of all the containment system components are within their respective allowable temperature limits, and that the temperatures used are consistent with those used in the Thermal and Structural chapters of the SARP. The plutonium metal and oxides conform to DOE-STD-3013.

The review confirmed that the evaluation of release rates and performance of leakage testing was in accordance with the American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment, ANSI N14.5.

4.3.1.3 Special Requirements for Plutonium

The requirements specified in 10 CFR 71.63(b) state that all plutonium bearing materials in excess of 20 Ci must be provided with double-containment for shipment, with the following exceptions: reactor fuel elements, metal or metal alloys, glass logs certified for high-level waste, and any other plutonium bearing solids that the Commission determines should be exempt from the double-containment requirement. Although not necessary for the shipment of high-purity plutonium metal and alloys, the applicant has elected to use the double-containment approach for the shipment of all plutonium bearing materials. (The conclusions reached in Section 6 of this SER have also determined that the double-containment requirement specified in 10 CFR 71.63(b) must be invoked for criticality safety.) The review verified that each containment system separately satisfies the requirement of §71.51(a)(1) for normal conditions of transport and §71.51(a)(2) for hypothetical accident conditions.

Because the 9975 package is to be used for the shipment of unirradiated plutonium bearing materials, additional requirements for spent nuclear fuel are not applicable to this SER.

4.3.2 General Considerations for Containment Evaluations

4.3.2.1 Type B Packages

The 9975 is a Type B package and must satisfy the quantitative *release* rates specified in §71.51(a)(2) for normal conditions of transport, and hypothetical accident conditions, respectively. The double-containment requirements specified in 10 CFR 71.63(b) apply to the package (see SER Section 4.3.1.3). As is also noted in the NRC's Regulatory Guide 7.4, the methods outlined in ANSI N14.5 provide an acceptable method to determine the maximum permissible volumetric leakage rates for both containment vessels based on the allowable release rates as specified in §71.51(a)(1) and §71.51(a)(2) respectively.

In order to meet the requirements specified in §71.51(a)(1), §71.51(a)(2), §71.63(b), and Regulatory Guide 7.4, the applicant has elected to adopt the ANSI N14.5 definition of leaktight, for both containment boundaries, for both normal conditions of transport and hypothetical accident conditions. (Note: According to ANSI N14.5, leaktight is defined as being a leakage rate of air that is less than or equal to 1×10^{-7} reference cm^3/sec, at an upstream pressure of 1 atmosphere and a downstream pressure of 0.01 atmosphere or less, regardless of the type or the form of radioactive contents.) In order to verify that the ANSI N14.5 specification of leaktight can be met for all required leakage tests, a sensitivity of 5.0×10^{-8} reference cm^3/sec has also been adopted by the application.

The review also verified that the package does not incorporate a feature intended to allow continuous venting during transport and that the containment system does not rely on filters or a mechanical cooling system.

Applicant analyses presented in the containment section of the SARP are based on a bounding content described in Table 1.5 of the SARP. This content type consists of $^{238}PuO_2$ heat source plutonium. The staff review also used this bounding plutonium isotopic composition for analyses described in the containment section. The use of $^{238}PuO_2$ results in conservative limits for containment releases.

4.3.2.2 Combustible-Gas Generation

The review staff has determined that when shipping the plutonium metal, contents identified in Table 1.14 and any combustible gases generated in the package during a period of one year do not exceed 5% (by volume) of the free gas volume in any confined region of the package.

The staff determined that the only flammable species potentially present in the 9975 Package with plutonium metal contents would be due to the radiolysis of the water in the containment vessel fill gas and the water adsorbed onto the thin oxide layer on the plutonium metal surface. Radiolysis of the small amount of water vapor in the fill gas within the inner product can (if any) and the water absorbed onto the thin plutonium oxide layer would result in a hydrogen concentration below the lower flammability limit of hydrogen in air of 5 molar percent.

The staff has determined that the only flammable species potentially present in the 9975 Package with plutonium oxide contents would be due to the hydrogen produced by the decomposition of the moisture associated with the oxide contents, which is limited to 25 grams of moisture by the 3013 Standard. While the hydrogen produced by the decomposition of the 25 grams of moisture exceeds the lower flammability limit of hydrogen in air of 5 molar percent, the oxygen concentration in the 3013 container is not allowed to exceed 5% by volume at any time after final seal welding of the container. The oxygen content in the primary containment vessel is reduced by inerting the 3013 cans and diluting the primary containment vessel by a minimum of 75% CO_2.

The staff also confirmed applicant analyses that demonstrated that failure to inert either the 3013 container or primary containment vessel would not result in conditions outside the containment design conditions.

4.3.3 Containment Under Normal Conditions of Transport (Type B Packages)

Containment under NCT is addressed in Section 4.2 of the SARP.

4.3.3.1 Containment Design Criterion

As noted in Section 4.3.2.1 of this SER, the applicant has elected to adopt the ANSI N14.5 definition of leaktight for both containment boundaries for normal conditions of transport. This was verified as part of the Containment review.

The review also verified that the maximum normal operating pressure and maximum temperature under normal conditions of transport are consistent with those determined in the Thermal Evaluation chapter of the SARP.

4.3.3.2 Demonstration of Compliance with Containment Design Criterion

The applicant has demonstrated the containment design and performance criteria by test. The review confirmed that the SARP demonstrates that the package meets the containment requirements specified in §71.51(a)(1) for normal conditions of transport.

4.3.4 Containment Under Hypothetical Accident Conditions (Type B Packages)

The review procedures for containment under HAC were similar to those under NCT. Containment under HAC is addressed in Section 4.3 of the SARP.

4.3.4.1 Containment Design Criterion

As noted in Section 4.3.2.1 of the SER, the applicant has elected to adopt the ANSI N14.5 definition of leaktight for both containment boundaries for hypothetical accident conditions. This was verified as part of the Containment review.

4.3.4.2 Demonstration of Compliance with Containment Design Criterion

The applicant has demonstrated the containment design and performances criteria by test. Also, as was demonstrated in the Structural and Thermal evaluation chapters of the SARP, the package closure system is not degraded by any of the hypothetical accident condition tests. The review confirmed that the SARP demonstrates that the package meets the containment requirements specified in §71.51(a)(2) for hypothetical accident conditions.

4.3.5 Leakage Rate Tests for Type B Packages

The review confirmed that the maximum allowable leakage rates were determined in accordance with ANSI N14.5. The fabrication, periodic, and maintenance leakage rate test criteria are each specified to meet the ANSI N14.5 definition of leaktight, i.e., # 1×10^{-7} reference cm^3/sec under reference air leakage test conditions. This was also verified in the Acceptance Test and Maintenance Program chapter of the SARP, i.e., Chapter 8. The pre-shipment leakage rate test criterion is 10^{-3} reference cm^3/sec, which is also consistent with ANSI N14.5. This was verified in the Operating Procedures chapter of the SARP, i.e., Chapter 7.

4.3.6 Appendix

Chapter 4 of the 9975 SARP contains one appendix, which provides a bounding calculation of the oxidation of plutonium metal exposed to humid air in the inner containment vessel. Originally written for the 9965 SARP, the analysis presented is no longer relevant to the 9975 packaging.

4.4 Evaluation Findings

The review ensured that the information presented in the SARP supports a conclusion that the regulatory requirements in Section 4.2 above are satisfied.

Based on review of the statements and representations in the 9975 SARP, the staff concludes that the containment design has been adequately described and evaluated and that the package design meets the containment requirements specified in 10 CFR 71. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

4.4.1 Conditions of Approval

The following are given as conditions of approval for use of the 9975 Package to transport plutonium metal and oxide:

- The plutonium contents are limited to material identified in Table 1.14 and 1.15 of the SARP.

4.5 References

American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code,” United Engineering Center, 345 East 47th Street, New York, NY, 1992.

Institute of Nuclear Materials Management, American National Standard for Radioactive Materials, “Leakage Tests on Packages for Shipment,” ANSI N14.5-1997, New York.

International Atomic Energy Agency (IAEA), Safety Series No. 6, “Regulations for the Safe Transport of Radioactive Material,” 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), “Compatibility with the International Atomic Energy Agency (IAEA),” 60 FR 50248, September 28, 1995, as amended.

U.S. Department of Energy Standard, “Stabilization, Packaging, and Storage of Plutonium Bearing Materials,” DOE-STD-3013-2000, September 2000, Washington, DC.

U.S. Nuclear Regulatory Commission. “Leakage Tests on Packages for Shipment of Radioactive Materials,” Regulatory Guide 7.4, June 1975.

Westinghouse Savannah River Company, “Safety Analysis Report–Packages 9972-9975 Packages (U),” Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

5. SHIELDING REVIEW

Chapter 5, Shielding, in the Safety Analysis Report —Packages (SARP) for the 9975 Package was reviewed for external radiation requirements.

5.1 Areas of Review

The Shielding review included the following:

5.1.1 Description of Shielding Design

- Design Features
- Codes and Standards
- Summary Table of Maximum Radiation Levels

5.1.2 Radiation Source

- Gamma Source
- Neutron Source

5.1.3 Shielding Model

- Configuration of Source and Shielding
- Material Properties

5.1.4 Shielding Evaluation

- Methods
- Input and Output Data
- Flux-to-Dose-Rate Conversion
- External Radiation Levels

5.1.5 Appendix

5.2 Regulatory Requirements

Regulatory requirements of 10 CFR 71 applicable to the shielding review are as follows:

- The package design must be described and evaluated to demonstrate that it meets the shielding requirements of 10 CFR 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Under the tests specified in §71.71 for NCT, the external radiation levels must meet the requirements of §71.47(a) for non-exclusive-use or §71.47(b) for exclusive-use shipments. [§71.47]

- The package must be designed, constructed, and prepared for shipment so that the external radiation levels will not significantly increase under the tests specified in §71.71 for NCT. [§71.43(f), §71.51(a)(1)]
- Under the tests specified in §71.73 for HAC, the external radiation level must not exceed 10 mSv/h (1 rem/h) at one meter from the surface of a Type B package. [§71.51(a)(2)]

5.3 Review Procedures

Chapter 5 of the 9975 SARP includes the information essential for a shielding evaluation including: the drawings, the packaging materials and densities, and the radioisotopic composition and mass. The shielding information in the 9975 SARP was reviewed by the staff for completeness and compliance with regulatory requirements.

5.3.1 Description of Shielding Design

5.3.1.1 Design Features

The 9975 Package design includes a double containment system. The radioisotopic contents are generally placed in a product or convenience can. For metals, from one to three product cans may be placed within the PCV. Oxides must be enclosed in a 3013 container. Therefore, if oxide is in a convenience can it must be placed in a 3013 container, which in turn is placed within the PCV. This SER applies to plutonium metal or plutonium oxide with limits of 500 g of beryllium and/or 1.0 kg of graphite impurities. These materials are specified as Content H for plutonium metal in Table 1.14 and as Content I for plutonium oxide in Table 1.15.

The design of the 9975 Package does not include specific neutron-absorbers, but it does include hydrocarbon insulating-spacing material for thermal insulation. This insulation material also serves as a neutron moderator for neutron dose shielding, although no credit is taken for it in HAC studies. The design of the 9975 Package also includes a layer of lead surrounding the containment system for gamma dose shielding. Shielding control through package geometry occurs because the minimum package length and diameter provide a minimum separation between the radioisotopes and the package surface. Therefore, the various dose measurements required must be at least an assured minimum distance from the radioisotopic sources.

The quantity and composition of the radioisotopes is also considered to be a primary shielding control for the 9975 Package. Because the expected neutron dose rates, predominantly from (α,n) reactions, for either Content H or Content I with 500 g of beryllium fully homogenized is much greater than the regulatory limits, a bounding shielding analysis for neutrons is not possible. It is necessary to use another methodology to make certain that the 9975 dose rates will not exceed regulatory limits. This methodology must ensure that (1) the composition and form of the material to be shipped cannot change during shipment to another composition and form that generates a neutron dose rate that exceeds these limits, or (2) the contents cannot change position and cause the neutron dose rate to exceed these limits.

The radioactive material to be shipped is either metal or oxide, and the thermal environment expected during HAC will not generate temperatures sufficient to melt either the metal or the oxide. The minimum allowed size of metal pieces is too large to permit significant dose rate changes arising from rearrangement of the metal pieces. This is not the case for oxides where grains are in the range of about 5 to 250 μm in diameter. However, process steps leading to oxide calcination, together with process experience, in general, ensure that significant dose rate changes arising from rearrangement of the oxide grains do not result. This means that changes in the composition and form of the packaging contents during shipment will not lead to a significant change in dose rate. However, this radioactive material could move to another location during transport, thus allowing higher dose rates to result. In addition, the dose rate measuring instruments have inherent uncertainties that must be considered to preclude measuring a higher dose rate after the shipment has begun. The SARP chooses to control these variabilities by introducing dose rate correction factors that reduce the allowed limiting dose rates to lower values that will prevent the regulatory limits from being exceeded at any time during a shipment due to content movement and/or detector uncertainties. These features, including those mentioned in the preceding paragraphs, ensure that the single package and contents meet the shielding criteria under NCT and HAC.

The staff confirms that the shielding design features presented in the General Information and Shielding Evaluation chapters of the SARP are consistent and complete concerning location, dimensions, tolerances, and densities of material for gamma and neutron shielding, including those packaging components considered in the shielding evaluation. In addition, the structural components that maintain the integrity of the shielding and the contents in restricted locations within the package are sufficient. Heat transfer is also sufficient to maintain allowable temperatures of the shielding so that the lead does not melt.

The staff confirms that the text and sketches describing the shielding design features are consistent with the engineering drawings and the models used in the shielding evaluation. The staff concludes that the 9975 Package conforms to the general standards for all packages as prescribed by 10 CFR 71 [§71.31(a)(1), §71.31(a)(2), §71.31(c), §71.33, §71.35(a)].

The SARP has demonstrated that the measurements conducted using the dose rate correction factor methodology developed to account for the uncertainty in measured dose rate will ensure that the maximum measured dose rates would be less than 200 mrem/h at the package surface and less than 10 mrem/h at 1 m from the package surface. Therefore, the measured shielding transport index (TI) would be less than 10 for the proposed contents for the 9975 Package. (A projected maximum NCT dose rate of 10 mrem/h at 1 m for each package surface type is the TI as prescribed by 10 CFR 71 [§71.4]). The radiation dose rates for the 9975 Package are less than the limits prescribed in 10 CFR 71 [§71.47(a)], so that this package and payload can be shipped by non-exclusive use. Therefore, no specific dimensions of the transport vehicle are required.

The PCV for the 9975 Package consists of a cylindrical pressure vessel constructed from 5-inch, Schedule 40, Type 304L stainless steel pipe. The SCV is constructed from 6-inch, Schedule 40, Type 304L stainless steel pipe. Both the PCV and the SCV comply with the stress criteria of the ASME B&PV Code Section III, Subsection NB. The PCV is placed within the SCV. The PCV-SCV combination is placed within a specially fabricated 35-gallon removable-head drum constructed of Type 304L stainless steel with a minimum OD of 18.22 inches (the drum rolling hoops are somewhat larger and are responsible for a slightly larger minimum diameter). The PCV-SCV combination is enclosed within a 0.5-inch-thick layer of lead which is kept centered within the drum by about 11 inches of fiberboard insulation material.

5.3.1.2 Codes and Standards

The flux-to-dose-rate conversion factors are listed in Appendix 5.1 and are consistent with ANSI 6.1.1-1977.

The 9975 Package containment vessel design for the PCV and the SCV complies with the stress criteria of the ASME B&PV Code Section III, Subsection NB.

5.3.1.3 Summary Table of Maximum Radiation Levels

The summary table, Table 5.1, of maximum radiation levels does not present the maximum possible dose rates for the proposed contents for either NCT or HAC for Content H or Content I. However, for the gamma dose rates, the values listed for Content E in that table would bound the gamma dose rates for Content H. The expected gamma dose rates for plutonium metal from Content H in Table 1.14 bounds the expected gamma dose rates for plutonium oxide from Content I in Table 1.15. For the neutron dose rates, the proposed contents would far exceed the regulatory limits if all beryllium impurities were homogeneously mixed with either the plutonium metal or the plutonium oxide. Dose rate measurements, restricted by a dose rate correction factor methodology presented in the SARP, and taken at the appropriate locations for non-exclusive use shipments, will be used to ensure that the 9975 Package and payload are within the regulatory limits for NCT as required by 10 CFR 71 [§71.47(a)], for non-exclusive use shipment of 200 mrem/h at the package surface, and 10 mrem/h at 1 m from that surface. In addition, the radiation levels for the 9975 Package will be within the regulatory limits for HAC as required by 10 CFR 71 [§71.51(a)(2)], of 1000 mrem/h at 1 m from the package surface. The dose rate measurements performed under the dose rate correction factor methodology account for potential variability in measured dose rate and will ensure that the maximum measured dose rates do not exceed regulatory limits. The staff maintains that the maximum neutron and gamma dose rates for the proposed contents do belong in a summary table, even though the SARP intends to satisfy the regulatory dose rate limits by measurements and not by choosing contents that are inherently bounded by the regulatory limits.

5.3.2 Radiation Source

The contents used in the shielding analysis are consistent with those specified in the General Information section of the SARP. There are two payloads for the 9975 Package currently seeking certification, namely, plutonium metal with impurity limits as given for Content H and plutonium oxide with impurity limits as given for Content I. Each location is clearly identified where the highest external dose rates are expected for either Content H or Content I. These locations will be included among the positions at which dose rates are to be measured.

5.3.2.1 Gamma Source

The gamma dose rate expected for plutonium metal from Content H with composition and impurities given in Table 1.14 bounds the gamma dose rate expected for plutonium oxide from Content I with composition and impurities given in Table 1.15. For the purposes of the gamma source, the beryllium and graphite impurity content for the proposed payload have little effect on the dose rate and so the Content H payload is bounded by Content H with 0.976 g of beryllium. For Content H with 0.976 g of beryllium, the maximum gamma-source strength and spectra are calculated by the RASTA code that calculates the source contribution from all radioactive daughter products. The SARP has input the key parameters required in the input file listings appearing in Appendix 5.8 for Content H with 0.976 g of beryllium. The SARP sums up the limiting values for all allowed radioisotopes for both contents. For Content H with 0.976 g of beryllium, the total gamma source mass used is the optimized neutron source discussed in Section 5.3.2.2 in this SER. The staff agrees that this approximation is acceptable since the gamma dose rates are less than about 5% of the regulatory dose rate limits on the surface or at 1 m from the surface.

The SARP presents a listing of the gamma-source term in gammas per second as a function of energy for Content H with 0.976 g of beryllium. The mass of each nuclide that contributes significantly to the source term is listed in Table 5.2.5-1 for this content. For Content H with 0.976 g of beryllium, the source age where neutron emission rate is a maximum is determined to be at 72.9 years of decay. Confirmatory analyses of the gamma-source terms were satisfactorily conducted using the computer codes ORIGEN-S and GAMGEN; satisfactory agreement achieved with the values in Table 5.2.5.3.

5.3.2.2 Neutron Source

The neutron dose rate expected for plutonium metal from Content H with composition and impurities given in Table 1.14 bounds the neutron dose rate expected for plutonium oxide from Content I with composition and impurities given in Table 1.15. The proposed beryllium content will produce significant neutrons from (α,n) reactions whereas the graphite impurities are not an important source of (α,n) neutrons. For the Content H payload the maximum neutron-source strength and spectra are calculated by the RASTA code, which is an appropriate method for these studies. RASTA considers neutrons from both spontaneous fission and from (α,n) reactions. The SARP used a bounding neutron source that represented the optimum composition of actinide nuclides that give the maximum neutron source corresponding to the 4.4 kg plutonium mass limit. The SARP has input the key parameters required in the input file listings appearing in Appendix 5.8. The SARP sums up the limiting values for all allowed radioisotopes for the 9975 Package with this payload. The neutron emission rate was optimized as a function of decay time and the maximum was found for the bounding plutonium metal source after 72.9 years of decay. The staff accepts that this approximation (1) is conservative when producing maximum neutron-source strength, and (2) bounds the production of neutrons from subcritical multiplication in the fissile material in the payload. The neutron activity coming from light nuclides other than beryllium in the payload is not significant.

The SARP presents a listing of the neutron-source term in neutrons per second as a function of energy for Content H. The mass of each nuclide that contributes significantly to the source term is listed in Table 5.2.5-1 for this content. Confirmatory analyses of the neutron-source terms were conducted using the computer codes ORIGEN-S and SOURCES, and satisfactory agreement was obtained with the values in Table 5.2.6-1.

5.3.3 Shielding Model

The staff concurs that the models used in the shielding calculations are consistent with the effects of the NCT and HAC tests on the 9975 Package.

5.3.3.1 Configuration of Source and Shielding

The dimensions of the source and packaging used in the shielding models correspond to those given in the SARP drawings. The contents are positioned at appropriate locations, considering tolerances, and with appropriate densities that ensure that maximum external radiation levels are calculated. Conservative choices were used for both NCT and HAC package models.

The dose point locations in the shielding model are given at the package surface and 1 m from that surface as prescribed in 10 CFR 71 [§71.47(a)], for NCT non-exclusive use shipments. Also, the dose point locations in the shielding model are given at 1 m from the package surface for HAC as prescribed in 10 CFR 71 [§71.51(a)(2)]. The points chosen give the location of the maximum radiation levels expected from each payload. All voids, streaming paths, and irregular geometries are treated in an adequate manner.

5.3.3.2 Material Properties

Accepted values for the density of all package materials are used in the SARP. All calculations were performed using fissile material in the metal form. Accepted values for the source-material densities are used in the shielding calculations in the SARP. The shielding model considers a spherical source region at maximum theoretical density. The staff considers that such an approach is justified, and that the mass densities used are correct.

The NCT tests demonstrated that there was no significant damage to the package or packaging materials that would significantly affect the shielding of source radiation. The HAC shielding studies assumed that all packaging materials outside the containment system are absent, even though the HAC tests demonstrated that most would survive. This is a conservative assumption. The staff concludes that the shielding properties of the lead layer and the fiberboard insulation and spacer will not degrade during the normal service life of the packaging.

5.3.4 Shielding Evaluation

5.3.4.1 Methods

All dose rates on the 9975 Package for Content H with both 0.976 g of beryllium and 500 g of beryllium were determined using the three-dimensional Monte Carlo transport code MCNP. This is an acceptable code to use for these calculations. The MCNP computer program is referenced properly. The cross sections used in MCNP were taken from the MCNP (ENDF/B-V) libraries.

Secondary gamma production is included in the analyses. Subcritical neutron multiplication is accounted for explicitly in MCNP.

Confirmatory calculations show that streaming paths do not play a significant role in the dose rates determined in this SARP. Although streaming paths could potentially arise in the 9975 Package for HAC conditions, the SARP HAC shielding model excludes all packaging materials outside the SCV. Therefore streaming paths are irrelevant.

5.3.4.2 Input and Output Data

Key input data for the shielding calculations are identified for the computer codes employed. Representative input files used in the analyses are presented in Appendix 5.8. The shielding model input parameters were properly entered into MCNP and RASTA input listings in Appendix 5.8. No output listings are included in the SARP. However, confirmatory calculations generally verify the dose rates listed in the SARP and establish that proper convergence was achieved.

5.3.4.3 Flux-to-Dose-Rate Conversion

The SARP evaluation properly converts the gamma and neutron fluxes to dose rates. The flux-to-dose rate conversion factors (from ANSI 6.1.1-1977) used in the shielding calculation are properly tabulated as a function of the energy group structure in Appendix 5.1.

5.3.4.4 External Radiation Levels

The NCT tests caused no significant damage to the packaging that would alter its shielding effectiveness or its ability to prevent loss or dispersal of radioactive contents. The SARP evaluation properly addresses package damage due to the HAC tests by ignoring all protective packaging outside the containment system. This is conservative since the HAC tests did not cause this much damage.

The external gamma radiation levels were not specifically calculated for Content H with 500 g of beryllium in the 9975 Package. The SARP lists the gamma dose rates for Content H with 0.976 g of beryllium. These gamma dose rates can be considered as acceptable bounds for Content H with 500 g of beryllium, since beryllium does not affect gamma production, and the lack of 499.024 grams of beryllium is replaced by plutonium metal, which is conservative. The calculated values are less than about 5% of the regulatory limits as required by 10 CFR 71 [§71.47(a)] for non-exclusive use shipment of 200 mrem/h at the package surface, and 10 mrem/h at 1 m from the surface.

The gamma dose rates expected for Content H with 500 g of beryllium are less than about 5% of the regulatory dose rate limits on the surface or at 1 m from the surface. The bounding gamma dose rates on the surface are about 10 mrem/hr and at 1 m from the surface are about 0.6 mrem/hr. The SARP considers the gamma dose rates to be subject to the same dose rate correction factors introduced in the SARP because of the potential for very large neutron dose rates. The effect of this assumption is to introduce additional conservatism into the dose rate correction factor methodology.

The SARP calculates the external neutron dose rates for Content H with 500 g of beryllium. The SARP also examines the effect of shape and density of the plutonium metal mass on the neutron dose rates and chose the one that gave the largest dose rate values, which is the solid metal sphere.

The SARP analyses show that the locations selected to determine radiation doses are those that give the maximum dose rates. The external radiation levels for Content H appear to be reasonable and their variations with location are consistent with the geometry and shielding characteristics of the 9975 Package. Uncertainties in composition, form, and location of the contents lead to the need to measure package dose rates in some acceptable manner to try to establish that dose rates will not exceed regulatory limits during transportation. Acceptable composition and form can be established for metal or oxide contents by measurement since conditions during shipment are not sufficient to modify them. However, content movement does need to be accounted for. Also neutron dose rate detection devices have measurement uncertainties that must also be considered. To accommodate these features, the SARP introduces correction factors for dose rate uncertainties due to neutron detector type and for possible source movement. The staff agrees that the IAEA large-detector correction factors given in Table 5.2.9-2 will result in a conservative dose rate adjustment for neutron detector type. The staff agrees that the source movement correction factors given in Table 5.2.9-1 for package top and package bottom will result in a conservative dose rate adjustment for source movement in the vertical direction. However, the staff disagrees that the source movement correction factors given in Table 5.2.9-1 for package side (radial) will result in a conservative dose rate adjustment for source movement in the horizontal direction. Conservative source movement correction factors for package side (radial) should be 1.4 at the surface and 1.1 at 1 meter. The measured dose rates limited by these correction factors for top, bottom, and side of a package, as well as detector type, will ensure that the external radiation levels under NCT and HAC meet the limits prescribed in 10 CFR 71 [§71.47(a) and §71.51(a)(2)]. These bounding values are not listed in the summary tables discussed above in Section 5.3.1 in this SER. The analyses in the SARP demonstrate that any increase in dose rates under NCT will remain below the regulatory limits and hence are not significant, as prescribed in 10 CFR 71 [§71.51(a)(1)].

Confirmatory analyses for gamma dose rates were conducted using the deterministic code MICROSHIELD (version 4.2). Also, calculations of selected neutron and gamma dose rates were confirmed using the Monte Carlo radiation transport code MCNP (version 4b) with the point wise .60c cross section sets (ENDF/B VI), as appropriate. The staff agrees that the 9975 Package meets the requirements prescribed by 10 CFR 71 [§71.43(f), §71.47(a), and §71.51(a)(2)].

5.3.5 Appendix

A list of references was included just before the appendix. The SARP appendix consists of a section giving a summary of the energy group structure for cross-section sets, and the flux-to-dose conversion factors used in the analyses. It also includes sections giving modeling details, material properties, isotopic decay, and methodology used to determine the source terms. The appendix presents the relevant information concerning several computer codes, as well as selected input computer files. A more detailed presentation of the shielding analysis for Content H is also contained in the appendix. In addition, the appendix contains information on uncertainties for several dose rate detection instruments.

5.4 Evaluation Findings

5.4.1 Findings

The 9975 Package design has been shown to meet the shielding requirements of 10 CFR 71 [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]. The 9975 Package has been shown to be designed, constructed, and prepared for shipment so that the external radiation levels will not significantly increase under the tests specified in §71.71 as required by [§71.43(f), §71.51(a)(1)].

The 9975 Package with the plutonium metal payload given by Content H with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.47(a) for non-exclusive-use shipments under the tests specified in §71.71 for NCT. The 9975 Package with Content H with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.51(a)(2) of 1 rem/h at one meter from the surface of the 9975 Package under the tests specified in §71.73 for HAC.

The 9975 Package with the plutonium oxide payload given by Content I with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.47(a) for non-exclusive-use shipments under the tests specified in §71.71 for NCT. The 9975 Package with Content I with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.51(a)(2) of 1 rem/h at one meter from the surface of the 9975 Package under the tests specified in §71.73 for HAC.

Based on review of the statements and representations in the application, the staff concludes that the shielding design has been adequately described and evaluated and that the package meets the external radiation requirements of 10 CFR 71. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

5.4.2 Conditions of Approval

Section 5 of the certificate of compliance must contain the restriction that the 9975 Package must contain a lead shield and cane fiber insulation with the dimensions, density, and composition as specified on the engineering drawings in the SARP. Package surface dose rate limits at the time of shipment given in Section 5 of the CoC must be based on IAEA large-detector correction factors for the dose rate measurement instruments Eberline WENDI-2 and Eberline NRD in Table 5.2.9-2. Section 5 of the CoC specifies that package surface dose rate limits at the time of shipment must be based on the correction factors for source movement given in Table 5.2.9-1, except that a factor of 1.4 must be used for package side (radial) at the surface and a factor of 1.1 must be used for package side (radial) at 1 meter from the surface. This requirement is reflected in Condition 5(d)(5) of the CoC. In addition, the only contents permitted in the 9975 Package are those corresponding to the restrictions given for Content H, plutonium metal, in Table 1.14 of the SARP, or Content I, plutonium oxide, in Table 1.15 of the SARP. The CoC must also contain the restriction that these contents must be doubly contained, i.e., that both PCV and SCV must be used.

5.5 References

"MCNP—A General Monte Carlo N-Particle Transport Code," Version 4B, Judith F. Briesmeister, ed. Los Alamos Report, LA-12625-M, RSICC Computer Code Collection CCC-660, March 1997.

"MICROSHIELD Version 4.2 User's Manual," Grove Engineering, Inc., 15215 Shady Grove Road, Rockville, Maryland, 20850, September 30, 1996.

"SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluations," NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I, II, III, October 1995, RSICC Computer Code Collection CCC-545, SCALE v. 4.3.

American Nuclear Society, "American National Standard for Neutron and Gamma-Ray Flux to Dose Rate Factors," ANSI/ANS 6.1.1-1977, LaGrange Park, Illinois.

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International Atomic Energy Agency (IAEA), Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1985 Edition (as amended 1990), Vienna, Austria (1990).

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Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Compatibility with the International Atomic Energy Agency (IAEA)," 60 FR 50248, September 28, 1995, as amended.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

Wilson, W.B. et al. "SOURCES 4A: A Code for Calculating (α,n) Spontaneous Fission and Delayed Neutron Sources and Spectra," Los Alamos Report, LA-13639-MS, RSICC Computer Code Collection CCC-661, September 1999.

6. CRITICALITY REVIEW

Chapter 6, Criticality, of the Safety Analysis Report —Packages (SARP) for the 9975 Package was reviewed for criticality safety requirements.

6.1 Areas of Review

The criticality review included the following:

6.1.1 Description of Criticality Design

- Design Features
- Codes and Standards
- Summary Table of Criticality Evaluations

6.1.2 Fissile Material Contents

6.1.3 General Considerations for Criticality Evaluations

- Model Configuration
- Material Properties
- Demonstration of Maximum Reactivity
- Computer Codes and Cross-Section Libraries

6.1.4 Single Package Evaluation

- Configuration
- Results

6.1.5 Evaluation of Undamaged-Package Arrays (Normal Conditions of Transport)

- Configuration
- Results

6.1.6 Evaluation of Damaged-Package Arrays (Hypothetical Accident Conditions)

- Configuration
- Results

6.1.7 Transport Index for Nuclear Criticality Control

6.1.8 Benchmark Evaluations

- Applicability of Benchmark Experiments
- Bias Determination

6.1.9 Appendix

6.2 Regulatory Requirements

Regulatory requirements of 10 CFR 71 applicable to the criticality review of fissile material packages are as follows:

- The package design must be described and evaluated to demonstrate that it meets the criticality requirements of 10 CFR 71. [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]
- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Unknown properties of fissile material must be assumed to be those which will credibly result in the highest neutron multiplication. [§71.83]
- A single package must be subcritical under the conditions of §71.55(b), §71.55(d), and §71.55(e).
- The package must be designed, constructed, and prepared for shipment so that there will be no significant reduction in the effectiveness of the packaging under the tests specified in §71.71 for NCT. [§71.43(f), §71.51(a)(1), §71.55(d)(4)]
- An array of undamaged packages must be subcritical under the conditions of §71.59(a)(1).
- An array of damaged packages must be subcritical under the conditions of §71.59(a)(2).
- A fissile material package must be assigned a transport index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59, §71.35(b)]

6.3 Review Procedures

Chapter 6 of the 9975 SARP includes the information essential for a criticality evaluation including the drawings, the packaging materials and densities, and the fissile isotopic composition and mass. This criticality information in the 9975 SARP was reviewed by the staff for completeness and compliance with regulatory requirements. Of particular importance is the response of the containment vessel and the contents to the imposed NCT [10 CFR §71.71] and the HAC [10 CFR §71.73].

6.3.1 Description of Criticality Design

6.3.1.1 Design Features

The 9975 is the only package design that is addressed in the review. The 9975 Package has double containment. The fissile contents are generally placed in a product or convenience can. A product or convenience can may be placed in one or more LDPE bags, provided that no more than 100 grams of polyethylene are involved. For metals, from one to three product cans may be placed within the PCV. Oxides must be enclosed in a 3013 container. Therefore, if oxide is in a convenience can it must be placed in a 3013 container, which in turn is placed within the PCV.

The design of the 9975 Package does not include any specific neutron-absorbing material for criticality control. The package utilizes the geometry of the containment vessel and control of the quantity and composition of the fissile material to ensure that the single package contents are subcritical under NCT and HAC. In addition to the control of the geometry and specific fissile content, interaction control is also established by the fact that each package is enclosed in a drum structure ensuring a center-to-center separation of at least the diameter of the drum in the lateral direction (perpendicular to the drum axis). Furthermore, the hydrocarbon insulating-spacing material (with a nominal minimum density of 0.24 g/cc) is a neutron moderator and acts to further isolate a package from neighboring packages. These features ensure that the arrays of packages are subcritical under NCT and HAC.

The staff confirms that the text and sketches describing the criticality design features are consistent with the engineering drawings and the models used in the criticality evaluation. The staff also concludes that the 9975 Package conforms to the general standards for all packages as prescribed by 10 CFR 71 [§71.31(a)(1), §71.31(a)(2), §71.31(c), §71.33, §71.35(a)]. In addition, the staff concludes that the SARP has assigned a proper TI of 2.0 for the 9975 Package with metal or oxide payloads as prescribed by 10 CFR 71 [§71.59, §71.35(b)].

The PCV for the 9975 Package consists of a cylindrical structure with a maximum 5.174-inch ID, Type 304L stainless steel pressure vessel. The SCV for the 9975 Package consists of a maximum 6.345-inch ID, Type 304L stainless steel cylindrical pressure vessel. Both the PCV and the SCV comply with the stress criteria of the ASME B&PV Code Section III, Subsection NB. The PCV is placed within the SCV. The PCV-SCV combination is placed within a specially fabricated 35-gallon removable-head drum constructed of Type 304L stainless steel with a minimum OD of 18.22 inches (the drum rolling hoops are somewhat larger and are responsible for a slightly larger minimum diameter). The PCV-SCV combination is enclosed within a 0.5-inch-thick layer of lead which is kept centered within the drum by about 11 inches of fiberboard insulation material.

6.3.1.2 Codes and Standards

The containment vessels are leak tested to the ANSI N14.5-1987 standard.

The 9975 Package containment vessel design for the PCV and the SCV complies with the stress criteria of the ASME B&PV Code Section III, Subsection NB.

Single package subcriticality is based on use of ANSI 8.1-1988.

6.3.1.3 Summary Table of Criticality Evaluation

The SARP summary table, Table 6.1, addresses the following cases for the 9975 Package: a single package, under the conditions of §71.55(b), (d), and (e); an array of undamaged packages, under the conditions of §71.59(a)(1); and an array of damaged packages, under the conditions of §71.59(a)(2). Table 6.1 includes the maximum value of the effective multiplication factor (k_{eff}) for each package payload, including two standard deviations. It also lists the safe value for the multiplication factor (k_{safe}) for which the appropriate uncertainty and bias have been subtracted from 0.95 (which includes the accepted criticality safety margin of 0.05). It also lists the number of packages evaluated in the arrays. The table either demonstrates appropriate subcriticality by showing that the value of k_{eff} is less than k_{safe} for that package and payload, or else it invokes the ANSI 8.1-1988 subcritical limit to show sufficient subcriticality.

6.3.2 Fissile Material Contents

The contents used in the criticality analyses are consistent with those specified in the General Information section of the SARP. There are two contents being reviewed: up to 4.4 kg of plutonium metal or 4.4 kg of plutonium in 5.0 kg of oxide. The only allowed fissile material for the 9975 Package is dry metals containing ^{239}Pu; namely, Content E or Content H or dry oxides containing ^{239}Pu; namely, Contents A, B, C or I. The density for any allowed fissile material is its maximum theoretical density. Full enrichment is allowed for ^{239}Pu.

6.3.3 General Considerations for Criticality Evaluations

6.3.3.1 Model Configuration

The configurations for the calculational models for a single package and for the arrays of packages used to perform the criticality evaluation of the 9975 Package are described in Section 6.2.3 of the SARP.

The criticality modeling for the 9975 Package makes several assumptions for the package models to be used for a single package. The SARP presents different package models for the NCT and HAC array analyses.

The model for the single 9975 Package assumes that the PCV is a simple cylinder. The maximum inner cylinder diameter is chosen for the PCV, as this choice maximizes the PCV volume and the reactivity. The calculational model assumes full water reflection of the PCV, as required by 10 CFR 71, [§71.55(b)].

For the single package analyses, the fissile materials are treated as being spherical metal with a beryllium shell and surrounded by water. Plutonium metal bounds plutonium oxide from a criticality standpoint, independent of whether the beryllium shell is present. Also the possible LDPE bags surrounding the fissile material in metal form are considered by allowing a 100-gram shell of CH_2 to surround the fissile sphere. All three of these treatments maximize the reactivity.

The NCT tests did not cause any damage to the 9975 Package that significantly affected criticality. Analyses reported in the SARP show that an infinite number of undamaged packages remain subcritical, whereas only 125 undamaged packages would need to remain subcritical to give a TI for criticality equal to 2.0. This is the TI assigned to the package.

The HAC tests did cause package damage that affected the criticality calculations. The fire test charred the Celotex insulation by a maximum of 2.5 inches. The calculational model treats this fire damage by replacing the outer 3 radial inches and 3.75 axial inches of Celotex by air. The drop test crushed the Celotex insulation in the radial direction by a maximum of 1.5 inches. The calculational model treats crush damage by displacing the PCV by 4.5 radial inches and 6 axial inches within the Celotex. The displacements of the PCV in neighboring packages in an array are treated to maximize their interaction and produce maximum reactivity. This is a very conservative treatment of the HAC damage.

HAC array sensitivity calculations demonstrated that the most reactive configuration resulted when the damaged portion of the removed Celotex within the drum was replaced by air and not by water of any density.

For the HAC array calculations, the fissile materials are located within the PCV to give the closest interaction with respect to the fissile materials in other neighboring packages. This treatment maximizes the reactivity.

The closest packed array of 9975 Packages achievable is hexagonal in a lateral plane (perpendicular to the package axes), but square in the vertical direction for subsequent layers of packages. This is because the packages in layers above or below cannot be physically nested into the layer in question because they have a square vertical areal cross section. The SARP analyses used square arrays in both directions, but decreased the lateral pitch by 7% to account for this approximation in the lateral-plane layers.

Because the 9975 Package has double containment and no inleakage occurred during HAC tests, the HAC array calculation model assumes that the PCV is dry. For single package calculations, the fissile materials are treated as spherical metal with a beryllium shell and surrounded by water. For the NCT and HAC calculations, the fissile material is assumed to be a dry metal sphere with a beryllium shell.

6.3.3.2 Material Properties

Accepted values for the density of all packaging materials are used in the SARP. The SARP used a density value for the fiberboard material of 0.20 g/cc that is somewhat less than the nominal minimum density specified of 0.24 g/cc. (The minimum permissible density of fiberboard is greater than 0.20 g/cc.) This lower Celotex density is a conservative assumption for criticality analyses. All calculations were performed using fissile material having a metal form. Accepted maximum values for the fissile material densities are used in the SARP. The staff concludes that the fissile material properties for the 9975 Package conform to 10 CFR 71.83. In addition, the staff concludes that the properties of the fiberboard insulation-spacer affecting criticality will not degrade during the normal service life of the packaging.

6.3.3.3 Demonstration of Maximum Reactivity

Maximum reactivity was demonstrated for single packages with fissile material with an optimum thickness shell of beryllium with no graphite present (see Section 6.3.4 in this SER). An optimum thickness shell corresponds to about 200 to 300 grams of beryllium. LDPE bags surrounding metal fissile material are treated as a 100 gram shell of CH_2. Analyses of the configuration with the polyethylene shell give slightly more reactivity than without it. Confirmatory calculations verify these conclusions.

The most reactive individual package appropriate to the specific conditions was used for NCT and HAC array analyses. Maximum reactivity was demonstrated for both NCT and HAC array analyses for the mass and position of fissile material, and internal and interspersed moderation (see Sections 6.3.5 and 6.3.6, respectively, in this SER). Confirmatory calculations verify this conclusion.

The SARP analyzed the effect of surrounding the PCV with various reflective regions on its reactivity. The effect of alloying plutonium metal with 5 wt% gallium on the reactivity was also considered. The metal RFETS-3013 configuration was found to be the most reactive configuration for the single package and was used to bound the other packaging options. The SARP analyzed the effect of various combinations of flooding and reflection of the PCV in determining the most reactive configuration. The staff confirms that the SARP has used the most reactive configuration in determining the radiation levels.

6.3.3.4 Computer Codes and Cross-Section Libraries

The older parametric criticality studies of the 9975 Package were performed to find the most reactive configuration, using the Monte Carlo criticality code CSAS25, and a module of SCALE 4.2 that invokes the criticality module, KENO VA. The cross sections used were taken from the SCALE 44-group (ENDF/B V) library. The newer criticality studies used the 238-group cross-section library with CSAS25 in SCALE 4.3. These computer codes and cross-section libraries are appropriate for the criticality calculations and are consistent with the neutron spectrum of the package. Also these cross-section libraries properly account for resonance absorption and self-shielding effects. The benchmark evaluations and resulting biases were determined using the same codes and cross-section sets.

The SARP study used between 90,000 and about 400,000 neutron histories to obtain the k_{eff} values. The number of neutron histories is adequate to assure that the fissile systems analyzed will be sampled in a statistically acceptable manner.

No output listings are included in the SARP, but confirmatory calculations verify the criticality multiplication factors. The model input parameters, material densities, and cross sections were properly entered into the CSAS25 input listings in Appendix 6.2.

6.3.4 Single Package Evaluation

The staff concludes that the 9975 Package conforms to the criticality requirements as prescribed by 10 CFR 71, [§71.43(f), §71.51(a), §71.55(b), §71.55(d), §71.55(e)].

6.3.4.1 Configuration

The SARP determined that the maximum reactivity occurs when the PCV in the 9975 Package contains a solid 4.4 kg sphere of ^{239}Pu metal with a tight fitting shell of beryllium of optimum thickness (4.4 kg includes both plutonium and beryllium) with both completely surrounded by water (fully flooded), and with full water reflection of the containment vessel, as required in §71.55(b).

6.3.4.2 Results

The 9975 Package also meets the additional specifications of 10 CFR 71 [§71.55(d)(2) through §71.55(d)(4)] under NCT.

The criticality results of the most reactive case for the single package analysis are consistent with the information presented in the summary table discussed in Section 6.3.1 of this SER.

ANSI-8.1-1988 gives 5.0 kg of ^{239}Pu metal as the subcritical limit. The SARP argues that a single 9975 Package with a solid 4.4 kg sphere of ^{239}Pu metal is subcritical because it is 600 grams less than the ANSI-8.1 subcritical limit and that the packaging surrounding the PCV (lead, fiberboard, drum, etc.) is essentially statistically equivalent to water. The SARP shows that 600 grams of ^{239}Pu metal accounts for not less than approximately 2.9% of the package reactivity. The maximum additional reactivity effect of a beryllium shell (including reduced Pu mass) is found to be about 1%. Therefore, the surrounding beryllium reflector material increases k_{eff} much less than 600 grams of plutonium decreases k_{eff}. Mixing the beryllium homogeneously with the fissile material decreases k_{eff}. Therefore, 4.4 kg of ^{239}Pu metal in any configuration in a full water-flooded PCV and fully water-reflected containment vessel is appropriately subcritical. The staff concurs with this assessment. This metal content bounds 4.4 kg of plutonium in 5.0 kg of plutonium oxide, independent of whether the beryllium shell is present.

Confirmatory analyses were conducted using the criticality code MCNP (version 4a) with the point wise .60c cross-section sets (ENDF/B VI) where possible. Also, selected calculations were confirmed using the CSAS25 module of SCALE 4.3, with SCALE 44-group (ENDF/B V) cross sections. Confirmatory analyses verify that the SARP conclusions are valid.

6.3.5 Evaluation of Undamaged-Package Arrays (Normal Conditions of Transport)

The NCT tests did not result in any water leakage into the containment system or damage that significantly affected the criticality of the packages. The staff concludes that the 9975 Package is designed, constructed, and prepared for shipment so that there will be no significant reduction in the criticality safety of any package during NCT. The staff also concludes that the 9975 Package conforms to the NCT criticality requirements for all packages as prescribed by 10 CFR 71, [§71.59(a)(1), §71.59(a)(3)].

6.3.5.1 Configuration

The SARP evaluated the most reactive dry fissile contents in an undamaged 9975 Package for the NCT analyses. The most reactive dry fissile content was a solid 4.4 kg sphere of ^{239}Pu metal with an optimum thickness shell of beryllium (4.4 kg includes both plutonium and beryllium) in a PCV. No water is present within the containment vessel and there is no interspersed moderation between packages. The plutonium sphere with a beryllium shell is located within the center of each PCV. The SARP analyses evaluated an infinite array of packages to demonstrate subcriticality.

6.3.5.2 Results

The most reactive dry individual 9975 Package was used for the NCT analyses. No containment flooding or interspersed moderation is required for these NCT studies. The array analyses reported in the SARP showed that an infinite array of packages, with each fissile mass located at the center of the PCV in each package, is appropriately subcritical. A TI of 0.0 would result for the 9975 Package based on these NCT analyses.

Confirmatory analyses were conducted using the criticality code MCNP (version 4a) with the point wise .60c cross-section sets (ENDF/B VI) where possible. Also, selected calculations were confirmed using the CSAS25 module of SCALE 4.3, with SCALE 44-group (ENDF/B V) cross sections. Confirmatory calculations used the actual hexagonal lattice packing for the lateral layers in order to confirm that the SARP results are acceptable. Confirmatory analyses verify that the SARP conclusions are valid.

6.3.6 Evaluation of Damaged-Package Arrays (Hypothetical Accident Conditions)

The staff concludes that the 9975 Package conforms to the HAC criticality requirements for all packages as prescribed by 10 CFR 71, [§71.59(a)(2), §71.59(a)(3)].

6.3.6.1 Configuration

The SARP uses the most reactive contents in a damaged 9975 Package for the array calculations under HAC analyses. Since the 9975 has double containment and did not leak during HAC tests, and because the 9975 containment vessel design for the PCV and the SCV complies with the stress criteria of the ASME B&PV Code Section III, Subsection NB, the PCV is assumed to not leak water. Therefore the contents are assumed to remain dry. The most reactive fissile content is a solid 4.4 kg sphere of ^{239}Pu metal with an optimum thickness beryllium shell (4.4 kg includes both plutonium and beryllium) within a PCV.

The most reactive configuration of packages in the HAC calculations is with no interspersed moderation between packages. The plutonium sphere with beryllium shell is located within each PCV so that the closest interaction exists between fissile masses in neighboring packages. In the damaged condition, the PCV and Celotex material, modified as described in Section 6.3.3.1 in this SER, should also be displaced within the packages to give rise to the maximum interaction between neighboring packages. That is, the bottom level packages have the plutonium sphere with beryllium shell near the top of the PCV and moved laterally toward the PCV sidewall nearest the vertical axis through the packages. Each PCV-SCV assembly is then moved vertically near the top of the package and moved laterally toward the vertical axis through the center of the eight packages as much as allowed by the damaged condition of the insulation material, as given in Section 6.3.3.1 in this SER. Whereas, the top level packages have the plutonium sphere with beryllium shell near the bottom of the PCV and moved laterally toward the PCV side wall nearest the vertical axis through the packages. Each PCV-SCV assembly is then moved vertically near the bottom of the package and moved laterally toward the vertical axis through the center of the eight packages as much as allowed by the damaged condition of the insulation material, as given in Section 6.3.3.1 in this SER. The SARP evaluates a $5 \times 5 \times 2$ array to demonstrate subcriticality.

6.3.6.2 Results

There is no evidence in the SARP text in Chapter 6 or in the input files in the Chapter 6 appendices to indicate that full water reflection of the arrays was considered in the analyses. However, confirmatory calculations show that the effect of full water reflection of the arrays on the multiplication factor is not statistically significant.

The most reactive single 9975 Package with appropriate damage was used for the HAC, except without water flooding in the PCV. For the 9975, this configuration is described in the preceding section. The array analyses performed assumed the plutonium sphere with beryllium shell was located within each PCV, and each damaged PCV-Celotex combination is displaced so that the closest separation exists between fissile masses in neighboring packages. This results when the plutonium spheres in each set of eight neighboring packages (4-in. top layer and 4-in. bottom layer immediately below them) are at their closest possible approach. This arrangement gives the maximum interaction between neighboring packages. The most reactive array is, in addition, when no interspersed moderation is present between packages. This is a very conservative model. The SARP analyses find that a 5 × 5 × 2 array of HAC packages is appropriately subcritical. Confirmatory calculations support this conclusion. A TI of 2.0 is determined for 50 packages being subcritical for HAC.

Confirmatory analyses were conducted using the criticality code MCNP (version 4a) with the point wise .60c cross-section sets (ENDF/B VI) where possible. Also, selected calculations were confirmed using the CSAS25 module of SCALE 4.3, with SCALE 44-group (ENDF/B V) cross sections. Confirmatory calculations used the actual hexagonal lattice packing for the lateral layers in order to confirm that the SARP results are acceptable. Confirmatory analyses verify that the SARP conclusions are valid.

6.3.7 Transport Index for Nuclear Criticality Control

A minimum criticality TI of 2.0 is assigned to the 9975 Packages based on the HAC array calculations showing that 50 packages in any configuration have a multiplication factor plus bias and uncertainties that is less than 0.95. The TI is consistent with that reported in Chapter 1 on General Information in the SARP. The staff concurs with this value.

6.3.8 Benchmark Evaluations

The SARP used the same criticality computer code, hardware, and cross-section library sets to determine the bias values from benchmark experiments as those used to calculate the multiplication factors for the packages. Additional benchmark information is given in Appendix 6.1.

6.3.8.1 Applicability of Benchmark Experiments

The benchmark experiments used in this study were taken from the various volumes of the "International Handbook of Evaluated Criticality Safety Benchmark Experiments" (NEA 1998), and are appropriately referenced. The fissile systems considered in the SARP for this certification were for plutonium metal. This collection of benchmark experiments is the accepted standard in the criticality community.

The benchmark experiments are applicable to the actual packaging design and contents. The benchmark experiments have, to the maximum extent possible, the same fissile materials, moderation, neutron spectra, and configuration as the package evaluations.

6.3.8.2 Bias Determination

Contributions from uncertainties in experimental data are included for all benchmark experiments reported in the Handbook. Also, a sufficient number of appropriate benchmark experiments are analyzed and the results of these benchmark calculations are used to determine an acceptable bias for each fissile payload. These bias values are then used in the calculation of a safe multiplication factor for the package payloads. The statistical and convergence uncertainties of the benchmark calculations and package evaluations are essentially consistent and do not significantly affect the determination of bias values.

The SARP determined an acceptable value for the bias for plutonium metal. Acceptable statistical analyses demonstrate that this value is accurate, but conservative. The staff concurs that the benchmark experiments and corresponding bias value are applicable and conservative as applied to the 9975 Package.

6.3.9 Appendix

The appendix consists of a summary of the critical benchmark experiments and bias determination, a selection of CSAS25 input files, a comparison of the criticality aspects of triangular and square arrays of 9975 Packages, and the HAC array configurations. The staff received separate copies of the reports referenced in the SARP.

6.4 Evaluation Findings

6.4.1 Findings

Based on review of the statements and representations made in the SARP, the 9975 Package design has been shown to meet the criticality requirements of 10 CFR 71 [§71.31(a)(1), §71.31(a)(2), §71.33, §71.35(a)]. The 9975 Package has been shown to be designed, constructed, and prepared for shipment so that there will be no significant reduction in the effectiveness of the packaging under the tests specified in §71.71 for NCT [§71.43(f), §71.51(a)(1), §71.55(d)(4)].

The 9975 Package with Content E or Content H for plutonium metal with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.55(b), §71.55(d), and §71.55(e) under which a single package must be subcritical. The 9975 Package with Content E or Content H for plutonium metal with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.59(a)(1) and §71.59(a)(2) under which an array of undamaged packages and an array of damaged packages must be subcritical, respectively.

The 9975 Package with Contents A, B, C or I for plutonium oxide with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.55(b), §71.55(d), and §71.55(e) under which a single package must be subcritical. The 9975 Package with Contents A, B, C or I for plutonium oxide with 500 g of beryllium and/or 1.0 kg of graphite has been shown to meet the requirements of §71.59(a)(1) and §71.59(a)(2) under which an array of undamaged packages and an array of damaged packages must be subcritical, respectively.

Based on review of the statements and representations in the application, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated and that the 9975 Package meets the subcriticality requirements of 10 CFR 71. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

6.4.2 Conditions of Approval

The certificate of compliance must contain the restriction that the 9975 Package contain a lead shield and cane fiber insulation with the dimensions, density, and composition as specified on the engineering drawings in the SARP. In addition, the only contents that may be permitted in the 9975 Package are those corresponding to the restrictions given for Content E or Content H, plutonium metal, in Table 1.10 or 1.14 of the SARP, respectively, or Contents A, B, C or I, plutonium oxide, in tables 1.8, 1.7, 1.6 or 1.15 of the SARP, respectively. The current revision of the CoC restricts the authorized contents to Tables 1.14 and 1.15. The CoC must also contain the restriction that these contents be doubly contained, i.e., that both PCV and SCV must be used.

6.5 References

"MCNP—A General Monte Carlo N-Particle Transport Code," Version 4B, Judith F. Briesmeister, ed. Los Alamos Report, LA-12625-M, RSICC Computer Code Collection CCC-660, March 1997.

"SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluations," NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R4), Vols. I, II, III, October 1995, RSICC Computer Code Collection CCC-545, SCALE v. 4.3.

"SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluations," NUREG/CR-0200, Rev. 4 (ORNL/NUREG/CSD-2/R3), Vols. I, II, III, April 1994, RSICC Computer Code Collection CCC-545, SCALE v. 4.2.

American Nuclear Society, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," ANSI/ANS 8.1-1998, revision of ANSI/ANS 8.1-1983, R1988, LaGrange Park, Illinois.

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," United Engineering Center, 345 East 47th Street, New York, NY, 1992.

CSAS25. A functional criticality control module within the SCALE Code System. *See SCALE v. 4.2 or 4.3*.

Institute of Nuclear Materials Management, American National Standard for Radioactive Materials, "Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York.

International Atomic Energy Agency (IAEA), Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1985 Edition (as amended 1990), Vienna, Austria (1990).

Keno Va. A three-dimensional criticality module within the SCALE Code System. *See SCALE v.4.2 or 4.3*.

Nuclear Energy Agency, Organization for Economic Co-Operation and Development, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC Doc (95) 03, September 1998.

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Compatibility with the International Atomic Energy Agency (IAEA)," 60 FR 50248, September 28, 1995, as amended.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

7. OPERATING PROCEDURES

Chapter 7, Operating Procedures, of the Safety Analysis Report —Packages (SARP) for the 9975 Package was reviewed to verify that it (1) meets the requirements of 10 CFR 71, and (2) is adequate to assure that the package will be operated in a manner consistent with its evaluation for approval. These are the generic operating procedures from which the formal, site-specific operating procedures will be developed.

7.1 Areas of Review

The staff reviewed the controls and procedures to ensure that the 9975 Package will be operated in a manner consistent with its evaluation for approval. The Operating Procedures review included the following:

7.1.1 Package Loading

- Preparation for Loading
- Loading of Contents
- Preparation for Transport

7.1.2 Package Unloading

- Receipt of Package from Carrier
- Removal of Contents

7.1.3 Other Procedures

7.1.4 Preparation of Empty Package for Transport

7.1.5 Appendices

7.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Operating Procedures review are as follows:

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The application must include any special controls and precautions for transport, loading, unloading, and handling of a fissile material shipment, and any special controls in case of accident or delay. [§71.35(c)]
- A package must be conspicuously and durably marked with the model number, serial number, gross weight, and package identification number. [§71.85(c), §71.13(a), §71.13(b)]
- The application must include operating procedures that ensure that the package meets the routine-determination requirements of §71.87. [§71.81, 71.87]
- Unknown properties of fissile material must be assumed to be those which will credibly result in the highest neutron multiplication. [§71.83]
- Packages that require exclusive-use shipment, because of increased radiation levels, must be controlled by providing written instructions to the carrier. [§71.47(b-d)]

- The transport index of a package in a nonexclusive-use shipment must not exceed 10, and the sum of the transport indices of all packages in the shipment must not exceed 50. [§71.47(a), §71.59(c)(1)]
- The sum of the transport indices for nuclear criticality control of all packages in an exclusive-use shipment must not exceed 100. [§71.59(c)(2)]
- Prior to delivery of a package to a carrier, any special instructions needed to safely open the package must be provided to the consignee for the consignee's use in accordance with 10 CFR 20.1906(e). [§71.89]

7.3 Review Procedures

The following procedures are generally applicable to the review of the Operating Procedures chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 7.1 of this SER.

The operating procedures in the SARP should generally be listed in sequential order. Additional guidance on operating procedures is provided in the "Guide for Preparing Operating Procedures for Shipping Packages" (NUREG/CR-4775).

7.3.1 Package Loading

7.3.1.1 Preparation for Loading

The procedures for loading the package are contained in Section 7.1 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- It was noted that the package will be loaded and closed in accordance with site-specific, written procedures.
- Special controls and precautions for loading and handling were noted and described.
- A requirement to verify that the package is in unimpaired physical condition, and that all required periodic maintenance requirements have been performed, is included.
- A specific requirement to ensure that the package is conspicuously and durably marked with the model number, serial number, gross weight, and package identification number is not included in the procedures. It is, however, included indirectly in the drawings for the 9975 Packaging, which were included in Appendix 1.1 of the SARP. It is also included indirectly in the Acceptance Test requirements specified in Section III of Appendix 8.1.
- A requirement is included to verify that the package is appropriate for the contents to be shipped.
- A requirement is included to ensure that the use of the package complies with all other conditions of approval in the CoC.

7.3.1.2 Loading of Contents

The procedures for loading the contents into the package are contained in Sections 7.1.2 and 7.1.3 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- Special handling equipment was specified, where needed.
- Special controls and precautions for loading were specified, where needed.
- The method of loading the contents was specified.
- Although there is no requirement to ensure that moderator or neutron absorbers are present and in proper condition, such a requirement is not necessary for the shipment of high-purity plutonium metals and/or alloys. Such a requirement is also not necessary for the shipment of plutonium oxides, as long as the oxides in question have been properly stabilized. There is, however, a requirement to ensure that physical spacers are in place to minimize potential cell sizes and mitigate the potential for a deflagration-to-detonation transition for the shipment of oxides.
- Although there is no description of the method used to remove water from the package, such a requirement is not necessary for this package.
- Because the package is loaded at ambient pressures, there is no requirement to vent excess gases during the loading of the PCV for the shipment of high-purity plutonium metals and/or alloys. There is, however, a requirement to inert the PCV, to minimize the potential for the build-up of flammable gas mixtures in the SCV, should hydrogen gas leak from the PCV to the SCV. This requirement is a requirement for the shipment of oxides, only.
- Specific requirements are in place to ensure that the closure devices of the package, including seals and gaskets, are properly installed, secured, and free of defects.
- A specific requirement is in place which notes that the bolts are to be torqued to 30 ±2 ft-lbs. Although it is noted that no specific tightening sequence is required, it is also noted that each bolt must be re-tightened to confirm that none of the bolts were omitted from the initial tightening sequence.
- Based on the procedures provided, it has been determined that the contents will be loaded correctly, and that the package will be closed appropriately.

7.3.1.3 Preparation for Transport

The procedures for preparation for transport are contained in Section 7.1.3 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- Procedures are in place to ensure that the non-fixed (removable) radioactive contamination on the external surface of the package is as low as reasonably achievable, and within the limits specified in 49 CFR 173.443. Procedures are also in place to ensure that the non-fixed (removable) radioactive contamination on the external surface of the package is within the limits specified in Appendix D of 10 CFR 835. (The requirements specified in Appendix D of 10 CFR 835 are about two orders of magnitude more conservative than those specified in 49 CFR 173.443.).
- Procedures are in place to ensure that the pre-shipment radiation surveys confirm that the allowable external radiation levels are as specified in §71.47, and that they are not exceeded.
- Although there are no specific temperature surveys required to verify that limits specified in §71.43(g) are not exceeded, such a requirement is not necessary for this package.

- Specifications are in place to require that the assembly verification leakage rate tests are performed, and to ensure that the package closures are leakage rate tested in accordance with ANSI N14.5.
- Although there are no requirements to ensure that any system for containing liquid is properly sealed and that it has adequate space or other specified provision for expansion of the liquid, such requirements are not necessary for this package.
- Although there are no requirements to verify that any pressure relief devices are operable and set, the design of the packaging does not incorporate the use of pressure relief devices.
- Although there are no requirements to ensure that any structural components that could be used for lifting or tie-down during transport are rendered inoperable for those purposes unless it meets the design requirements of §71.45, the design of the packaging does not incorporate the use of lifting or tie-down structures.
- A specific requirement is in place to ensure that the tamper-indicating device has been installed.
- Although there are no specific requirements to ensure that impact limiters, personnel barriers, or similar devices have been properly installed or attached, the design of the packaging does not incorporate the use of such features.
- Although there are no requirements that describe, for fissile material shipments, any special controls and precautions for transport, loading, unloading, and handling and any appropriate actions in case of an accident or delay which should be provided to the carrier or consignee, all such requirements are provided indirectly by the inclusion of the DOE/AL Transportation Safeguards Division (TSD) procedures for the use of SSTs and/or Safe-Guard Trailers (SGTs).
- Although there are no specific requirements that identify any special controls which should be provided to the carrier for a package shipped by exclusive use under the provisions of §71.47(b)(1), all such requirements are provided indirectly by the inclusion of the DOE/AL TSD procedures for the use of the SSTs/SGTs.
- Although there are no specific requirements that identify any special controls which should be provided to the carrier for a fissile-material package in accordance with §71.35(c), all such requirements are provided indirectly by the inclusion of the DOE/AL TSD procedures for the use of the SSTs/SGTs.
- There are no specific requirements that describe any special instructions that should be provided to the consignee for opening the package in Section 7.1 of the SARP. These procedures are provided in Section 7.2 of the SARP, and in Section 7.3.2 of this SER.
- Although there is a specific requirement to ensure that a criticality transport index of 2.0 has been noted on the labels for each package, procedures for determining the sum of the criticality transport indexes for the shipment are provided indirectly by the inclusion of the DOE/AL TSD procedures for the use of the SSTs/SGTs. For the shipment of high-purity, plutonium metals and alloys, and for the shipment of oxides, specific procedures are also in place to confirm that the allowable external radiation levels specified in §71.47 are not exceeded.

7.3.2 Package Unloading

7.3.2.1 Receipt of Package from Carrier

The procedures for receipt of the package from the carrier are contained in Section 7.2 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- Specific procedures are in place to ensure that the package is examined for visible damage, status of the tamper-indicating device, surface contamination, and external radiation levels.
- Specific procedures are in place that describe any special actions to be taken if the package is damaged, if the tamper-indicating device is not intact, or if surface contamination or radiation survey levels are too high.
- Although there are no specific requirements that identify any special handling equipment needed, all such requirements are provided indirectly by the inclusion of the DOE/AL TSD procedures for the use of the SSTs/SGTs.
- Specific procedures are in place, which describe any proposed special controls and precautions for handling and unloading.

7.3.2.2 Removal of Contents

The procedures for removal of contents are contained in Section 7.2 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- Specific procedures are in place which describe the appropriate method to open the package.
- Specific procedures are in place which identify the appropriate method to remove the contents.
- Specific procedures are in place which ensure that the contents are completely removed.

7.3.3 Additional Procedures

The Operating Procedures of the SARP adequately describe the procedures to be used for the shipment of high-purity, plutonium metals and alloys. Additional procedures have also been identified for the shipment of oxides.

7.3.4 Preparation of Empty Package for Transport

The procedures for the preparation of an empty packaging for transport are contained in Section 7.3 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- An indirect requirement is specified to verify that the package is empty (see below).
- Specific procedures are in place to ensure that the external surface contamination levels meet the requirements specified in Appendix D of 10 CFR 835. (As was noted previously, the requirements specified in Appendix D of 10 CFR 835 are at least two orders of magnitude more conservative than the corresponding limits specified in 49 CFR 173.443.) Specific procedures are also in place to ensure that an empty package that is internally contaminated should be prepared for shipment as specified in 49 CFR 173.421 or 49 CFR 173.428, depending on the level of residual contamination.
- There are no specific descriptions of the packaging closure requirements.
- An additional requirement is in place to note that, if the package is to be shipped as an Empty Radioactive Materials Packaging per 49 CFR 173.428, the labels and the nameplate are to be covered with tape and the package will be marked empty.

7.3.5 Appendices

There are no appendices for Section 7 of the 9975 Packaging SARP.

7.4 Evaluation Findings

7.4.1 Findings

The operating procedures presented in the SARP for the 9975 Package were reviewed by the staff for completeness and compliance with the regulatory requirements. The information provided by the applicant was in the format prescribed directly by NRC Regulatory Guide 7.9. The information in Section 7 of the SARP was not provided in the format outlined in NUREG/CR-4775. However, the applicable information on operating requirements, general information, package loading, shipment preparation, package receipt, and package unloading was provided in the Operating Procedures chapter of the SARP, in the appropriate level of detail. Supplemental information on inspection and maintenance, and on records and reporting requirements, has also been provided in the appropriate level of detail in Chapters 8 and 9 of the SARP, respectively.

Although the specific requirements noted above for fissile class materials (i.e., those specified in 10 CFR 71.35) were not included in Chapter 7 of the SARP, the requested authorized contents, in this case, are high-purity, plutonium metals and/or alloys, as defined in Table 1.14 of the SARP, and plutonium oxide, as defined in Table 1.15 of the SARP. Considered by the DOE to be Special Nuclear Materials (SNM), the requirements specified in the orders DOE O 474.1A, DOE 5610.14, and DOE AL SD 5610.14, and their supplements are also applicable. Specifically, these orders are applicable to the nuclear materials accountability aspects, and to the transport of SNM.

Of particular importance to this SER are the requirements specified in DOE AL SD 5610.14, which state that any form of plutonium, in quantities of 2 kg or more, shall be transported by TSD. For the shipment of the materials requested, therefore, all shipments must be made in SSTs and/or SGTs. In addition, all shipments must also be made in accordance with the detailed operating procedures for SST/SGT shipments, as delineated in the appropriate DOE/AL TSD documents. Controls to be implemented when 9975 Package shipments are made in an SST/SGT are listed in the SARP, Section 7.1.4.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR 71, and that the procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

7.4.2 Conditions of Approval

Because they represent the framework from which the formal, site-specific operating procedures will be developed for each user/shipper, the staff concludes that the generic operating procedures delineated in Chapter 7 of the SARP must be incorporated in their entirety into the Certificate of Compliance as a condition of package approval.

The staff also concludes that all shipments made under this application will be limited to the shipment of high-purity plutonium metals and/or alloys, as defined in Table 1.14 of the SARP, and plutonium oxide, as defined in Table 1.15 of the SARP.

The staff further concludes that, based on the requirements specified in DOE AL SD 5610.14, all shipments must be made in SSTs and/or SGTs.

7.5 References

Institute of Nuclear Materials Management, American National Standard for Radioactive Materials, “Leakage Tests on Packages for Shipment,” ANSI N14.5-1997, New York.

International Atomic Energy Agency (IAEA), Safety Series No. 6, “Regulations for the Safe Transport of Radioactive Material,” 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), “Compatibility with the International Atomic Energy Agency (IAEA),” 60 FR 50248, September 28, 1995, as amended.

Title 10, Code of Federal Regulations, Part 835 (10 CFR 835), “Occupational Radiation Protection,” 58 FR 65485, December 14, 1993, as amended.

Title 20, Code of Federal Regulations, “Procedures for Receiving and Opening Packages,” 10 CFR 20.1906, 56 FR 23401, May 22, 1991, as amended.

Title 49, Code of Federal Regulations, Part 173 (49 CFR 173), “Shippers—General Requirements for Shipments and Packagings,” 57 FR 20953, May 15, 1992, as amended.

U.S. Department of Energy, “Control and Accountability of Nuclear Materials,” DOE O 474.1, August 11, 1999. See also, Manual for Control and Accountability of Nuclear Materials, DOE M 474.1-1A, November 22, 2000.

U.S. Department of Energy, “Transportation Safeguards System Program Operations,” DOE AL SD 5610.14, May 12, 1993.

U.S. Nuclear Regulatory Commission, "Guide for Preparing Operating Procedures for Shipping Packages," NUREG/CR-4775, July 1988.

U.S. Nuclear Regulatory Commission, Standard Format and Content of Part 71, "Applications for Approval of Packaging for Radioactive Material," Task FC 416-4, Division 7, Proposed Revision 2 to Regulatory Guide 7.9, May 1986.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8, Acceptance Tests and Maintenance Program of the Safety Analysis Report —Packages (SARP) for the 9975 Package was reviewed to verify that the Acceptance Tests meet the requirements of 10 CFR 71, and that the Maintenance Program is adequate to assure packaging performance during its service life. Commitments specified in the Acceptance Tests and Maintenance Program chapter of the SARP are typically included in the CoC as conditions of package approval.

8.1 Areas of Review

8.1.1 Acceptance Tests

Acceptance Tests and Maintenance procedures that assure that the 9975 packaging will be fabricated, accepted, and maintained in a manner consistent with its evaluation for approval were reviewed. The Acceptance Tests portion of this review included the following:

- Visual Inspections and Measurements
- Weld Examinations
- Component Tests
- Materials Tests
- Structural and Pressure Tests
- Leakage Rate Tests
- Shielding Tests
- Thermal Tests
- Additional Tests

8.1.2 Maintenance Program

The Maintenance Program portion of the review included:

- Component Tests
- Material Tests
- Structural and Pressure Tests
- Leakage Rate Tests
- Thermal Tests
- Additional Tests

8.1.3 Appendices

8.2 Regulatory Requirements

8.2.1 Acceptance Tests

The regulatory requirements of 10 CFR 71 applicable to the Acceptance Tests portion of this review are as follows:

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- Before first use, the fabrication of each packaging must be verified to be in accordance with the approved design. [§71.85(c)]
- Before first use, each packaging must be inspected for cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its effectiveness. [§71.85(a)]
- Before first use, if the maximum normal operating pressure of a package exceeds 35 kPa (5 psi) gauge, the containment system of each packaging must be tested at an internal pressure at least 50% higher than maximum normal operating pressure to verify its ability to maintain structural integrity at that pressure. [§71.85(b)]
- Before first use, each packaging must be conspicuously and durably marked with its model number, serial number, gross weight, and a package identification number. [§71.85(c)]
- The licensee must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]

8.2.2 Maintenance Program

The regulatory requirements of 10 CFR 71 applicable to the Maintenance Program portion of the review are:

- The application must identify the established codes and standards used for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of such codes, the application must describe the basis and rationale used to formulate the quality assurance program. [§71.31(c)]
- The packaging must be maintained in unimpaired physical condition except for superficial defects such as marks or dents. [§71.87(b)]
- The presence of any moderator or neutron absorber, if required, in a fissile material package must be verified prior to each shipment. [§71.87(g)]
- The licensee must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]

8.3 Review Procedures

The following procedures are applicable to the review of the Acceptance Tests and Maintenance Program Chapter of the SARP for the 9975 Packaging. In general, these procedures correspond to the Areas of Review listed in Section 8.1 of this SER. Where appropriate, however, these requirements are also supplemented by the guidance and/or the requirements specified in “Fabrication Criteria for Shipping Containers” (NUREG/CR-3854), “Welding Criteria for Use in the Fabrication of Radioactive Material Shipping Containers” (NUREG/CR-3019), and the “American National Standard for Radioactive Material-Leakage Tests on Packages for Shipment” (ANSI N14.5).

8.3.1 Acceptance Tests

Chapter 8 of the SARP indicates that Acceptance Tests are performed prior to the first use of each package. Information presented on each test includes a description of the test and its acceptance criteria, as appropriate. Also, where applicable, sections of the Quality Assurance program (Chapter 9 of the SARP) and the Operating Procedures (Chapter 7 of the SARP) have been be referenced, as applicable.

8.3.1.1 Visual Inspections and Measurement

The applicant for the 9975 Packaging has required the following visual inspections and measurements:

8.3.1.1.1 Overpack Assembly

The applicant has stated that the overpack assembly, consisting of the stainless steel drum for the 9975 Packaging and the cane fiberboard insulation assembly, shall be inspected prior to first usage in accordance with the inspection criteria provided in Appendix 8.1 of the SARP. The inspection criteria require verification of workmanship quality, correct fit of components, manufacture of components to dimensions within specified tolerances, and correct overpack marking information.

8.3.1.1.2 Containment Vessels

The applicant has stated that the package owner shall confirm that the materials of the primary and secondary containment vessels (PCVs and SCVs) are as specified. The applicant has further stated that the package owner shall confirm that the significant features of the PCVs and SCVs have been verified to be within the prescribed design parameters. In addition, the applicant has required that the verification has been documented, and that the documentation has been archived into the quality control file for each PCV and SCV.

Features critical to the containment function of the PCVs and SCVs are defined as “Q” Items. These significant features (i.e., dimensions, surfaces, etc.) of the PCV and SCV components are identified by a “Q” Number on the applicable drawings in Appendix 1.1 of the SARP. They are also tabularized with other packaging “Q” Items in Appendix 8.2 of the SARP.

8.3.1.1.3 Aluminum Honeycomb Impact Absorber

The applicant has stated that the owner/shipper shall verify that the impact absorber material and crush strength are as specified on Drawings R-R4-F-0013, or R-R4-F-0054, as applicable, in Appendix 1.1 of the SARP.

8.3.1.1.4 O-rings

The applicant has stated that the owner/shipper shall verify that the O-rings are within the 10-year shelf life limit. The applicant has also stated that the owner/shipper shall verify that the material is Viton GLT, that the size is correct, and that there are no nicks, cracks, pits, or flat spots on the O-rings as specified by the O-ring vendor quality assurance requirements.

8.3.1.1.5 PCV Sleeve

The applicant has stated that the owner/shipper shall verify that the outside diameter, inside diameter and height of the 9975 Packaging PCV Sleeve is as specified on Drawing R-R4-F-0055 in Appendix 1.1 of the SARP.

(Note: The PCV Sleeve is not required for the shipment of high-purity metals and/or alloys.)

8.3.1.1.6 3013 Top Spacer

The applicant has stated that the owner/shipper shall verify that the outside diameter and height of the 9975 Packaging 3013 top spacer is as specified on Drawing R-R4-F-0055 in Appendix 1.1 of the SARP.

(Note: The 3013 Top Spacer is not required for the shipment of high-purity metals and/or alloys.)

8.3.1.2 Weld Examinations

Although no weld examinations were specified directly in the body of Chapter 8 of the SARP, a number of weld examinations are specified in Section III of Appendix 8.1. It has been verified, however, that all applicable welding requirements have been specified in the drawings in Appendix 1.1 of the SARP.

8.3.1.3 Component Tests

No component tests were specified in Chapter 8 for the 9975 Packaging.

8.3.1.4 Material Tests

No materials tests were specified directly in Chapter 8 for the 9975 Packaging. It has been verified, however, that all applicable materials tests have been specified in the drawings in Appendix 1.1 of the SARP.

8.3.1.5 Structural and Pressure Tests

The applicant has stated that:

> "The owner shall perform the following test or verify that the test was performed by the manufacturer before first use of the package.
>
> "The requirement of 10 CFR 71.85(b) states that when the maximum normal operating pressure (MNOP) in the containment system exceeds 5 psig, the containment system must be tested at an internal pressure at least 50% higher than the maximum normal operating pressure (MNOP). The design pressure considered in Chapter 2 [of the SARP] is 900 psig for the PCV and 800 psig for the SCV. In practice, testing is conservatively specified at 1,365 ± 10 psig and 1,235 ± 10 psig, respectively.
>
> "This bounds a possible MNOP up to the design pressure. Before first use the containment vessels shall be filled with water, the closures sealed as described in 7.1.3 [of the SARP], and the vessels hydrostatically pressurized through the leak test port to the specified pressure at about 70°F.
>
> "Pressurization of the vessel bodies to the prescribed test pressures shall be verified."

8.3.1.6 Leakage Rate Tests

For the fabrication verification leakage test, the applicant has stated that:

> "…(F)ollowing manufacture, the containment vessels shall be leakage tested with helium at 14.7 psig. Slight overpressure (e.g., 5 psig) is acceptable. The leakage test, performed in accordance with the evacuated envelope method of ANSI N14.5, must demonstrate that the leakage rate is less than 1×10^{-7} ref·cm^3/sec air, per the ANSI N14.5 definition of "leak tight"…."

8.3.1.7 Shielding Tests

The applicant has stated that no shielding integrity tests are required for the 9975 Package. For supporting information, the applicant has also provided a broad-based cross-reference to the information presented in Chapter 5 of the SARP.

8.3.1.8 Thermal Tests

The applicant has stated that no thermal acceptance tests are required for the 9975 Package. For supporting information, the applicant has also provided a broad-based cross-reference to the information presented in Chapter 3 of the SARP.

8.3.1.9 Additional Tests

No additional acceptance tests were specified for the 9975 Packaging.

8.3.2 Maintenance Program

Maintenance Program tests are performed to ensure that packaging effectiveness is maintained throughout its service life. Information presented on each test includes a description of the test and its acceptance criteria, as appropriate.

8.3.2.1 Component Tests

The applicant has stated that no subsystem maintenance is specified for the 9975 Packaging. The applicant has further noted, however, that an inspection of selected packaging components is required before each shipment, as specified in Section 7.1 of the SARP.

The applicant also notes that the primary and secondary containment vessels have no valves or rupture discs, and that the containment closure sealing surfaces are inspected before each usage, and are verified to be free of gouges, cracks, or scratches that could significantly affect the containment capability.

The applicant states that new Viton GLT O-rings, lubricated with silicone high-vacuum grease, shall be installed on the containment vessel cone seal prior to the periodic leakage test (every 12 months), or when visual inspection or post-loading leakage tests indicate that replacement is needed. (See also Section 8.3.2.4 of this SER, below.)

The applicant also notes that spare part O-rings shall be received and controlled by the shipper. In addition, the applicant notes that:

> "The shipper shall verify upon receipt that the O-rings are dated and the date indicates at least 7.5 years remaining of the 10-year shelf life. The shipper shall maintain O-rings until an established shelf life limit of 10 years is reached, at which time they will be disposed of as surplus. The shipper shall be responsible for traceability of each O-ring, using its control number, from the date of manufacture throughout the shelf life period…."

8.3.2.2 Material Tests

No materials tests were specified for the Maintenance Program for the 9975 Packaging.

8.3.2.3 Structural and Pressure Tests

In Section 8.1.2 of the SARP, the applicant has stated that the pressure test of the containment vessel shall be repeated after any structural modifications (i.e., rebuilding) to the containment vessel weldments, the cone seal nut, or the cone seal plug. The applicant has further noted, however, that replacement of the cone seal gland nut (over the leak-test port), the port plug, or the containment vessel's O-rings with like components, does not constitute a structural modification, and does not require pressure testing of the containment vessel.

8.3.2.4 Leakage Rate Tests

For the leakage rate tests portion of the Maintenance Program, the applicant has divided this discussion into two sections, i.e., Post-loading Leakage Tests, and Periodic Leakage Tests. Although the former pertains primarily to the Operating Procedures (i.e., Chapter 7 of the SARP), and the latter pertains primarily to the Maintenance Program, a disclaimer is provided that differentiates between the two requirements:

> "CAUTION: If any of the containment boundary components have been changed since the last annual leakage test, then a leakage test is performed in accordance with the annual leakage test requirement before any radioactive material is loaded. If no components have been changed and the annual leakage test is current, then proceed with the post-load leakage test as described."

The procedures describe the appropriate leakage test to be performed, i.e., the standard, pre-shipment leakage test (with a leakage test requirement of 1×10^{-3} ref·cm^3/sec) vs. the standard, periodic leakage test, for acceptance tests and maintenance (with a leakage test requirement of 1×10^{-7} ref·cm^3/sec).

8.3.2.5 Thermal Tests

The applicant has stated that no periodic thermal testing is required for the 9975 Package.

8.3.2.6 Additional Tests

The applicant has stated that no periodic shielding testing is required for the 9975 Package.

8.3.3 Appendices

Appendices that were included as part of the SARP included Appendix 8.1, Section III, that describes the Visual Inspection Criteria for the 9975 Packaging Overpack Assembly, which defines, in detail, the requirements specified in Sections 8.3.1.1.1 through 8.3.1.1.6, above. Also included was Appendix 8.2, Table 8.2, which lists the Quality Assurance "Q" Items for the 9975 packaging, noted in Section 8.3.1.1.2, above.

8.4 Evaluation Findings

8.4.1 Findings

The staff has reviewed the Acceptance Tests and Maintenance Program information presented in the SARP for the 9975 Package for completeness and compliance with the regulatory requirements. For both, the information provided by the applicant was provided in the format prescribed directly by NRC Regulatory Guide 7.9. Supplemental information on inspections and maintenance, and on records and reporting requirements, has also been provided, in the appropriate level of detail, in Chapters 7 and 9 of the SARP, respectively.

Based on the staff's review of the statements and representations in the application, the staff concludes that the Acceptance Tests for the 9975 Package meet the requirements of 10 CFR 71, and that the Maintenance

Program is adequate to assure packaging performance during its service life. The staff also concludes that the information provided for the Acceptance Tests and Maintenance Program is adequate, regardless of the contents specified in Section 1.2.3 of the SARP.

This review also confirms that the Acceptance Tests and Maintenance Program information included in the SARP meets the requirements of IAEA Safety Series No. 6.

8.4.2 Conditions of Approval

As was noted in the introduction to this section, commitments specified in the Acceptance Tests and Maintenance Program chapter of the SARP are typically included in the CoC as a condition of package approval. The staff concurs and concludes that the Acceptance Tests and Maintenance Program Chapter (Chapter 8) of the SARP must be incorporated, in its entirety, into the CoC as a condition of package approval.

8.5 References

Institute of Nuclear Materials Management, American National Standard for Radioactive Materials, “Leakage Tests on Packages for Shipment,” ANSI N14.5-1997, New York.

International Atomic Energy Agency (IAEA), Safety Series No. 6, “Regulations for the Safe Transport of Radioactive Material,” 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), “Compatibility with the International Atomic Energy Agency (IAEA),” 60 FR 50248, September 28, 1995, as amended.

U.S. Nuclear Regulatory Commission, “Fabrication Criteria for Shipping Containers,” NUREG/CR-3854 (UCRL-53544), March 1985.

U.S. Nuclear Regulatory Commission, “Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials,” NUREG/CR-3019
(UCRL-53044), March 1985.

U.S. Nuclear Regulatory Commission, Standard Format and Content of Part 71, “Applications for Approval of Packaging for Radioactive Material,” Task FC 416-4, Division 7, Proposed Revision 2 to Regulatory Guide 7.9, May 1986.

Westinghouse Savannah River Company, “Safety Analysis Report–Packages 9972-9975 Packages (U),” Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

9. QUALITY ASSURANCE REVIEW

Chapter 9, Quality Assurance (QA) of the Safety Analysis Report —Packages (SARP) for the 9975 Package identifies the applicant's QA requirements for the 9975 Package that are required to assure that the package is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation in the SARP. The review includes an evaluation of the applicant's quality assurance (QA) program with the requirements of 10 CFR 71.

9.1 Areas of Review

The applicant's QA program description and package-specific QA requirements were reviewed. The QA review included the following:

9.1.1 Description of Applicant's QA Program

- Scope
- Program Documentation and Approval
- Summary of 18 Quality Criteria
- Cross-Referencing Matrix

9.1.2 Package-Specific QA Requirements

- Graded Approach for Structures, Systems, and Components Important to Safety
- Package-Specific Quality Criteria and Package Activities

9.1.3 Appendix

9.2 Regulatory Requirements

Regulatory requirements of 10 CFR 71 applicable to the QA review are as follows:

- The application must describe the quality assurance program for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the package. [§71.31(a)(3), §71.37]
- The application must identify established codes and standards proposed for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the application must describe the basis and rationale used to formulate the package quality assurance program. [§71.31(c)]
- Package activities must be in compliance with the quality assurance requirements of Subpart H (§71.101-§71.137). A graded approach is acceptable. [§71.81, §71.101(b)]
- Sufficient written records must be maintained to furnish evidence of the quality of the packaging. These records include results of the determinations required by §71.85; design, fabrication, and assembly records; results of reviews, inspections, tests, and audits; results of maintenance, modification, and repair activities; and other information identified in §71.91(c). Records must be retained for three years after the life of the packaging. [§71.91(c)]
- Records identified in §71.91(a) must be retained for three years after shipment of radioactive material. [§71.91(a)]

- Records must be available for inspection. Records are valid only if stamped, initialed, or signed and dated by authorized personnel or otherwise authenticated. [§71.91(b)]
- Any significant reduction in the effectiveness of a packaging during use must be reported to the certifying authority. [§71.95(a)]
- Details of any defects with safety significance in a package after first use, with the means employed to repair the defects and prevent their reoccurrence, must be reported. [§71.95(b)]
- Instances in which a shipment does not comply with the conditions of approval in the certificate of compliance must be reported to the certifying authority. [§71.95(c)]

9.3 Review Procedures

The following procedures are generally applicable to the review of the QA chapter of the SARP. These procedures correspond to the Areas of Review listed in Section 9.1 of this SER.

9.3.1 Description of Applicant's QA Program

9.3.1.1 Scope

Section 9 of the SARP, Quality Assurance, was reviewed to determine compliance with the acceptance criteria. Section 9 states that the applicant's packaging QA program satisfies the intent of 10 CFR 71 Subpart H. It is also stated that the applicant implement the activities described in SRS WSRC 1Q Quality Assurance Manual for the 9975 Package.

9.3.1.2 Program Documentation and Approval

Section 9 of the SARP, Quality Assurance, was reviewed to determine compliance with the acceptance criteria. Section 9.2 of the SARP identifies a QA Plan developed to comply with DOE O 414.1, the governing QA document. The procedures to be followed in implementing the QA plan are identified in the Westinghouse Savannah River Company (WSRC) Quality Assurance Manual (WSRC 1Q Manual). In the preface of the SARP, the applicant identifies the following activities that are covered by the QA Program: package design, purchasing, fabrication, handling, shipping, storage, cleaning, assembly, inspection, testing, operation, maintenance, repair, and component modification. WSRC use of the package is governed by WSRC 1Q. Non-WSRC users must follow their DOE-approved QA program. In Section 9.2.1 of the SARP, the applicant states that the QA program complies with DOE Order 460.1A, 10 CFR 71 Subpart H, and ASME NQA-1. The QA Program scope and applicable procedures are listed in Table 9-1 for each of the required 18 quality elements identified in 10 CFR 71 Subpart H and ASME NQA-1. The QA program and its approval are based on Subpart H of 10 CFR 71. Procedures are identified for all activities performed during SARP preparation as described in Regulatory Guide (RG) 7.10, Annex 1.

9.3.1.3 Summary of 18 Quality Criteria

Table 9.1 of the SARP identifies the QA Procedures and Corresponding Regulatory Elements. The 18 identified procedures correspond to WSRC 1Q Manual Procedures and the 18 criteria of 10 CFR 71, Subpart H. The NMS&S QA Department is responsible for monitoring the activities of WSRC package users by performing QA audits required by §71.137.

9.3.1.4 Cross-Referencing Matrix

Table 9.1 includes a cross-reference matrix from the QA program to the requirements of 10 CFR 71, Subpart H. Each of the 18 criteria in 10 CFR 71 Subpart H is addressed by written procedures.

9.3.2 Package-Specific QA Requirements

9.3.2.1 Graded Approach for Structures, Systems, and Components Important to Safety

Table 9.2a of the SARP provides a correlation between the NRC Regulatory Guide 7.10 Safety Categories and WSRC Safety Designations. There is a one-to-one correspondence between the NRC Regulatory Guide 7.10 and WSRC characterization methods. Table 9.2b provides a summary of the safety designation of each component critical to safety (Q-List) for the 9975 Package. The safety designations indicate a graded approach and provide a summary that is consistent with other analyses in the SARP.

9.3.2.2 Package-Specific Quality Criteria and Package Activities

Table 9.3 of the SARP identifies the QA elements that apply to each safety category listed in Table 9.2 of the SARP. The SARP addresses each of the 18 quality criteria in Subpart H as they apply to the 9975 Package.

Requirements for many fabrication processes (e.g., welding, heat treating, and nondestructive examination) and materials are compliant with the ASME B&PVC, Section III, Subsection NB (see Table 9.4). The mechanical properties of the following materials are specified by ASTM standards: Celotex (ASTM Specification C208), lead (ASTM B749), inner and outer O-rings (Viton GLT O-rings, Parker compound No. V835-75 or equivalent, and fluorocarbon rubber with ASTM F104 designation, conforms to MIL-R-83485). No specification is provided for honeycomb spacers. The package user is responsible for assuring that acceptable honeycomb is used.

Section 9.6 and Table 9.5 identify documents that must be controlled by DOE-approved QA programs. The controlled records include operating procedures (SARP Section 9.6) and records from the acceptance testing/maintenance program (SARP Section 9.11). The records identified are consistent with the requirements of 10 CFR 71. Retention periods for documents are given in Table 9.5.

9.3.3 Appendix

The appendix includes a list of references. The Quality Assurance Plan includes detailed WSRC QA procedures.

9.4 Evaluation Findings

9.4.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the quality assurance program has been adequately described and meets the quality assurance requirements of 10 CFR 71. Package-specific requirements are adequate to assure that the package is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation. By meeting the requirements of 10 CFR 71, the package also meets the requirements of IAEA Safety Series 6.

9.4.2 Conditions of Approval

The applicant shall maintain and follow an appropriate QA program that is compliant with ASME-NQA-1 and 10 CFR 71, Subpart H.

Each non-WSRC user of the package shall maintain and follow an appropriate QA program compliant with ASME-NQA-1 and 10 CFR 71. Each user's QA program shall be DOE-approved. Each user's QA organization shall monitor the activities of the package users by performing audits, surveillances, and inspections.

9.5 References

American Society for Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facilities," ASME NQA-1.

American Society for Testing and Materials, "Standard Classification System for Nonmetallic Gasket Materials," ASTM F104, Philadelphia, PA.

International Atomic Energy Agency (IAEA), Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1985 Edition (as amended 1990), Vienna, Austria (1990).

Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Compatibility with the International Atomic Energy Agency (IAEA)," 60 FR 50248, September 28, 1995, as amended.

U.S. Air Force, Military Specification, "Rubber, Fluorocarbon Elastomer, Improved Performance at Low Temperatures, O-Rings, Sizes and Tolerances," MIL-R-83485/1, Wright Patterson Air Force Base, Dayton, OH, September 8, 1976.

U.S. Department of Energy Order, "Packaging and Transportation Safety," DOE O 460.1A, Washington, DC, October 2, 1996.

U.S. Department of Energy Order, "Quality Assurance," DOE O 414.1, Washington, DC, November 24, 1998.

U.S. Nuclear Regulatory Commission, Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," Rev. 1, Washington, DC, June 1986.

Westinghouse Savannah River Company, "Safety Analysis Report–Packages 9972-9975 Packages (U)," Radioactive Materials Packaging Technology, Savannah River Technology Center, WSRC-SA-7, Revision 12, June 2001.

ADDITIONAL REFERENCES

Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1985 Edition), Third Edition (as amended 1990), International Atomic Energy Agency, Vienna, Austria, 1990.

Mok, G.C. et al. “Guidelines for Conducting Impact Tests of Shipping Containers for Radioactive Material,” UCRL-ID-121673, Lawrence Livermore National Laboratory, September 1995.

U.S. Nuclear Regulatory Commission Bulletin 97-02, “Puncture Testing of Shipping Packages under 10 CFR 71,” September 23, 1997.

U.S. Nuclear Regulatory Commission, “Buckling Analysis of Spent Fuel Basket,” NUREG/CR-6322 (UCRL-ID-119697), May 1995.

U.S. Nuclear Regulatory Commission, “Compatibility with the International Atomic Energy Agency,” Federal Register, Volume 60, No. 188, September 28, 1995, p. 50257.

U.S. Nuclear Regulatory Commission, “Control of Heavy Loads at Power Plants,” NUREG-0612, July 1980.

U.S. Nuclear Regulatory Commission, “Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages, Volume 3: Final Summary Report,” NUREG/CR-2146, Vol. 3, October 1983.

U.S. Nuclear Regulatory Commission, “Hydrogen Gas Ignition during Closure Welding of a VSC-24 Multi-Assembly Sealed Basket,” NRC Information Notice 96-34, May 31, 1996.

U.S. Nuclear Regulatory Commission, “Methods for Impact Analysis of Shipping Containers,” NUREG/CR-3966 (UCID-20639), November 1987.

U.S. Nuclear Regulatory Commission, “SCANS (Shipping Package ANalysis System): A Microcomputer Based Analysis System for Shipping Package Design Review,” NUREG/CR-4554 (UCID-20674), February 1990.

U.S. Nuclear Regulatory Commission, “Shock and Vibration Environments for a Large Shipping Container During Truck Transport (Part II),” NUREG/CR-0128, August 1978.

U.S. Nuclear Regulatory Commission, “Stress Analysis of Closure Bolts for Shipping Packages,” NUREG/CR-6007 (UCRL-ID-110637), January 1993.