Safety Evaluation Report
for the

Safety Analysis Report for Packaging
9977

S-SARP-G-00001, Revision 2, August 2007

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Overview

This Safety Evaluation Report (SER) summarizes the review findings for the Safety Analysis Report for Packaging (SARP) for the 9977 B(M)F-96 shipping container. The content analyzed for this submittal is Content Envelope C.1, Heat Sources, in assemblies of Radioisotope Thermoelectric Generators or food-pack cans. The SARP under review, i.e., S-SARP-G-00001, Revision 2 (August 2007), was originally referred to as the General Purpose Fissile Material Package.

The review presented in this SER was performed using the methods outlined in Revision 3 of the Department of Energy's (DOE's) *Packaging Review Guide (PRG) for Reviewing Safety Analysis Reports for Packages*. The format of the SARP follows that specified in Revision 2 of the Nuclear Regulatory Commission's, Regulatory Guide 7.9, i.e., *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*. Although the two documents are similar in their content, they are not identical. Formatting differences have been noted in this SER, where appropriate.

The 9977 is a 35-gallon drum package design that has evolved from a family of packages designed by DOE contractors at the Savannah River Site. The 9977 design includes a single, 6-inch diameter, stainless steel pressure vessel containment system (i.e., the 6CV) that was designed and fabricated in accordance with Section III, Subsection NB, of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code.

The earlier package designs, i.e., the 9965, 9966, 9967 and 9968 Packages, were originally designed and certified in the 1980s. In the 1990s, updated package designs that incorporated design features consistent with new safety requirements, based on International Atomic Energy Agency guidelines, were proposed. The updated package designs were the 9972, 9973, 9974 and 9975 Packages, respectively. The 9975 Package was certified by the Packaging Certification Program, under the Office of Safety Management and Operations.

Differences between the 9975 Package and the 9977 include:

- The lead shield present in the 9975 Package is absent in the 9977;
- The 9975 Package has eight allowable contents, while the 9977 has a single allowable content.
- The 6CV of the 9977 is similar in design to the outer Containment Vessel of the 9975 Package that also incorporates a 5-inch Containment Vessel as the inner Containment Vessel.
- The 9975 Package uses a Celotex®-based impact limiter while the 9977 uses Last-A-Foam®, a polyurethane foam, for the impact limiter.
- The 9975 Package has two Containment Vessels, while the 9977 has a single Containment Vessel.

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Acronyms and Abbreviations

6CV	6-inch inside diameter Containment Vessel
9977	General Purpose Fissile Package
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
B&PVC	Boiler and Pressure Vessel Code
BD	Bottom Down
CFR	Code of Federal Regulations
CG	Center of Gravity
CGOC	Center of Gravity over Corner
CGOT	Center of Gravity over Top
CoC	Certificate of Compliance
CSI	Criticality Safety Index
CV	Containment Vessel
DOE	Department of Energy
DOT	Department of Transportation
DR	Digital Radiography
EM	Office of Environmental Management
ENDF	Evaluated Nuclear Data Files
FEA	Finite Element Analysis
FR-3716	General Plastics FR-3716 Polyurethane Foam
GPFP	General Purpose Fissile Package (working name for the 9977)
HAC	Hypothetical Accident Conditions
IAEA	International Atomic Energy Agency
ID	Inside Diameter
kPa	kilo-Pascals
LDF	Load Distribution Fixture
LLNL	Lawrence Livermore National Laboratory
MCNP	Monte Carlo n-Particle Transport Code
MNOP	Maximum Normal Operating Pressure
NCR	Nonconformance Report
NCT	Normal Conditions of Transport
NRC	Nuclear Regulatory Commission
ppm	parts-per-million
psi	pressure in pounds per square inch
psig	pressure in pounds per square inch, gauge
QA	Quality Assurance
RASTA	Radiation Source Term Analysis
Ref	Reference
Reg. Guide	Regulatory Guide

Acronyms and Abbreviations
(Cont'd.)

RTG	Radioisotope Thermoelectric Generator
SARP	Safety Analysis Report for Packaging
SER	Safety Evaluation Report
SGT	Safe-Guards Transporter
SRNL	Savannah River National Laboratory
SRPT	Savannah River Packaging Technology
SRS	Savannah River Site
SS	Stainless Steel
SST	Safe-Secure Trailer
TD	Top Down
TID	Tamper-Indicating Device
TIL	Temperature Indicating Labels
UN	United Nations
UNS	Unified National — Special
WSMS	Washington Safety Management Solutions
WSRC	Washington Savannah River Company, LLC

1.0 General Information Review

This Safety Evaluation Report (SER) documents the review of Chapter 1, General Information Review, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[1-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[1-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[1-3]

1.1 Areas of Review

The following elements of the General Information Chapter were reviewed. Details of the review are provided in Section 1.3, below.

1.1.1 Introduction

- Purpose of Application
- Summary Information
- Statement of Compliance
- Summary of Evaluation

1.1.2 Package Description

- Packaging
- Contents
- Special Requirements for Plutonium
- Operational Features

1.1.3 Appendices

- Drawings
- Other Information

1.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the General Information Review of the 9977 include:

- An application for package approval must be submitted in accordance with Subpart D of 10 CFR 71. [§71.0(d)(2)]
- 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)]
- 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)]
- The application must reference or describe the quality assurance program applicable to the package. [§71.31(a)(3), §71.37]

- 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 smallest overall dimension of the package must not be less than 10 cm (4 in.). [§71.43(a)]
- The outside of the package must incorporate a feature that, while intact, would be evidence that the package has not been opened by unauthorized persons. [§71.43(b)]
- A package with a transport index greater than 10, a Criticality Safety Index greater than 50, or an accessible external surface temperature greater than 50°C (122°F) must be transported by exclusive-use shipment. [§71.43(g), §71.47(a), §71.47(b), §71.59(c)]
- The maximum activity of radionuclides in a Type A package must not exceed the A_1 or A_2 values listed in 10 CFR 71, Appendix A, 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)]
- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).
- A fissile material package must be assigned a Criticality Safety Index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59, §71.35(b)]
- Plutonium in excess of 0.74 TBq (20 Ci) must be shipped as a solid. [§71.63]
- The package must be conspicuously and durably marked with its model number, serial number, gross weight, and package identification number. [§71.19, §71.85(c)]

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. These procedures correspond to the *Areas of Review*, listed above in Section 1.1.

1.3.1 Introduction

1.3.1.1 Purpose of Application

The 9977 under the current submission was docketed as a new package. The purpose of the application is to document that the 9977, under Revision 2 of WSRC S-SARP-G-00001, satisfies the regulatory requirements of 10 CFR 71 and the *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1.

The application is complete and, with a few exceptions, contains all of the required information identified in 10 CFR 71, Subpart D. (Note: The exceptions referred to here will be addressed in each of their respective Chapters.)

1.3.1.2 Summary Information

The 9977 is designed to transport actinide oxides in excess of Type A quantities. The package is designed for an internal pressure of 5,600 kPa (800 psi). The package type, and model number, i.e., 9977 B(M)F-96, is provided on Drawing R-R2-G-00017, Rev. 1 of the SARP. Packages are shipped in a closed conveyance under *non-exclusive use* dose-rate limits in the Safe-Secure Trailer (SST), the Safe-Guards Transporter (SGT), or by commercial carrier, as determined by the contents and by DOE Order 470.4.[1-4] Package users may also ship 9977s in accordance with *exclusive use* dose-rate limits via the SST or the SGT, as long as they have prior written approval from the DOE Office of Secure Transportation. The 9977 is not authorized for shipments by air.

Based on Content Envelope C.1 in Table 1.2, the calculated hazard (in A_2) and total activity are 63,300 A_2 and >1,800 Ci, respectively. Therefore, Section 2.1.4 states that the package is designed as Category I, per Regulatory Guide 7.11.[1-5]

Section 1.2 of the SARP includes a summary of the design criteria for the package. The American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, 2004 Edition, will be used to determine package containment design, fabrication, and inspection requirements.[1-6] The drum is designed, analyzed, and fabricated in accordance with Section III, Division 1, Subsection NF of the ASME B&PV Code.[1-7] General Plastics FR-3716 polyurethane foam (Last-A-Foam®) occupies the space between Fiberfrax® attached to the liner and the drum wall. General Plastics FR-3716 has a nominal density of 16 lb/ft^3. The Last-A-Foam® acts as an impact limiter. Additional discussion of applicable codes and standards is included in Chapters 2 through 9 of the SER.

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

Limits on package contents are based on nuclear criticality, radiation shielding, and decay heat rate. The stated nuclear Criticality Safety Index (CSI) for the package is 0. (See the related discussion in Section 6.4.1, below.) The Transport Index, based on the estimated dose rate, is calculated to be a maximum of 2.6, and shall be established for each package at the time of shipment.

1.3.1.3 Statement of Compliance

Section 1.1 and Section 1.2.5 contain a statement that the 9977 satisfies the regulatory safety requirements of 10 CFR 71.

1.3.1.4 Summary of Evaluation

Section 1.2.5 of the SARP discusses compliance of the 9977 with the regulatory safety requirements of 10 CFR 71. Criticality requirements are covered in Section 1.2.5.6. General requirements for all packages are covered in Section 1.2.5.2. Structural requirements for lifting and tie-down devices are given in Section 1.2.5.3. External radiation requirements are in Section 1.2.5.4. Requirements for Type B packages are in Section 1.2.5.5. Special requirements for plutonium-containing packages are covered in Section 1.2.5.7. Structural and thermal performance is given in Section 1.2.5.1. Requirements for operating controls and procedures are listed in Section 1.2.5.8. Requirements for quality assurance are in Section 1.2.5.9.

1.3.2 Package Description

1.3.2.1 Packaging

The 9977 is depicted in Figures 1.1 through 1.4. The packaging outer container is a 35-gallon drum meeting the performance requirements of 49 CFR 178[1-8] for an open-head drum that is modified with a bolted-flange closure. The drum shell and liner are fabricated of 18-gauge (0.048-inch) Type 304L stainless steel (SS). The drum is designed, analyzed, and fabricated in accordance with Section III, Subsection NF of the ASME B&PV Code. Four, ¾-inch diameter vent holes are drilled at locations around the drum, approximately 90° apart and at each of three elevations, for a total of twelve vent holes along the drum side wall. Five additional holes, i.e., two 1-inch diameter fill holes and three ¾-inch diameter vent holes are located on the bottom of the drum. All of the holes are filled with appropriately-sized Caplug® fusible plastic plugs.

The top portion of the drum incorporates a 3/16-inch thick reinforcing rim (vertical flange) and reinforces the drum head and protects both the closure lid and bolts during Hypothetical Accident Condition (HAC) events. The rim includes eight, 1-inch diameter drain holes that are qualified as package lifting and tie-down points. The drum bottom includes a rolled "wear ring," 0.060-inch thick by ¾-inch inside diameter attached by welds that are external to the drum shell. All overall dimensions of the package are greater than 4 inches.

The drum closure lid is fabricated from 1/8-inch thick Type 304L SS plate. Eight, 5/8-inch by 1 1/4-inch long heavy hex-head bolts with 5/8-inch plain, narrow, Type B washers secure the lid to the top deck plate of the drum body. The threaded inserts that receive the drum-closure bolts are welded to the underside of the drum's top deck plate. The bolt heads are drilled through with a 1/8-inch hole to received Tamper-Indicating Devices (TIDs). The Lid Top and Lid Bottom chambers are fabricated from 18-gauge (0.048-inch) and 14-gauge (0.07-inch) Type 304L SS, respectively. Four, ¼-inch diameter holes through the Lid Plate allow the Lid Top and Lid Bottom volumes to exchange gases. The Lid Top chamber is vented by four, ¼-inch diameter holes that are covered with Caplug® fusible plastic plugs.

1.3.2.1.1 Shielding Features

Radiation shielding is provided primarily by the geometry of the package.

1.3.2.1.2 Criticality Control Features

The 9977 design does not incorporate materials specifically for the purpose of poisoning or moderating neutron radiation.

1.3.2.1.3 Insulation

Two, ½-inch thick blankets of Fiberfrax® insulation are wrapped around and attached to the sides and bottom of the liner. The Fiberfrax® is backed on both sides with fiberglass cloth held in place by fiberglass thread stitched longitudinally at 4-inch intervals. The remaining volume between the Fiberfrax® and the drum wall is filled with General Plastics FR-3716 polyurethane foam, poured through fill holes in the drum bottom and foamed in place. The General Plastics FR-3716 is poured using two, 1-inch diameter fill holes located on the bottom of the drum. The closure lid incorporates two chambers of insulation. The Lid Top chamber contains a 1-inch thick, 14-inch diameter disk of Thermal Ceramics Min-K 2000® insulation. The Lid Bottom chamber contains a rigid disk of Thermal Ceramics TR-19® Block insulation, 4.3-inch thick by

8-inch diameter. When installed, the TR-19 disk compresses two, 8-inch diameter by ½-inch thick blankets of Fiberfrax® to a total thickness of ½ inch.

1.3.2.1.4 Load Distribution Fixtures

The Top and Bottom Load Distribution Fixtures are made from 6061-T6 aluminum round bar, and fit within the Drum Liner cavity, above and below the 6-inch diameter Containment Vessel (6CV).

1.3.2.1.5 Primary Containment Vessel

The 9977 is designed with a CV having a nominal inner diameter of six inches. The 6CV is a stainless steel pressure vessel designed, analyzed, and fabricated in accordance with Section III, Subsection NB of the ASME B&PV Code. The 6CV is fabricated from 6-inch, Schedule 40 seamless, Type 304L SS pipe (0.280-inch nominal wall). A standard Schedule 40 Type 304L SS pipe cap (also 0.280-inch nominal wall) is welded to the pipe segment to form a blind end. A stayed head is machined from a 7½-inch diameter by 2¼-inch long Type 304L SS bar and welded to the open end of the pipe segment, completing the body weldment. The stayed head is machined to include 6½-12UNS-2B internal threads and an internal cone-seal surface with a 32-micro-inch finish.

Both vessel body joints are Category B, full-penetration circumferential welds, as described in Table 9.5 of the SARP.

The 6CV Closure Assembly consists of a Type 304L SS Cone-Seal Plug, shaped in part like a truncated cone, and a threaded Cone-Seal Nut made from Nitronic 60 SS. The two Closure Assembly components rotate freely relative to one another and are coupled by a snap-ring. Both internal and external sealing surfaces are machined to the same angles, surface finishes, and with matching diameters so that they mate with a maximum radial clearance of 0.0007 inches. Two O-ring grooves are machined into the face of the external Cone-Seal Plug. Viton® GLT/GLT-S O-rings fit into the grooves to complete the leak-tight Closure Assembly. A 0.094-inch diameter hole is located in the stayed head between the threads and the internal sealing surface. Unscrewing the Cone-Seal Nut a few turns will unseat the Cone-Seal Plug from the internal cone-seal surface and route any pressurized gases from inside of the 6CV through the vent hole.

A leak-test port is incorporated into the Cone-Seal Plug and connected by a drilled passage to the annular volume between the two O-ring grooves in the Cone-Seal Plug. The leak-test port provides a means of verifying proper assembly of the vessel closure, and is itself closed by the Leak-Test Port Plug. The vessel containment boundary is formed by the vessel body weldment, the Cone-Seal Plug, the Cone-Seal Port Plug (Leak-Test Port Plug), and the Outer O-ring.

1.3.2.2 Contents

Type B quantities of radioactive materials may be shipped in the 9977. A single content is allowed for the 9977, Content Envelope C.1, Heat Sources, as described in Table 1.2 of the SARP. The maximum allowable radioactive decay heat is 19 W. Small concentrations (<1000 parts-per-million [ppm] each) of other actinides, fission products, decay products, and neutron activation products are permitted except, as noted in Table 1.2. Inorganic material impurity quantities of less than 100 ppm each are permitted as long as the total mass is less than 0.1 weight percent of the total content mass. The maximum weight of the payload is not to exceed 100 pounds. The Maximum Normal Operating Pressure (MNOP) of the 6CV is 41.2 psig.

Contents containers are detailed in Section 1.2.2.1 of the SARP. One Radioisotope Thermoelectric Generator (RTG) assembly configuration holding a maximum of four RTGs is one contents container. The other contents container is a food-pack can. Up to one hundred grams of plastic may be present for this contents container form.

1.3.2.3 Special Requirements for Plutonium

All 9977 contents shall be in solid form.

1.3.2.4 Operational Features

All components are handled manually. Rectangular notches are located in the base of the 6CV support skirt that can be used to prevent rotation of the vessel body while installing/removing the Closure Assembly. Four, ¼-inch diameter by ¼-inch deep holes are located in the square flat faces of the Cone-Seal Nut to facilitate lifting the Closure Assembly or the complete 6CV. Special Tools that may be useful for working with the 9977 are listed in Appendix 7.1.

1.3.3 Appendices

1.3.3.1 Drawings

Drawings of the 9977 are provided in Appendix 1.1 of the SARP as follows:

R-R5-G-00002	Rev. 1	9977-Drawing Tree
R-R1-G-00020	Rev. 2	9977-Assembly with 6-inch Diameter Containment Vessel
R-R2-G-00017	Rev. 1	9977-Drum and Liner Subassembly
R-R2-G-00018	Rev. 2	9977-Drum Lid Subassembly
R-R2-G-00019	Rev. 1	9977-Insulating Blanket Subassembly
R-R4-G-00032	Rev. 1	9977-Load Distribution Fixtures Details
R-R2-G-00042	Rev. 2	9977-Six-Inch Diameter Containment Vessel Subassembly

Drawing R-R2-G-00017, Rev. 1, Section 1.2.1.7 and Figure 7.4 of the SARP indicate that the 9977 meet the requirements specified in §71.85(c).

1.3.3.2 Other Information

A list of references is included in Section 1.3.

1.4 Evaluation Findings

1.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above with the following clarifications:

- In a number of places throughout the SARP, the applicant has referred to the *Leak-Test Port Gland Nut* as the *Cone-Seal Gland Nut*. Although this may not directly applicable to the terminology used in Chapter 1, the terminology should be standardized for the Leak-Test Port Gland Nut, throughout, in order to more clearly differentiate between the

Cone-Seal Nut and *Cone-Seal Gland Nut*. (Note: In Chapter 1, there is no description of the Leak-Test Port Gland Nut, at all. While it is shown in Figure 1.4, there is no associated description.)

- A welding symbol is missing from Drawing R-R2-G-00017, Detail C.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above clarifications notwithstanding, and based on review of the statements and representations in the SARP, the Staff concludes that the package design has been adequately described to meet the requirements of 10 CFR 71 and of IAEA Safety Standards Series No. TS-R-1.

1.4.2 Conditions of Approval

In addition to a summary package description and specifications of authorized contents, the following additional Conditions of Approval are applicable to the General Information review of the 9977:

- The maximum allowable radioactive decay heat rate is 19 W;
- The maximum weight of the payload is not to exceed 100 pounds;
- Contents are restricted to those described in Table 1.2 of the SARP, with additional restrictions provided by the applicable Table 1.2 footnotes;
- Content Envelope loading configurations are further restricted to those described in Sections 1.2.2.1 and 1.2.2.2 of the SARP;
- The 9977 was not evaluated to the requirements of §71.55(f), therefore transport by air of fissile material is not authorized.

1.5 References

[1-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[1-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[1-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[1-4] *Safeguards and Security Program*, DOE Order 470.4, U.S. Department of Energy, Washington, DC (August 2005).

[1-5] Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 7.11, *Fracture Toughness of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches*, Washington, DC (June 1991).

[1-6] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code Section III*, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NB, ASME, New York, NY (2004).

[1-7] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code Section III*, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NF, ASME, New York, NY (2004).

[1-8] Department of Transportation, 49 CFR Parts 171, 172, 173, 174, 175, 176, 177 and 178, *Hazardous Materials Regulations; Compatibility With the Regulations of the International Atomic Energy Agency; Final Rule*, 69 F.R. 3632, pp. 3632–3896, January 26, 2004, as amended.

2.0 Structural Evaluation

This Safety Evaluation Report (SER) documents the review of Chapter 2, Structural Evaluation, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[2-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[2-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[2-3]

2.1 Areas of Review

The following elements of the Structural Evaluation Chapter were reviewed. Details of the review are provided in Section 2.3, below.

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 of Lifting and Tie-Down Devices

- Lifting Devices
- Tie-Down Devices

2.1.6 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

2.1.7 Structural Evaluation for Hypothetical Accident Conditions

- Free Drop
- Crush
- Puncture
- Thermal
- Immersion — Fissile material
- Immersion — All packages

2.1.8 Structural Evaluation for Special Pressure Conditions

- Special Requirement for Packages >10^5A_2
- Analysis of Pressure Test

2.1.9 Appendices (as applicable)

2.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Structural Evaluation Review of the 9977 are as follows:

- 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 contents. 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 normal conditions of transport. [§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 normal conditions of transport. [§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 normal conditions of transport.
- The performance of the package must be evaluated under the tests specified in §71.73 for hypothetical accident conditions. [§71.41(a)]
- The package design must meet the lifting and tie-down requirements of §71.45.
- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).
- A Type B package, containing more than $10^5 A_2$, must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or inleakage of water. [§71.61]
- 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 following subsections describe the review methods for the Areas of Review applicable to the Structural Evaluation Chapter of the SARP. These procedures correspond to the *Areas of Review*, listed above in Section 2.1.

2.3.1 Description of Structural Design

2.3.1.1 Design Features

As illustrated in Figures 1.1 and 1.2 of the SARP, the 9977 is a single-containment, drum-type package. Its primary structural components are:

- A 35-gallon drum overpack (the overpack), that confines and protects the single containment vessel under both Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) events, and acts as a form for pouring the polyurethane-foam insulation material during the foaming operation;
- A 4.5-inch-thick (axially), poured-in-place polyurethane-foam overpack insulation, and other thermal insulation materials, that provide both impact and thermal protections for the containment vessel;
- A 6-inch-diameter cylindrical containment vessel (6CV), that contains the radioactive contents (i.e., a Radioisotope Thermoelectric Generator (RTG), or a food-pack can with the allowable radioactive contents described as Content Envelope C.1 in the SARP); and
- Two aluminum spacer assemblies, or Load Distribution Fixtures, one at each end of the 6CV, that hold the 6CV in place within the drum cavity, and help soften the interaction of the 6CV and the drum liner (inner wall) by spreading the loads transmitted through the two components.

2.3.1.1.1 Drum Overpack

The overpack consists of an insulated drum and an insulated closure lid, and the closure does not incorporate a gasket. The drum design meets the performance requirements of 49 CFR 178,[2-4] for an open head drum, but is modified with a bolted-flange closure and a drum liner. The drum body is a hollow structural assembly consisting of a cylindrical outer shell, a cylindrical inner liner, a circular drum bottom, a circular liner bottom, and an annular drum top deck plate. The outer edge of the top deck plate is connected to the top of the drum shell using a circular vertical flange or drum rim, and the inner edge of the plate is connected to the drum liner using a similar liner rim. The volume between the drum shell and liner is filled with shock-absorbing and thermal-insulating materials, which will be described later in this section of the SER. The drum shell, liner and their bottoms are fabricated of 18-gauge (0.048-inch) Type 304L stainless steel (SS). Butt welded to the drum shell is a "sanitary" style drum bottom, which has a radiused edge. The drum bottom includes a rolled "wear ring," which is a circular tube, 0.060-inch thick by ¾-inch inside diameter (ID), attached by fillet welds to the outer circumference of the drum bottom. The drum's top deck plate is fabricated of 3/16-inch-thick Type 304L SS plate. The drum rim is also 3/16-inch thick, while the liner rim is thinner (15 gauge, or 0.105 inches). In addition, about a half of the 3-inch length of the drum rim is above the top deck plate to provide protection for both the closure lid and the bolts during HAC events. The rim has eight (8), 1-inch diameter drain holes that are qualified as package lifting and tie-down points. Drum construction details are shown on drawings R-R2-G-00017 and R-R2-G-00018 in the SARP. As applicable, the drum is designed, fabricated, analyzed, and accepted in accordance with Section III, Subsection NF of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) Code.[2-5]

Four (4), ¾-inch diameter vent holes are drilled at locations around the drum, approximately 90°apart, and at each of three elevations, for a total of twelve vent holes along the drum sidewall. Five additional holes, two (2), 1-inch diameter fill holes and three (3), ¾-inch diameter vent holes, are located on the drum bottom. All of the holes are covered with appropriately sized Caplug® fusible plastic plugs. During an HAC fire event, the plugs combust or melt, allowing the drum to vent gases generated by intumescing foam insulation. The vent holes ensure that the drum cannot be ruptured by gas pressure.

The drum closure lid is fabricated from ⅛-inch thick Type 304L SS plate. Eight (8) ⅝-inch by 1¼-inch long heavy hex-head bolts with ⅝-inch plain, narrow Type B washers secure the lid to the top deck plate of the drum body. The closure lid incorporates chambers above and below the lid plate filled with thermo-ceramic shock-absorbing thermal-insulating materials. The lid-top and lid-bottom chambers are fabricated of 18-gauge (0.048-inch) and 14-gauge (0.07-inch) Type 304L SS, respectively. The top of the lid-top is approximately 0.275 inches below the top surface of the drum rim. Both the lid-top and lid bottom chambers reinforce the lid plate and provides thermal protection and shock absorption for the containment vessel during HAC events. The lid-bottom chamber also prevents the closure lid from shearing away from the bolts during HAC events.

Four (4), ¼-inch diameter holes through the lid plate allow the lid top and lid bottom chambers to exchange gases and equilibrate pressure. In addition, the lid top chamber is vented to the outside through four (4), ¼-inch diameter holes that are also covered with Caplug® fusible plastic plugs. The Caplugs® prevent water from entering the lid through the vent holes under NCT. In an HAC fire event, the plugs combust or melt, allowing the lid to vent heated air from the lid top and lid bottom chambers.

To simplify drum-closure operations, the threaded inserts that receive the drum-closure bolts are welded to the underside of the drum's top deck plate. During installation, the bolts are tightened to a torque value of 45 ±5 ft-lb. The bolt heads are drilled through with a ⅛-inch hole to receive Tamper-Indicating Devices (TIDs). Details are shown on Drawing R-R1-G-00020 in the SARP.

2.3.1.1.2 Overpack Insulation

Two layers of thermal insulation material fill the volume between the drum liner and shell. First, two (2), ½-inch thick blankets of Fiberfrax® insulation are wrapped around and attached to the sides and bottom of the liner. The Fiberfrax® is backed on both sides with fiberglass cloth held in place by fiberglass thread stitched longitudinally at 4-inch intervals. The fiberglass cloth gives the Fiberfrax® composite the mechanical strength and wear resistance to retard gas flows during the HAC fire event. The remaining volume between the Fiberfrax® and the drum wall is filled with General Plastics FR-3716 polyurethane foam (also known as Last-A-Foam®), poured through fill holes in the drum bottom and foamed in place. The nominal densities of Fiberfrax® and FR-3716 foam are 7–10 lb/ft^3 and 15–17 lb/ft^3, respectively. The thermal-physical properties of Fiberfrax® and of FR-3716 are listed in Tables 2.9, 2.10, and 3.8 of the SARP. The combined thickness of the two insulators is approximately 4.95 inches radially (i.e., between the liner and the drum shell), and approximately 4.52 inches axially (i.e., between the liner bottom and drum bottom). Details are shown in Figures 1.1 through 1.3, and Drawings R-R1-G-0020, R-R2-G-0017, and R-R2-G-0019 in the SARP.

The lid-top chamber of the closure lid contains a 1-inch-thick, 14-inch-diameter disk of Thermal Ceramics, Min-K 2000® insulation, while the lid-bottom chamber contains a rigid disk of Thermal Ceramics TR-19® Block insulation, 4.3-inch thick by 8-inch diameter. The TR-19® disk was installed over two (2), ½-inch thick blankets of Fiberfrax® insulation of the same diameter. The total thickness of the two blankets under compression is about ½ inch. Thus, the combined axial thickness of all the insulators in the lid-top chamber is approximately 5.75 inches. Details are shown in Figures 1.1 through 1.3 and Drawing R-R2-G-00018 in the SARP.

2.3.1.1.3 Containment Vessel (the 6CV)

The 9977 is designed with a single containment vessel (the 6CV) with a nominal ID of six (6) inches. As illustrated in Figure 1.3 of the SARP, the 6CV is a stainless steel pressure vessel that was designed, fabricated and analyzed in accordance with the requirements specified in Section III, Subsection NB of the ASME B&PV Code.[2-6] The design condition for the 6CV is 800 psig at 300°F, as listed in Tables 2.20 and 3.1 of the SARP.

The 6CV is fabricated from 6-inch, Schedule 40, seamless, Type 304L SS pipe (0.280-inch nominal wall). A standard Schedule 40 Type 304L SS pipe cap (also 0.280-inch nominal wall) is welded to the pipe segment to form a blind end. A stayed head is machined from a 7½-inch diameter by 2¼-inch long Type 304L SS bar, welded to the open end of the pipe segment, completing the vessel body weldment. The head is machined to include 6½-12UNS-2B internal threads and an internal cone-seal surface with a 32-micro-inch finish. Both vessel body joints are Category B, full-penetration circumferential welds, in accordance with the ASME B&PV

Code, Section III, Article NB-3350. A support skirt to stand the 6CV vertically is formed from a short segment of 5-inch, Schedule 40 Type 304L SS pipe, welded to the convex side of the cap. Two rectangular notches milled into the bottom edge of the skirt (180° apart) can engage a rectangular key to prevent vessel rotation during removal and installation of the closure assembly.

The 6CV closure assembly consists of a Type 304L SS cone-seal plug shaped, in part, like a truncated cone, and a threaded cone-seal nut, made from Nitronic 60 SS. The two closure-assembly components rotate freely relative to one another, and are coupled by a snap-ring that also ensures unseating of the closure seal during disassembly. As the cone-seal nut is threaded into the stayed head of the vessel, the cone-seal plug is thrust axially against the corresponding cone-seal surface of the vessel. Both internal and external sealing surfaces are machined to the same angles, surface finishes, and with matching diameters so that they mate with a maximum radial clearance of 0.0007 inches. To minimize the potential for thread galling, the cone-seal nut and the 6CV body are made from dissimilar materials. Two O-ring grooves (outer and inner) are machined in the conical face of the cone-seal plug, as shown in Figure 1.4 of the SARP. Viton® GLT/Viton® GLT-S O-rings fit into these grooves to complete the *leak-tight* closure assembly. For operator safety, a 0.094-inch diameter vent hole is located in the stayed head between the threads and the internal sealing surfaces. The vent hole is clocked 90° from the notches in the vessel support skirt. Unscrewing the cone-seal nut a few turns will unseat the cone-seal plug from the internal cone-seal surface and route any pressurized gases from the 6CV through the vent hole.

A leak-test port is incorporated into the cone-seal plug and connected by a drilled radial passage to the annular volume between the two O-ring grooves in the cone-seal plug. The leak-test port provides a means of verifying proper assembly of the vessel closure, and is itself closed by a Leak-Test-Port Plug and a Gland Nut. The vessel containment boundary is formed by the 6CV body weldment, the Cone-Seal Plug, the Leak-Test-Port Plug, and the outer O-ring.

The internal volume of a closed 6CV is approximately 608 cubic inches. The nominal assembly weight is 52.3 lb, and the nominal overall length is 24 inches. The usable cavity of the 6CV is a minimum of 20.25 inches deep, with a minimum diameter of 5.95 inches. Details are shown in Drawing R-R2-G-00042.

2.3.1.1.4 Load Distribution Fixtures (LDFs)

The Top and Bottom Load Distribution Fixtures (LDFs) are made from 6061-T6 aluminum round bar, and fit within the Drum Liner cavity, above and below the 6CV. The LDFs center the 6CV in the liner, stiffen the package in the radial direction, and distribute the loads away from the 6CV. (See Figures 1.1 through 1.3 of the SARP). The 6CV fits directly into the LDFs. Details are shown on Drawing R-R4-G-00032.

2.3.1.2 Codes and Standards

Table 2.5 of the SARP summarizes the codes and standards used for the design, material characterization, fabrication, examination, and acceptance of the packaging components.

The 6CV body and closure are fabricated in accordance with the drawings and specifications listed in Appendix 1.1 of the SARP, and follow the requirements of the ASME B&PV Code, Section III, Subsection NB. The O-ring seals are evaluated and used in accordance with the vendor's specifications.[2-7]

The overpack drum is procured to the United Nations (UN) drum specification UN/1A2. The drum liner insert and closure are fabricated in accordance with the drawings and specifications listed in Appendix 1.1 of the SARP. The materials specifications and fabrication requirements are listed in Tables 2.14 and 9.6 of the SARP and follow the requirements of ASME B&PV Code, Section III, Subsection NF, and Section VIII, Division 1.[2-8]

The overpack insulation fillers, namely, the Fiberfrax®, TR-19, Min-K-2000, and General Plastics Last-A-Foam® FR-3716 are commercial products which are evaluated and used in accordance with the vendors' specifications. The Last-A-Foam® can only be installed by General Plastics Manufacturing Company, in a proprietary process described in Appendix 8.5 of the SARP.

2.3.2 Materials of Construction

2.3.2.1 Material Specifications and Properties

Material specifications are tabulated in Table 2.6 of the SARP for all packaging components. The corresponding mechanical properties are listed in Tables 2.7 through 2.15. Static and dynamic engineering, stress-strain curves of the General Plastics Last-A-Foam® are also presented in Figures 2.4 and 2.5 for several temperatures.

The material specifications and properties are consistent with the codes and standards described above in Section 2.3.1.2.

2.3.2.2 Prevention of Chemical, Galvanic, or Other Reactions

Table 2.16 of the SARP identifies the dissimilar materials in contact in the 9977 Packaging, while Table 2.17 identifies the dissimilar materials in contact within the 6CV. Table 2.18 lists the chemical composition of these materials. Section 2.2.2 of the SARP evaluates the chemical, galvanic, and thermal compatibility of these materials and concludes the following:

- Stainless steel is compatible with the polyurethane foam, Fiberfrax,® TR-19, and Min-K 2000® based on users' experience and the chemical stability of the materials;
- The stainless steel and aluminum are corrosion resistant, and the neighboring insulation materials do not retain water;
- The minimal amount of adhesives and lubricants used are low-chloride and non reactive;
- The stainless-steel exterior of the RTG contents is compatible with the stainless interior of the 6CV, as are the food-pack cans and their associated aluminum foil dunnage;
- There are no compatibility or reactivity issues associated with the gases generated by thermal decomposition of the overpack foam or the plastics that are associated with the contents.

2.3.2.3 Effects of Radiation on Materials

Section 2.2.3 of the SARP states that, according to the tests reported, a radiation dose in excess of 10^7 rads is required before significant changes to physical properties of the O-rings are observed.[2-9] Using the dose value reported in Chapter 5 of the SARP for Content Envelope C.1, the SARP shows that the most conservative of the dose received by the O-rings is 4.73×10^5 rads in two years. Thus, the limit of one-year service life for the O-ring specified by the SARP is adequate for mitigating any radiation-related effects on the O-rings.

No degradation or activation of the stainless steel structural components is expected at the neutron and photon dose rates calculated in Chapter 5.

2.3.3 Fabrication, Assembly, and Examination

Section 2.3 of the SARP describes the general procedure for fabrication as follows: The 9977 Packaging is constructed in accordance with the design drawings provided in Appendix 1.1, the quality assurance requirements of Chapter 9, and the fabrication processes delineated in Section 2.3.1. Based on these data, manufactures develop detailed fabrication drawings and specifications that demonstrate their understanding of the requirements. No material purchases or fabrication activities shall commence until the requisite drawings and specifications have been reviewed and approved by the Packaging Design Authority.

2.3.3.1 Fabrication and Assembly

Section 2.3.1 of the SARP describes the details of the fabrication and assembly process of the 6CV and the overpack. The description of the overpack includes the drum body, the drum closure lid, the Fiberfrax® insulation, and the foam.

The 6CV body is constructed of Type 304L SS by butt welding a seamless pipe cap to one end of a six-inch Schedule 40 seamless pipe, followed by a rough cut stayed head butt welded to the opposite end of the pipe. Both welded joints are circumferential, full-penetration, complete-fusion, Category B welds per the ASME B&PV Code, Section III, Subsection NB-3350. A short segment of 5-inch pipe is skip-welded to the convex side of the cap forming a skirt to support the 6CV vertically. The stayed head is rough-cut machined from a 7½-inch diameter by 2¼-inch long bar, and is finish-machined to include 6½-12UNS-2B internal threads and an interior cone-seal surface to a 32-micro-inch finish. Methods used to join the pipe to pipe cap and pipe to stayed closure head shall not reduce the wall thicknesses of the base materials below their minima, as specified by their supporting material standards.

The overpack is comprised of a modified UN 35-gallon drum incorporating a sanitary style bottom, a welded deck-lid/liner insert, and a flanged closure lid. The drum liner insert and closure are fabricated in accordance with the drawings and specifications listed in Appendix 1.1 of the SARP. The materials specifications and fabrication requirements for the overpack are listed in Tables 2.14 and 9.6, and follow the requirements specified in the ASME B&PV Code, Section III, Subsection NF, and Section VIII, Division 1. The overpack drum is procured to the UN drum specification UN/1A2.

Referring to Drawing R-R2-G00019 in Appendix 1.1, the SARP specifies the procedure for the installation of Fiberfrax® insulation on the drum liner. Basically, two (2), ½-inch-thick Fiberfrax® blankets are placed over, and attached to, the liner using tapes and thin wires. The blankets are expected to be compressed to a half of their original thicknesses when the overpack foam is installed. The overpack lid insulation also uses two (2), ½-inch-thick Fiberfrax® blankets.

The volume between the Fiberfrax®-insulated liner and the drum wall is filled with General Plastics Last-A-Foam® FR-3716. The liquid is poured into the volume through one of two (2), 1-inch fill holes located in the bottom of the drum, and sets to form rigid, closed-cell, intumescent polyurethane foam. Three (3), ¼-inch holes are also present to vent the air being displaced. The twelve (12), ¾-inch vent holes in the drum side-wall are closed with Caplugs® during foam emplacement. Installation is done by General Plastic's personnel, using a proprietary process and procedure. (See Appendix 8.5, *Acceptance Tests for Polyurethane Foam in the 9977 Packaging*.)

2.3.3.2 Examination

The surfaces and volume of the 6CV are examined in accordance with ASME B&PV Code, Section III, Subsection NB, as described in Table 9.5 of the SARP. In addition, the closed 6CV assembly shall accept, without binding, a 5.95-inch diameter by 20.25-inch long right circular cylinder, as shown in Figure 2.7 of the SARP.

The overpack drum shell procurement requires the drum to be tested and certified to be acceptable per UN/1A2 standards and Table 9.6 of the SARP. Drawing R-R2-G-00017 in Appendix 1.1 of the SARP requires the drum-liner weldment to be tested hydrostatically at 5 psig for not less than ten minutes. This action is necessary to verify that the insert welds do not leak. All welds on the liner weldment (liner plus top deck plate), and its attachment to the drum shell, are examined visually per ASME B&PV Code, Section III, Division 1, Subsection NF-5000.

As the Last-A-Foam® expands and cures, it will exert some mechanical loads on any confining structure. Thus, the SARP specifies the perpendicularity and circularity of the drum liner be examined after the completion of the first of the two foam-installation stages (two pours). The examination will ensure the drum liner has not moved radially more than 1/8 inch from its assembled position. In addition, an 8-inch diameter by 6.5-inch long gauge block must be able to reach the bottom of the liner without binding. The gauge block dimensions mimic the outer dimensions of the bottom LDF.

During foaming operations, foam samples from each lot/batch are collected and tested, at a minimum, for compressive modulus (parallel and perpendicular to rise), thermal conductivity, specific heat, flame retardency, intumescence, leachable chlorides, density, and compressive strength (parallel and perpendicular to rise). Specific foam installation and acceptance testing requirements are stipulated in Appendix 8.5 of the SARP.

2.3.4 General Considerations for Structural Evaluations

2.3.4.1 Evaluation by Test

As indicated in Section 2.1.2 of the SARP, the regulatory compliance of the 9977 under NCT and HAC is predominately demonstrated by test, and by comparison to similarly licensed packagings, i.e., the 9965–9968 Packagings,[2-10] and the 9975 Packaging.[2-11]

2.3.4.2 Evaluation by Analysis

The non-linear plastic dynamic analyses presented in the SARP are used to augment package NCT and HAC test results. The remainder of the structural analyses are used for demonstrating compliance with codes and regulations.

2.3.5 Lifting and Tie-Down Standards for All Packages

The 9977 design incorporates two features for lifting and tie-downs. First, a set of blind holes is drilled into the cone-seal nut specifically to receive the apparatus for lifting the 6CV during handling operations. Second, the overpack includes a series of eight (8) holes through the reinforcing rim at the top of the drum as points for tie-down and for lifting the package.

2.3.5.1 Lifting Devices

Based on the stress analysis presented in Appendix 2.1, Section 2.6.1 of the SARP concludes:

- The lifting devices for the 6CV and for the package meets 10 CFR 71 requirements for strength;
- The lifting devices for the package may suffer localized permanent deformation, if only one of the eight (8) holes is used; and
- The failure of the lifting devices would not impair the ability of the package to meet other requirements.

2.3.5.2 Tie-Down Devices

Section 2.6.2 of the SARP concludes:

- Based on the stress analysis in Savannah River National Laboratory (SRNL) Document M-CLC-A-00317,[2-12] the tie-down device can withstand, without generating stress in excess of yield strength in any component of the package, the static force specified in 10 CFR 71.45(b)(1); and
- There are no other locations on the package that could be used for tie down. Therefore, §71.45(b)(2) does not apply.

2.3.6 Structural Evaluation for Normal Conditions of Transport

Section 2.6 of the SARP demonstrates, through full scale performance tests, analysis, and comparison to other package designs, that the 9977, under NCT, is in compliance with the performance requirements of 10 CFR 71, i.e. there is no loss or dispersal of radioactive contents (as demonstrated to a sensitivity of 10^{-6} A_2 per hr), no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging. The 9977 is also shown to meet the special requirements for fissile material packages.

2.3.6.1 Heat

Section 2.6.1 of the SARP shows that the stresses generated by the *Heat* condition in the 9977 components, except the 6CV, are insignificant. For the 6CV, the SARP concludes that the stresses are acceptable, based on the following considerations:

- While the 6CV temperature caused by the Heat condition is similar to the 6CV design temperature (300°F), the corresponding 6CV pressure (41.2 psig) is significantly lower than the 6CV design pressure (800 psig). The NCT results were obtained using an experimentally calibrated computer code, and the pressure result includes the effects of the temperature increase and possible gas generation from plastics in the 6CV. (See Section 2.6.1.1 of the SARP.)

- Other than the pressure stresses in the 6CV, there are no significant fabrication or thermal interference stresses. (See Sections 2.6.1.2, 2.6.1.3.2, and 2.6.1.3.3 of the SARP.)
- An ASME Code stress analysis shows that the combined pressure (primary) and thermal (secondary) stresses in the 6CV are within the corresponding allowable limits, as specified by ASME B&PV Code, Section III, Subsection NB, for Level A service loads. (See Sections 2.6.1.3 and 2.6.1.4 of the SARP.)
- The 6CV and its closure design have been hydrostatically-tested to withstand pressure as high as 4,400 psig before bursting. (See Section 2.6.1.3.1 of the SARP.)

2.3.6.2 Cold

The SARP addresses the *Cold* condition in Section 2.6.2, and concludes that the condition will not adversely affect ability of the 9977 Packaging to meet the regulatory requirements:

- The stainless steel has adequate toughness to preclude brittle fracture at -40°F, based on Regulatory Guides (Reg. Guides) 7.11 and 7.12.
- The Viton® GLT/Viton® GLT-S elastomeric O-rings of the 6CV have been repeatedly leak tested at -40°F. The O-rings remained leak-tight to sensitivity less than 10^{-7} reference cm^3/sec.
- The effect of *Cold* on the FR-3716, the TR-19, the Fiberfrax® and the Min-K® 2000 has been accounted for in the package analyses, and investigated through package tests. A prototypical 9977 was tested under exposure to an ambient temperature of -20°F (Appendix 2.10). Test and analysis results demonstrate the acceptable performance of the packaging and its insulation materials at low temperatures.
- The effect of freezing water in the drum-insert cavity was analyzed, and shown to be insignificant.

2.3.6.3 Reduced External Pressure

The reduced external pressure of 3.5 psia is equivalent to an increase of the 6CV pressure of 11.2 psig, which is negligible in comparison to the 6CV design pressure of 800 psi. Thus, the effects of this NCT condition are minimal.

2.3.6.4 Increased External Pressure

In Section 2.6.5, the SARP dismisses the significance of this NCT condition on the following basis:

- An ASME Code, Section III buckling analysis in Appendix 2.2 has demonstrated that the 6CV would not buckle under the increased external pressure.
- The acceptance test of the drum weldment will subject the drum liner to a drum internal pressure of 5 psig. Thus, the liner should not be affected even if the drum cavity is sealed.
- The drum body is not sealed, and an increased external pressure has no effect on its structural performance.

2.3.6.5 Vibration

In addition to introducing the simplified analyses reported in Appendix 2.1 of the random vibration and bolt-loosening effects, Section 2.6.6 of the SARP describes the details of a vibration test and the results for the 9977. The test specimen, identified as SN-2, is a prototypical packaging with a maximum-weight simulated contents. The specimen, in its tied-down position, was subjected to shock and vibration loads conservatively representative of both truck transport (closed conveyance per Section 1.1 of the SARP), and forklift handling. For 20 hours of vibrational testing, which equates to a 20,000-mile travel distance, the following are the findings:

- Digital X-ray examination of the tested package did not reveal visible damage to the foam and other packaging internals.
- The aluminum LDFs had noticeable scuffings, and had produced some aluminum shavings. The result appeared to be due the repeated contacts between the LDFs and its neighbors (i.e., the 6CV and the drum liner).
- Subsequent NCT and HAC tests of the vibration-tested package did not indicate that the package's effectiveness had been reduced by the prior vibration test.

Thus, the 9977 is deemed to be in compliance with the regulatory requirements for this NCT condition.

2.3.6.6 Water Spray

Section 2.6.6 of the SARP reports a water spray test of the SN-2 test package of 9977. The test package, standing right-side up, was subjected to the water spray at a rate that was later estimated to be double that of the regulatory requirement, based on the measurement of a rain gauge sitting on top of the drum lid. The following are the results:

- The test package was 15.2 lb heavier at the end of the test;
- The weight increase was attributed to the accumulation of water in the drum liner cavity, based on the following observations:
    - The cavity has a capacity for storing 13.9 to 15.6 lb of water; and
    - No indication of absorbed water in the foam and other drum insulation materials was evident during the NCT drop, penetration, and compression tests that were performed immediately after the spray test.
- The possible effects of frozen water in the drum cavity were analyzed in Appendix 2.1 and were found to be insignificant.

Therefore, the SARP concludes that the 9977 meets the regulatory requirements for this NCT condition.

2.3.6.7 Free Drop

The NCT 4-ft drop and HAC 30-ft drop and HAC puncture test were performed on an unyielding surface in an environmentally controlled test facility, located in building 723-A, at the Savannah River Site (SRS). Section 2.6.7 and Appendix 2.7 of the SARP describe the unyielding surface as a 5-ft square and 6¼-inch thick high-strength (battleship) steel plate, anchored in a 6-ft square by 36-inch thick reinforced concrete slab. The concrete-steel monolith weighs approximately

19,475 lb, greater than 50 times the maximum weight of the 9977. The SARP compares this unyielding target to the IAEA example unyielding target, which is described in IAEA Advisory Material paragraph 717.2,[2-13] as one that includes a steel plate at least 1.57 inches thick, floated to a concrete block, mounted on firm soil or bedrock, where the combined mass of the steel and the concrete is at least 10 times that of the test package.

Although Section 2.6.7 of the SARP states that "Prototype 9977s were dropped in the top-down and Center of Gravity Over Corner (CGOC) orientations from 4 feet onto a flat unyielding surface," only records of the top-down drop can be found in this Section of the SARP and/or in Appendix 2.10. A 4-ft top-down drop was performed on the prototype model SN-2 following the vibration and water spray NCT tests. Although the drop orientation was estimated to produce the greatest damage to the drum closure and the 6CV closure, negligible surface damage was observed on the lid and drum rim, but the placement of a straight edge across the top of drum identified that a portion of the lid pan was domed. It is uncertain whether the domed lid pan is a result of the 4-foot drop, or how the pan is fabricated and installed. On close inspection of the digital radiographs taken after the 4-ft drop, there was no discernible damage to the drum closure plug from the vertical impact of the top LDF that would be expected if the lid domed due to the drop. DP-1, an earlier 9977 prototype, was also dropped 4 feet top-down. Similar to SN-2, no significant damage was observed to the lid or the drum rim.[2-14]

Finite element analysis of the 4-foot top-down drop, described in Section 2.6.7.2 of the SARP, also showed negligible package deformation. The same finite element model was used for analyzing other NCT drops (i.e., bottom-down, CGOC, and side drops) in extreme environments. However, the SARP presents no details of the NCT analysis results, except the general description of minimal damage to the drum, and negligible stress in the package 6CV closure assembly. (Appendix 2.6 describes analysis results of HAC 30-ft drop and HAC crush test and Appendix 2.9 the results of HAC puncture test.)

Based on the foregoing-described test and analysis results, the SARP concludes that the top-down drop satisfies the regulatory requirement and no additional NCT analyses were performed.

2.3.6.8 Corner Drop

No 1-ft corner drops were performed. The *Corner* drop evaluation, per 10 CFR 71.71(c)(8), is not applicable to the 9977 because its minimum weight of 250 lb exceeds the maximum weight requirement of 220 lb for a cylindrical fissile material package.

2.3.6.9 Compression

This regulatory requirement is satisfied by analysis and test. Section 2.6.9 and Appendix 2.10 of the SARP describe the test, in which a compressive load of 1,750 lb was applied to the 9977 prototype specimen SN-2 for 24 hours. There was no observable deformation to the 9977 at the end of the 24-hour test.

2.3.6.10 Penetration

This regulatory requirement is satisfied by test. As described in SARP Section 2.6.10 of the SARP, the hemispherical end of a vertical steel cylinder of 1.25-in diameter and 13-lb mass was dropped vertically from a height of 40 inches onto the exposed surface of the overpack closure lid. After impact, the steel bar rebounded and impacted the lid surface several times, resulting in multiple indentations on the surface. The indentations were deemed too slight to have reduced the effectiveness of the package.

2.3.6.11 Structural Requirements for Fissile Material Packages

The SARP does not address this requirement in detail. Only a general statement, "Additionally, the 9977 is shown to meet the requirements in 10 CFR 71.55(d) when subjected to the tests specified," appears in Section 2.6 of the SARP.

2.3.7 Structural Evaluation for Hypothetical Accident Conditions

As described in Section 2.7 of the SARP, the 9977 is shown to meet the performance requirements of 10 CFR 71.73 by a combination of physical testing, and comparison to packages of similar design. Analysis results are compared against test results, and are used to augment testing. Appendix 2.8 presents a detailed comparison of the 9977 with its earlier development designs. Physical testing has been performed on these development designs. Design changes made to the packages through the course of development testing are included in the final analysis models for the 9977.

Nine prototypical 9977 test Packagings plus one practice test package have been subjected to some, or all, of the HAC test sequence, and are evidence of the 9977 Packagings being able to meet the performance requirements under the HAC. The nine test packages used were:

- One Practice Package that was built with a 9975 Package drum and bolted flange for a trial HAC thermal test only;
- Five Prototype Series 1 test packages (DP-1, -2, -3, -5 and -6) that were built to compare various overpack designs (drums of 16-in or 18-in diameter and filled with 16-, 20-, or 24-lb/ft^3 polyurethane foam); and
- Four Prototype Series 2 test packages (SN-2, -3, -4, and -5) that differ from the (final) 9977, mainly in the drum bottom design. (The SN-series package design had a standard crimped bottom chime, while the final 9977 design replaced the drum bottom with a welded sanitary closure and the chime with a ¾-inch diameter rolled wear ring. This design change was necessitated because the original bottom crimped chime was split in some of the SN-series tests.)

All five of the DP test packages were subjected to the full suite of HAC sequential tests (30-ft drop, crush, puncture, and thermal). In addition, all of the packages were preconditioned by a 4-ft NCT drop. The packages performed well in these tests, and the containment vessels were verified to be leak-tight following the series of tests. However, the package performance suggested several changes that would enhance the safety margins. These changes led to the SN-series design.

All four of the SN-series test packages were also subjected to the full suite of HAC tests. In addition, the SN-2 test package was preconditioned, using several NCT tests, to assure that the NCT tests did not reduce the effectiveness of the 9977. The 6CV remained leak-tight after the series of HAC tests.

Nonlinear plastic dynamic analyses with a calibrated finite element model of the 9977 were conducted to confirm the regulatory compliance of the package, and to evaluate the package performance under other drop and environment conditions. The analyses showed some large (over 10%) plastic or permanent strains of the 6CV body, but no general collapse of the 6CV body. Permanent deformation in the 6CV closure region was predicted. However, prototype testing indicated that no significant permanent deformation occurs in this critical region.

The 3-ft immersion test was not performed because the total fissile mass in the content of 9977 is much less than the minimum critical masses for the isotopes specified in Content Envelope C.1.

The 50-ft immersion test requirement was satisfied by referencing the immersion testing performed on the containment vessels of the 9968 Packaging, which are identical in design to the 6CV. Test results for the 9968 Package showed no inleakage of water or structural degradation to its containment vessels following the 50-ft immersion test.

Section 5.1 of the SARP shows that calculated dose for a damaged 9977 is <<10 mrem/hour at 1 meter from the package surface, meeting the regulatory requirement of 1 rem/hour at 1 meter (i.e., ~40 inches) from the external surface of the package.

2.3.7.1 Free Drop

The HAC 30-ft drops and the 4-ft NCT drops used the same test facility and unyielding target surface that is described above in Section 2.3.6.7.

As described in the preceding section, nine test packages (i.e., DP-1, -2, -3, -5, -6, and SN-2, -3, -4, and -5) were subjected to the 30-ft HAC drop. Prior to the 30-ft drop, a 4-ft NCT drop was performed on all five DP packages (DP-1, -2, -3, -5 and -6), and one SN package (SN-2). The NCT drop for the DP packages was a Center of Gravity over Top Corner (CGOT) drop, while the NCT drop for SN-2 was a Top Down (TD) drop. The SN-2 test package had also been subjected to the vibration and water-spray NCT tests prior to the NCT drop.

All but one of the nine 30-ft drops were conducted with the test package at the ambient temperature of 75°F and the 6CV pressure at 1 atmosphere. The one exception was the SN-3 package, which was chilled to -20°F for the test.

Of the nine, 30-ft HAC drops, there were three (SN-4, DP-2, and -5) TD drops, two (SN-2 and DP-1) CGOT drops, one (SN-5) Bottom Down (BD) drop, and three (SN-3, DP-3, and 6) side drops. These drop orientations were considered to be most challenging for the 9977. None of the drops was a “shallow-angle” drop, i.e., a drop with the package axis oriented nearly horizontally or nearly vertically.

Sections 2.7.1 through 2.7.5 of the SARP summarize the observed damage caused by the 30-ft drops, i.e.,

- TD drop
    - Drum shell buckled below the drum rim, producing a rolling hoop
    - Drum-closure-lid top domed out above the drum rim
    - Drum-lid edge lifted between bolts (scalloping)

- Digital Radiograph (DR) revealed
    - Drum-lid-plug bottom collapsed and expended into the drum liner
    - Gap between top LDF and drum-lid-plug bottom widened
  - Package rebound height small, about one inch

- BD drop
  - Drum bottom flattened causing the drum shell to be buckled above the bottom chime
  - DR showed no definitive indication of
    - Deformation and displacement of the drum-liner to drum-top-deck connection
    - Change of foam thickness beneath the drum liner
    - Change of gap size between the drum-lid-plug bottom and the top LDF
  - Package rebound height very large, near 6 feet

- Side drop
  - Drum end rims slightly flattened (over a 15° sector at the top and 30° sector at the bottom)
  - Drum lid flange slightly bent caused by the flattened top rim
  - Digital Radiography showed
    - Symmetric buckle of the drum liner near the LDFs
    - No obvious deformation and displacement of the drum-liner to drum-top-deck connection
    - Cracks
  - Package rebound height about 3.5 feet

- CGOT drop
  - A 90° sector drum top rim collapse producing local buckling of the drum shell below the rim
  - Drum lid, and drum-lid plug showed damage similar to those of the TD drop
  - Package rebound height was about 6 inches.

The destructive examination that was performed after the HAC thermal test had proved that none of the above damage had caused the opening of the drum liner and the drum-liner to the drum-top-deck connection.

Non-linear plastic dynamic analyses were performed for the HAC drop tests at several temperatures between -22°F and 300°F. The analyses showed that the HAC drop performance of the 9977 is not sensitive to temperature changes. Therefore, no additional testing at other temperatures was deemed necessary. In Section 2.7.1.4 of the SARP, the analyses were also used to justify the omission of oblique (slap-down and shallow-angle) drops.

2.3.7.2 Crush

Section 2.7.2 of the SARP points out that the HAC crush test does not need to be performed on the 9977 because the (maximum) density of the package (65 lb/ft^3) is above the density limit for the crush-test requirement (62.4 lb/ft^3). However, the crush test was performed to evaluate the possible effects of the test on the 9977. The Staff disagrees with this interpretation of the regulation. This issue will be further addressed in Section 2.4.1, below.

The crush test and the drop test were performed on different test pads. The crush-test pad is an outside test pad, located on an abandoned concrete foundation for Building 8343, in N-Area, at SRS. Section 2.7.2 and Appendix 2.7 of the SARP describe the test pad as a qualified, unyielding impact surface, constructed from a steel plate grouted in place on top of the abandoned building footing. The steel pad is 4-ft square by 3-inch thick, is floated on approximately ¾ inches of grout, and is anchored to the building footer by five (5) 5/8-inch diameter, 7-inch long, Hilti® lag bolts, one at its plate center and four at the plate corners. The combined weight of the base plate and concrete footer is approximately 6,000 lb.

The crush plate is fabricated from a 40-inch square by 2½-inch thick, carbon steel plate, and has a measured weight of 1,170 lb. The plate is threaded for four (4) lifting eyes, located at the four plate corners. During testing, the plate is magnetically released.

The same nine test packages were HAC-crush tested after the 30-drop. In all but one case, the 30-ft drop orientation was also used for the crush test. The one exception is the SN-5 test package. The 30-ft drop was a BD drop, but the crush was a CGOT drop.

The CGOT crush test following the CGOT drop appeared to double the extent of the 30-ft-drop damage on the drum top corner. The corresponding top corner of the drum liner appeared to be slightly buckled, and the edge weld along the closure lid's top chamber developed a crack in the area of the crush. However, the scalloped areas of the drum lid edge, between the bolts that were initially opened by the 30-ft drop were now closed. The crush test also produced a large compression on the previously undamaged drum bottom corner, opposite to the damaged top corner.

The side crush test following the 30-ft side drop also greatly expanded the drop damage at the two ends of the package. The bottom end of the SN-3 test package was temporarily crushed to about one-third of its original diameter. At the peak of drum ovalization, the drum chime split open at about 90 degrees away from the point of crush plate impact. The split was 11-inches long, or about 20% of the drum circumference. Therefore, the drum bottom design was improved to have a welded sanitary closure with a rolled wear ring, instead of the standard crimped chime.

The top crush following the 30-ft TD drop increased the damage to the drum top. The drum shell under the drum rim showed increased buckling deformation, and the drum-lid plug was clearly pushed into the drum liner. The top crush also deformed the bottom rolling hoop of the drum.

Test Package SN-5 was dropped bottom-down from 30 ft and then crushed CGOT, as is shown in Figure 2.49 of the SARP. The drop and crush sequence was intended to challenge the liner connection to the top deck. External damage to the drum, as a result of the bottom-down free drop, was minimal. The crush caused the majority of the drum damage. The crush produced roughly an equivalent level of deformation on the opposing corners of the drum. With the exception of some slight buckling in the liner around the top LDF and the bottom of the lid, DR showed no other apparent damage inside the overpack.

2.3.7.3 Puncture

Section 2.7.3.1 of the SARP describes the puncture testing of the 9977. The puncture bar used for the test was 6 inches in diameter by 40-inch long. Its top edge included a machined ¼-inch radius. The bar was welded to an 18-inch square by 2¼-inch thick steel plate. The base of the bar was also gusseted with three equally spaced 5½-inch by ½-inch thick triangular steel plates. The plate was, in turn, welded to the unyielding target surface that was used for the NCT 4-foot drop and the HAC 30-foot drop.

A horizontal (side) drop over the package center of gravity was selected to do the most damage. Lesser angles over the package CG may exert more localized stress to the drum surface, but, because of the drum assembly's rigidity, due to the foam stiffness, the package will tend to slide and/or rotate from the point of impact, and, therefore, the full inertia of the package would not be realized at the point of impact.

All SN-test packages were puncture-tested with the horizontal drop over the CG. The puncture damage, typical of all the packages, was only a 1/8-inch deep dent on the drum surface. No rupture or indication of a potential rupture of the drum surface was observed in any of the puncture tests.

The SN-3 test package was also subjected to a bottom-down puncture test in an attempt to exploit a rip in the drum chime that occurred during the crush test, described in the preceding section. The puncture test did not increase the size of the split; however, it did cause a section of the split to protrude from the drum bottom, exposing more of the foam.

Appendix 2.9 of the SARP presents the results of a set of nonlinear plastic dynamic analyses of the puncture tests over the CG at two orientations, i.e., 45° and 90° to the drum surface. The analyses concluded that the drum surface would not rupture, based on the comparison of the maximum effective plastic strain to the minimum elongation of the 304L stainless steel. The analyses also indicated that the puncture test can produce plastic strain of more than 2% in the 6CV.

2.3.7.4 Thermal

Section 2.7.4 of the SARP addresses the 9977 under the HAC thermal test conditions. Compliance with the thermal requirements of HAC was shown by test and analysis. Following the HAC drop tests, the four SN-test packages were each subjected to an HAC pool fire in accordance with 10 CFR 71.73(c)(4). One of these test packages (SN-2) had previously been subjected to the full suite of NCT tests. Prior to each pool fire, the package being tested was preheated in an environmental chamber at 200°F for a minimum of four days. The 200°F soak temperature was chosen based upon the calculated 6CV O-ring temperature, under NCT, without insolation, as summarized in Section 3.1.3.1 of the SARP. The 200°F also conservatively bounds the maximum foam temperature of 185°F under the same conditions. The packages were insulated when in transit from the environmental chamber to the test facility, and were heated and insulated while at the test facility prior to beginning the thermal tests.

Temperature-indicating labels (TILs) were used to determine the maximum temperatures reached during the thermal test. The TILs have “dots” which change from white (or yellow) to black, when a specific temperature is reached. The maximum 6CV temperature recorded from the SN-test series, with foam, was 270ºF on the top of the Cone-Seal Nut for SN-4. Based on thermal analysis of the 4-day, 200ºF soak, the temperature of the 6CV increased by about 90ºF as a result of the 30-minute fire.

Following the thermal test, each package was permitted to cool in the pool fire test structure. After the packages had cooled, they were digitally radiographed and destructively examined, and the 6CVs were leak tested. All 6CVs remained *leaktight*, as shown in Tables 2.25 and 2.26 of the SARP.

The destructive examinations following the regulatory burn showed that the degraded foam had a very similar char formation in all of the test packages. However, the volume ratio of degraded foam to un-degraded foam was very different, and might have been due to the different impact damage that the package had received during the HAC impact tests prior to the thermal tests. The large variability of the aforementioned volume ratios could not explain the similar temperature measurements within the drum liner of the test packages, unless the liner temperature did not depend much on the foam for thermal insulation. Thus, the thermal protection of the drum liner and 6CV has to depend on the integrity of the Fiberfrax®.

2.3.7.5 Immersion — Fissile Material

Section 2.7.5 of the SARP explains that the total fissile mass in Content Envelope C.1 is much less than the critical mass. Therefore, the 3-ft water immersion test was not performed following the thermal test.

2.3.7.6 Immersion — All Packages

Section 2.7.6 of the SARP claims exemption from this test on the basis of similarity of the CV closure designs for the 9977 and the 9966. The two packages employ the identical CV closure design. Inspection of the tested 9966 showed that water did not affect the containment vessels in the immersion test where the package was immersed in 52 ft of water for 24 hours.[2-15]

2.3.8 Structural Evaluation of Special Pressure Conditions

2.3.8.1 Special Requirement for Type B Packages Containing more than 10^5 A_2

Section 4.2 of the SARP shows the maximum number of A_2 in the 9977 as being 63,300 A_2, which is less than the 10^5 A_2 limit. Therefore, external water pressure testing of 10 CFR 71.61 is not required for the 9977.

2.3.8.2 Analysis of Pressure Test

Appendix 2.2 of the SARP provides a linear elastic finite element stress analysis of the 6CV hydrostatic pressure test. The stress results are within the allowable limits specified in ASME B&PV Code, Section III, Subsection NB, for hydrostatic test conditions. Thus, the hydrostatic test of the 6CV is safe to perform.

The test pressure used for the 6CV is 1.5 times the design pressure of 800 psig, which is greater than the regulatory requirement of 1.5 times the MNOP. The MNOP, which is listed in Table 2.20 of the SARP, is 41.2 psig.

2.3.9 Appendices

There are ten (10) appendices associated with Chapter 2 of the SARP:

- Appendix 2.1, entitled, *General Normal Condition Design Calculations for 9977 Packaging*;
- Appendix 2.2, entitled, *General Design and ASME Calculations for the 9977 Containment Vessels*;
- Appendix 2.3, entitled, *9977 General Purpose Fissile Packaging Weights*;
- Appendix 2.4, entitled, *9965 Cone-Seal Closure Performance at -40°F*;
- Appendix 2.5, entitled, *9977 Packaging Materials and Components of Fabrication*;
- Appendix 2.6, entitled, *Hypothetical Accident Condition Analysis Drop and Crush for the 9977*;
- Appendix 2.7, entitled, *SRS Drop Test Pad Infrastructure for Drums*;
- Appendix 2.8, entitled, *9977 Packaging Comparisons with the GPFP Development Prototypes*;
- Appendix 2.9, entitled, *Dynamic Analysis of the 9977 Package Puncture Strength for a 40-inch Drop onto a Steel Bar*; and
- Appendix 2.10, entitled, *9977 General Purpose Fissile Packaging Prototype Testing*.

2.4 Evaluation Findings

2.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following exceptions or clarifications:

- Section 2.3.1.2, Table 2.5 of the SARP summarizes the codes and standards used for the design, material characterization, fabrication, examination, and acceptance of the packaging components. It should also be noted, however, that under the "Acceptance" column in Table 2.5 of the SARP, Article NB-5000 of the ASME B&PV Code does not include *Fabrication* requirements. Fabrication requirements can be found in Article NB-4000 of the B&PV Code.

- As a result of the NCT *Vibration* testing, the aluminum LDFs had noticeable scuffings, and had produced some aluminum shavings. Since aluminum is significantly anodic with respect to iron (stainless steel, in this case), the Staff has some potential concerns about a possible scenario in which bulk aluminum and aluminum shavings are in intimate contact with the stainless steel walls of the 6CV, particularly in the presence of an electrolyte such as water. It was further noted above in Section 2.3.6.6 that, as a result of the NCT *Water Spray* testing, the test package was about 15.2 lb heavier at the end of the test, and that the weight increase was attributed to the accumulation of water in the drum liner cavity. In order to eliminate a possible scenario where bulk aluminum and aluminum shavings are in intimate contact with the stainless steel walls of the 6CV in the presence of water, the Staff recommends that the 9977 be shipped only in a closed conveyance.

(Note: This same requirement has been appropriately addressed in Section 7.4.1 of the SARP, i.e., *Packaging Storage*.) Additional emphasis should also be placed on the visual inspections for the LDFs in Chapter 7 of the SARP.

- The results of the four Prototype Series HAC tests led directly to a design change of the drum bottom design. The original SN-series package design had a standard crimped bottom chime, while the final 9977 design had replaced the standard drum bottom with a welded sanitary closure, and the standard drum chime had been replaced with a ¾-inch diameter rolled wear ring. The design change was necessitated because the original bottom crimped chime was split in some of the SN-series tests.

  The Staff agrees that the design changes were necessary, and that the design changes have enhanced the overall safety margins of the package. The Staff also notes, however, that the 9977 design has not actually been tested in its final design configuration.

- For the HAC impact tests described above in Section 2.3.7, the Staff has noted that the SARP included a series of nonlinear plastic dynamic analyses to augment the test results. However, the Staff is not convinced that the finite element model used for these analyses is sufficient for the quantitative prediction of complex deformations and failure modes. For example, the nonlinear analyses from some HAC tests predicted plastic strains in excess of 10% in the 6CV, which were not reported in the test results. Also, the analyses produced results showing plastic strains greater than 2% in the center of the 6CV and about 2% in the closure region of the 6CV for some of the 40-inch puncture drop simulations. These results for the closure region show more plastic strain than those for the 30-foot drop case. This does not seem to be a reasonable result since there is much less available kinetic energy to dissipate through plastic deformation than in the 30-foot drop cases.

  Nevertheless, the Staff still considers the nonlinear analyses useful in providing qualitative understanding of the structural behavior of the 9977. For example, based on the analysis results, the Staff believes that the package has a stable design where the 6CV is well protected. In the worst case, the 6CV would have only localized plastic deformation in small areas away from the CV closure. Thus the intent of Reg. Guide 7.6 to limit the impact energy entering the CV is largely met.

- Section 2.7.2 of the SARP incorrectly uses the maximum average density of the 9977 to justify exemption from the HAC crush test. As shown in Section 2.1.3 of the SARP, a fully loaded 9977 can weigh between 316.7 lb and 354.8 lb. Using the drum volume of 5.349 ft^3, the average density can, therefore, vary from 59.21 to 66.33 lb/ft^3, i.e., from *less* than, to *greater* than, the density limit of 62.4 lb/ft^3 for the crush test. Therefore, the Staff recommends that Table 2.27 of the SARP be revised accordingly.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, the Staff has concluded that, based on their review of the statements and representations in the SARP, the Structural Evaluation described meets the requirements of 10 CFR 71 and of IAEA Safety Standards Series No. TS-R-1.

2.4.2 Conditions of Approval

The 9977 must be shipped in a closed conveyance.

2.5 References

[2-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[2-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[2-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[2-4] Department of Transportation, 49 CFR Parts 171, 172, 173, 174, 175, 176, 177, and 178, *Hazardous Materials Regulations; Compatibility With the Regulations of the International Atomic Energy Agency; Final Rule*, 69 F.R. 3632, pp. 3632–3896, January 26, 2004, as amended.

[2-5] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section III*, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NF, ASME, New York, NY (2004).

[2-6] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section III*, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NB, ASME, New York, NY (2004).

[2-7] *Parker O-ring Handbook*, ORD-5700A, The Parker Seal Group, Parker Hannifin Corporation, Cleveland, OH., http://www.parker.com/O-ring (2001).

[2-8] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section VIII*, ASME, New York, NY (2004).

[2-9] T.E. Skidmore, *Radiation Resistance of Viton GLT O-rings for Mode 9975 Packaging Assemblies (U)*, Westinghouse SRTC Memo, SRT-MTS-98-4117 (September 30 1998).

[2-10] Westinghouse Savannah River Company, *Safety Analysis Report-Packages, USA/9965/B(U)F (DOE), USA/9966/B(U)F (DOE), USA/9967/B(U)F (DOE), and USA/9968/B(U)F (DOE), Packaging of Fissile and Other Radioactive Materials*, DPSPU 83-124-1, Revision 2, Savannah River Site, June 1984 (Revised February 1992).

[2-11] Westinghouse Savannah River Company, *Safety Analysis Report for Packaging, Model 9975, B(M)F-85*, WSRC-SA-2002-00008, Revision 0, Savannah River Packaging Technology, Savannah River National Laboratory (December 2003).

[2-12] T.T. Wu, *Tie-Down Load Analysis of 9977 Shipping Package*, M-CLC-A-00317, Rev. 1, Savannah River Site, Aiken, SC (August 2007).

[2-13] *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, Safety Guide TS-G-1.1 (ST-2), http://www.iaea.org/, International Atomic Energy Agency, Vienna, Austria (July 2002).

[2-14] L.G. Gelder, *General Purpose Fissile Material Package Prototype Testing*, M-TRT-A-00006, Revision 0, Westinghouse Savannah River Company (May 2005).

[2-15] *Immersion of the 9966 Package*, M-TSM-A-00005, Rev. 0, Westinghouse Savannah River Company, Aiken, SC (October 2003).

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3.0 Thermal Evaluation

This Safety Evaluation Report (SER) documents the review of Chapter 3, Thermal Evaluation, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[3-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[3-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[3-3]

3.1 Areas of Review

The following elements of the Thermal Evaluation Chapter were reviewed. Details of the review are provided in Section 3.3, below.

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 Test
- Evaluation by Analysis

3.1.4 Thermal Evaluation under Normal Conditions of Transport

- Initial Conditions
- Effects of Tests
- Maximum and Minimum Temperatures
- 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 Appendices

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

3.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Thermal Evaluation Review of the 9977 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 normal conditions of transport. [§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 normal conditions of transport. [§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 hypothetical accident conditions. [§71.41(a)]
- The package design must not rely on mechanical cooling systems to meet containment requirements. [§71.51(c)]

- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).

3.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Thermal Evaluation Chapter of the SARP. These procedures correspond to the *Areas of Review*, listed above in Section 3.1.

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 9977 to the thermal environment. These components, which primarily consist of the 6-inch diameter Containment Vessel (6CV) and the Last-A-Foam®-filled drum, are described in sufficient detail in Section 1.2 of the SARP, to provide a basis for the thermal evaluation of the package. The primary design features intended to protect the Containment Vessel and O-rings of the 9977 from structural damage and overheating are:

- Last-A-Foam® (or polyurethane foam) overpack, confined in the stainless steel drum, which acts as an impact limiter and provides insulation under hypothetical accident conditions; and
- The stainless steel pressure vessel with Cone Seal Plug and Cone Seal Nut, which provides the containment of the package contents during Normal Conditions of Transport- (NCT) and Hypothetical Accident Conditions- (HAC) imposed structural loads. The containment boundary for the Containment Vessel is completed by the use of a pair of Viton® GLT/Viton® GLT-S O-rings between the Cone Seal Plug and the vessel body.

The 9977 accommodates two types of payloads, either Radioisotope Thermoelectric Generators (RTGs) or nested food-pack cans, which are specified as Content Envelope C.1 in Table 1.2 of the SARP.

3.3.1.2 Decay Heat of Contents

The maximum decay heat rate for the 9977 is given in Table 3.4 of the SARP. The maximum allowable heat rate of 19 W was used in the thermal evaluation of the 9977. Provisions will be made to limit the maximum heat rate to 19 W.

3.3.1.3 Codes and Standards

The structural materials, used in the package, conform to Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.[3-4] The polyurethane foam, i.e., Last-A-Foam® FR-3716, used in the drum, is a proprietary material of General Plastics Manufacturing Company.[3-5]

3.3.1.4 Summary Tables of Temperatures

The maximum temperatures reached in the 9977 components during NCT in the shade and in the sun are given in Table 3.2 of the SARP. The temperatures for food-pack cans, evaluated at three elevation levels inside the 6CV, are listed in Table 3.16. The NCT temperature profile for the

RTG is shown in Figure 3.10. 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 9977, including a 19 W heat source, has a maximum accessible surface temperature of 106°F per Table 3.15 of the SARP, below the limit of 122°F, allowed for nonexclusive-use shipments.

The applicant presents the maximum temperature in the 9977 Packaging components during the hypothetical accident fire in Table 3.2 of the SARP. The temperature in the post-fire cool-down included the insolation specified in 10 CFR 71.73. These maximum temperatures are based on the fire tests of damaged packages, as well as post-fire analysis of undamaged 9977s with a simulated 19 W content decay heat rate. Table A-1, in an appendix to Appendix 3.4, presents the temperatures of the components of 9977s in the fire tests. Table 8 in Appendix 3.4 lists the maximum temperatures of undamaged 9977s, determined by Finite Element Analysis (FEA) simulations for the Containment Vessel and the O-rings. The maximum temperatures of the 6CV are reached at 30 minutes following cessation of the fire.

3.3.1.5 Summary Table of Maximum Pressures

The Maximum Normal Operating Pressure (MNOP) values inside the 6CV cavity of the 9977 for NCT are given in Table 3.19 of the SARP. The maximum pressure during HAC is given in Table 3.27 of the SARP. The pressure inside the 6CV is lower for the HAC fire event.

3.3.2 Material Properties, Temperature Limits, and Component Specifications

3.3.2.1 Material Properties

The required thermal properties for the materials used in the 9977 Packaging were presented in Section 3.2 of the SARP.

The polyurethane foam decomposes under HAC, resulting in a change of the thermal properties following the HAC thermal event. From the test results, three cases of damaged foam were used in the post-fire simulations:

- A one-inch thick layer of foam left surrounding 6CV, the remainder was in the form of char;
- A 2.5-inch thick layer of foam left surrounding 6CV, the remainder was in the form of char;
- All the foam burned into char.

The assumption of the thermal properties of char treated as air were reviewed by the Staff and determined to be acceptable.

3.3.2.2 Temperature Limits

The temperature limits of the 6CV, the Viton® GLT/Viton® GLT-S O-rings, and the Last-A-Foam® are given in Table 3.1 of the SARP, and thermal decomposition temperature of the Last-A-Foam® is described on the page 3-30 of the SARP. The pressure limit of the 6CV is also given in Table 3.1 of the SARP.

3.3.2.3 Component Specifications

The Staff has verified that the component specifications for the drum, the insulation, and the 6CV are presented in sufficient detail in the SARP. Component specifications include the emissivity and absorptivity of the drum, the temperature limits for the insulation, and the temperature limits for the Viton® GLT/Viton® GLT-S O-rings.

3.3.3 General Considerations for Thermal Evaluations

3.3.3.1 Evaluation by Test

Tests, described in Appendix 3.1 of the SARP, were performed on a prototype of the 9977 Packaging with a mock content of a 19 W heater to simulate the decay heat. These tests were used to benchmark the numerical analyses of the package. The package was tested for about 143 hours within a thermal chamber with the internal environmental temperature controlled to 100°F. The measured temperatures in the package were used to benchmark the analyses of the 9977s for NCT, as described in Appendix 3.3 of the SARP.

Following preheating in the thermal environmental chamber to the pre-fire condition, four damaged 9977s were tested in vertical and horizontal orientations in a pool fire for 30 minutes, as described in Appendix 3.2 of the SARP. Before the fire tests, the prototype 9977s had been subjected to the drop, puncture, and crush sequence, per 10 CFR 71.73. The temperature surrounding the outside of the 35-gallon drum exceeded 1,800°F, about 300°F higher than that specified in 10 CFR 71.73. For all the fire tests, the flame extended horizontally more than 1 m (40 in.) beyond the tested packages. Temperature-indicating labels were utilized to track the increase in temperature of the components during the fire. The maximum measured temperature difference in the packages caused by the fire was superimposed on the analytical model for the post-fire analyses, as described in Appendix 3.4 of the SARP.

The temperatures and pressures for the 6CV and the Viton® GLT/Viton® GLT-S O-rings, under both NCT and HAC, are less than the allowable limits given in Table 3.1 of the SARP. The temperature of the polyurethane foam is less than the allowable limit by a few degrees under NCT. During the HAC fire event, the Last-A-Foam® is expected to char and burn.

3.3.3.2 Evaluation by Analysis

The applicant used MSC/PATRAN-THERMAL® (M/PT), a general purpose heat transfer software.[3-6] The PATRAN module of M/PT is used as the pre-processor for creating the finite element model of the package and the post-processor for evaluating the modeling output. The P/Thermal module of M/PT performed the computations, including the determination of the radiation view factors between adjacent package components. Axisymmetric models were used for each package. The thermal properties of the packaging materials, including insulation and air, are listed in Section 3.2 of the SARP. The Staff has deemed these properties are appropriate for the thermal analyses. The Staff has also concluded that the expressions for the various modes of heat transport at the package boundaries are appropriate. The PATRAN-P/Thermal module descriptions are given in Appendix 3.5 of the SARP.

Material properties, convection coefficients, radiation surface properties, and heat generation and insolation data, used in analyses, are listed in Section 3.2 of the SARP. Analyses performed on the undamaged 9977 for NCT were benchmarked against experiments, as discussed in

Appendices 3.4 and 3.5 of the SARP. Thermal properties, verified in NCT, were used for calculating the temperature distribution corresponding to the initial condition before the HAC fire. The post-fire analyses of the undamaged packages, utilized the three Last-A-Foam® conditions noted above in Section 3.3.2.1.

3.3.4 Thermal Evaluation under Normal Conditions of Transport

3.3.4.1 Initial Conditions

The applicant performed a thermal evaluation for the 9977, under NCT thermal conditions, with insolation applied to the surfaces of the package in 100°F still air. The insolation is based on the values given in 10 CFR 71.71(c) for a 12-hour time period. The absorptivity of the stainless steel drum surface was assumed to be 1.0, while the surface emissivity was assumed to be 0.21. The Staff concurs with these values.

3.3.4.2 Effects of Tests

The applicant performed thermal evaluations of the various packages for NCT, using numerical analyses benchmarked against experiments on the 9977 Packaging, which used a 19 W heater to simulate the content decay heat. The polyurethane foam properties were adjusted to the analytical model to duplicate the experimental results. It was found that the thermal conductivity of the foam is proportional to its density. The foam is initially introduced as a liquid into the drum and expands as it cures. The solid foam density will be controlled in the region of 15-17 lb/ft^3 by the procedure specified in Appendix 8.5 of the SARP.

3.3.4.3 Maximum and Minimum Temperatures

The minimum temperature of -40ºC in the package occurs when the decay heat load is zero watts in an environment of -40ºC. As noted in Section 3.3.1 above, the *Cold* condition of -40ºC ambient temperature will not result in degradation of the 9977, or any of its components. The 304 L austenitic stainless steels, used for the 6CV and the drum, do not have a ductile-to-brittle transition temperature above -40ºC. 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 evaluated three axial positions in the 6CV of the food-pack cans and one of the RTG, as described in Appendix 3.3. For each configuration, the applicant determined (by analysis) the steady state component temperatures for the package in the shade, as well as with insolation. The decay heat rate of 19 W was used in the analyses. The maximum component temperatures are given in Table 3.2 of the SARP, as described in Section 3.3.1.4 above. The steady-state temperatures of the components during NCT do not exceed the limits for the packaging.

3.3.4.4 Maximum Normal Operating Pressure

The MNOP in the 6CV is the result of (1) the increase in temperature of the enclosed cavity air initially at atmospheric pressure and 70°F and (2) gas generation from decomposition of 100 grams plastic, e.g., polyurethane, low density polyethylene, nylon, and/or polyvinyl chloride tape. The MNOP calculated by the applicant is given for the 6CV in Summary Table 3.3 of the SARP. This pressure, 41.2 psig, obtained for the case of RTG, is an upper bound for the 6CV with food-pack can content. As shown in Chapter 3 of the SARP, this pressure does not produce stresses in the 6CV that exceed the allowable stress limits. A review of the calculation of the MNOP confirmed that the results were reasonable and conservative. The MNOP calculation is given in Section 3.3.2 of the SARP.

3.3.4.5 Maximum Thermal Stresses

The thermal stresses in the 9977 due to the differential thermal expansions between the package components are small, as described in Section 3.4.4 of the SARP.

The Staff finds that the 6CV of the 9977 remains fully effective as a containment boundary for the payloads during NCT. The resultant deformation of the vessel will not impair the containment and criticality safety functions of the package. The Staff also finds that NCT do not impair the ability of the 9977 to withstand HAC, as discussed below.

3.3.5 Thermal Evaluation under Hypothetical Accident Conditions

3.3.5.1 Initial Conditions

The thermal evaluations of the HAC were performed on the 9977 by test and by analysis. The analyses were, after benchmarking, performed to calculate temperatures on the 9977 in pre-fire and post-fire conditions with heat generation of 19 W. The pool fire tests were carried out to determine the temperature response of the 9977 under the conditions specified in §71.73(c)(4). Since the packages were exposed to temperatures higher than 1475°F for 30 minutes, the fire tests generated conservative results.

3.3.5.2 Effects of Thermal Tests

The post-fire transient analyses were modeled as an undamaged package and used the thermal properties of the charred and uncharred polyurethane foam with three conditions noted above in Section 3.3.2.1. The convection coefficient correlations in Table 3.12 of the SARP demonstrate that the analytical boundary conditions used in the analyses were appropriate. The analytical models with the content heat located at different levels were used to calculate the highest temperature in each key component of the package.

3.3.5.3 Maximum Temperatures and Pressures

The maximum temperatures experienced by the package components during the regulatory fire are given in Table 3.2 of the SARP, and are described above in Section 3.3.1.4. Temperatures of the key components during HAC do not compromise the functions of the package.

The maximum pressure in the 6CV is due to the increase in temperature of the cavity air and gas generation from the polyurethane blocks of the RTG payload. The maximum pressure, calculated by the applicant for the RTG contents, is given for the 6CV in Tables 3.19 and 3.27 of the SARP, as described in Section 3.3.1.5 above. These pressures bound the pressures produced in the 6CV with food-pack can contents.

3.3.5.4 Maximum Thermal Stresses

As described in Section 3.4.3 of the SARP, the pressure does not produce stresses in the 6CV that exceed the allowable stress limits. A review of the calculations of the pressures produced during HAC confirmed that the results were reasonable and conservative. Also, as described in

Section 3.4.4 of the SARP, the thermal stresses in the 9977, due to the differential thermal expansions between the package components, are small.

The Staff finds that the 6CV of the 9977 remains fully effective as a containment boundary for Content Envelope C.1 during HAC. The resultant deformation of the vessel will not impair the containment or criticality safety functions of the 6CV. Thus, the functions of the 9977 with Content Envelope C.1 payloads are not affected by HAC.

3.3.6 Thermal Evaluation of Maximum Accessible Surface Temperature

The maximum accessible surface temperatures of the 9977 with the 19 W decay heat 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 temperature is less than 122°F, which is one condition for allowing the package to be transported under nonexclusive use. The Staff concurs with this analysis and conclusion.

3.3.7 Appendices

There are nine Appendices associated with Chapter 3 of the SARP. They are:

- Appendix 3.1, entitled, *9977 Thermal Benchmark Test, S-TSM-A-00001*;
- Appendix 3.2, entitled, *SRNL Package Burn Test Report, NT-TDR-06-101*;
- Appendix 3.3, entitled, *NCT Thermal Model for the 9977 Package, M-CLC-A-00255*;
- Appendix 3.4, entitled, *HAC Thermal Model for the 9977 Package*, M-CLC-A-00256;
- Appendix 3.5, entitled, *MSC/PATRAN/THERMAL Version 2003 Software Test Documentation*;
- Appendix 3.6, entitled, *PATRAN-PLUS and P/Thermal Code Descriptions*;
- Appendix 3.7, entitled, *Pressure Contribution Due to Plastic Bags*;
- Appendix 3.8, entitled, *MNOP and Maximum Operating Pressure in 9977 Package GPFP, M-CLC-A-00257*; and
- Appendix 3.9, entitled, *Determination of the Volume of LAST-A-FOAM® remaining in the 9977 Packaging after the HAC Thermal Test*.

3.3.7.1 Description of Test Facilities and Equipment

The thermal environmental chamber tests of 9977 packagings were performed at the Savannah River National Laboratory (SRNL) site. The pool fire tests were performed at the South Carolina Fire Academy. The test report, including the test plan and the procedure for the instrumented 9977 packagings, is presented in Appendix 3.2 of the SARP. During the pool fire tests, the 9977 packagings were tested without the 19 W heater present to simulate the content heat source. Adjustments were made to the measured temperatures by adding conservatively-estimated effects of the decay heat. The test report includes the measured temperature histories of various components.

3.3.7.2 Test Results

Thermal tests were performed for NCT as described in Appendix 3.1 of the SARP. These tests provided the basis for the thermal properties of the polyurethane foam and other insulating materials used in the thermal model.

Four different 9977 packagings, in different orientations, were selected for fire testing. Details are given in Appendix 3.2 of the SARP.

An investigation of the volume of Last-A-Foam® remaining in the 9977 packaging after the pool fire tests was performed as described in Appendix 3.9. By the measurement of the tested packages and performing the necessary geometric calculations, an average thickness of the remaining foam was determined. Based on this average value, three cases of possible remaining foam thicknesses were input to the analytical model to calculate the post-HAC cool-down performance of the package.

3.3.7.3 Applicable Supporting Documents or Specifications

Supporting documents of thermal models and burn test reports are listed in Appendices 3.8 and 3.9. Engineering drawings, specifications for the O-rings, and thermal benchmark test reports are referenced in Appendix 3.3. Furthermore, the American Society for Testing Materials Standard, ASTM D 4635, and an evaluation of polymer film out-gassing are referenced in Appendix 3.7.

3.3.7.4 Details of Analyses

The P/Thermal module used in the analyses of the thermal responses of the 9977s for NCT and HAC is described in Appendices 3.5 and 3.6. Included in these same Appendices are the listings of the material properties data file, the file containing the convection correlation parameters, and the radiative surface properties file. Additionally, these Appendices contain internal and solar heat source data. Benchmarking of the P/Thermal module against a documented shipping package thermal problem is presented in Appendix 3.5. The results indicate that the P/Thermal module computes the thermal response of the benchmark problem to an acceptable accuracy.

The analyses of the Model 9977 packagings using the P/Thermal module were performed to simulate the packages under post-fire conditions. The analytical model was adjusted to bring the calculated temperatures of the 9977 packaging under NCT into a sufficiently accurate correspondence with the measured results (Appendix 3.1), and into correspondence with the calculated initial HAC post-fire temperatures after superimposing the temperature changes in the components obtained from the fire tests (Appendix 3.4).

The pressure change in the 6CV, due to gases generated by the decomposition of plastic bags used with food-pack cans, was analyzed in Appendix 3.7. The MNOP in the 6CV was calculated in Appendix 3.8.

3.4 Evaluation Findings

3.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following exceptions or clarifications:

- The limiting temperatures of the bolts, the drum, and the contents need to be specified, for NCT in Table 3.5, and for HAC in Tables 3.1, 3.6 and 3.20 of the SARP. Those values should at least be equivalent to those used in the 9975, a certified package of similar design.

- The density of the Last-A-Foam® in Table 3.8 of the SARP should be updated to be consistent with the density in Drawing R-R1-G00020, Rev. 2.

- The FEA input files of materials, initial conditions, and boundary conditions are not up-to-date in Appendix 3.6 for the NCT and HAC models in Appendices 3.3 and 3.4, respectively. However, the correct input files were used for the NCT and HAC analyses. Appendix 3.6 should be updated with the current input files.

- The test procedure, *Experimental Validation of GPFP Thermal Model, FP-1034*, to Appendix 3.1 should have been attached, as is stated in that Appendix.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, the Staff has concluded that, based on their review of the statements and representations in the SARP, the Thermal Evaluation described meets the requirements of 10 CFR 71, and of IAEA Safety Standards Series No. TS-R-1.

3.4.2 Conditions of Approval

As noted for the Conditions of Approval under Chapter 1 of the SER for the 9977, the shipment of Content Envelope C.1 must conform to Table 1.2 of the SARP, and the RTG and food-pack can configurations must include a maximum decay heat limit of 19 W.

3.5 References

[3-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[3-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[3-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[3-4] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code Section III*, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NB, ASME, New York, NY (2004).

[3-5] *General Plastics Last-A-Foam® FR-3700 for Crash and Fire Protection of Nuclear Material Shipping Containers*, General Plastics Manufacturing Company, Tacoma, WA (2002).

[3-6] *MSC/PATRAN THERMAL 2003 r2*, Online Manual, MSC Software Company, Santa Ana, California.

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4.0 Containment Evaluation

This Safety Evaluation Report (SER) documents the review of Chapter 4, Containment Evaluation, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[4-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[4-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[4-3]

4.1 Areas of Review

The following elements of the Containment Evaluation Chapter were reviewed. Details of the review are provided in Section 4.3, below.

4.1.1 Description of the Containment Design

- General Considerations for Containment Evaluations
  — Fissile Type A Packages
  — Type B Packages
  — Combustible-Gas Generation
- Design Features
- Codes and Standards
- Special Requirements for Plutonium
- Special Requirements for Spent Fuel

4.1.2 Containment under Normal Conditions of Transport

- Containment Design Criteria
- Demonstration of Compliance with Containment Design Criteria

4.1.3 Containment under Hypothetical Accident Conditions

- Containment Design Criteria
- Demonstration of Compliance with Containment Design Criteria

4.1.4 Leakage Rate Tests for Type B Packages

4.1.5 Appendices (as applicable)

4.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Containment Evaluation Review of the 9977 packaging 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 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 include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by 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)]
- 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 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 ("Normal Conditions of Transport") there would be no loss or dispersal of radioactive contents. [§71.43(f)]
- The package may not incorporate a feature intended to allow continuous venting during transport. [§71.43(h)]
- A Type B package must meet the containment requirements of §71.51(a)(1) under the tests specified in §71.71 for Normal Conditions of Transport.
- A Type B package must meet the containment requirements of §71.51(a)(2) under the tests specified in §71.73 for Hypothetical Accident Conditions.
- The maximum activity of radionuclides in a Type A package must not exceed the limits of 10 CFR 71, Appendix A, Table A-1. For a mixture of radionuclides, the provisions of Appendix A, paragraph IV apply, except that for krypton-85, where an effective A_2 equal to $10A_2$ may be used. [Appendix A, §71.51(b)]
- 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)]
- For packages that contain radioactive contents with activity greater than 10^5A_2, the requirements of §71.61 must be met. [§71.51(d)]
- A Type B package containing more than 10^5A_2 must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than 1 hour without collapse, buckling, or inleakage of water. [§71.61]
- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).
- A package containing plutonium in excess of 0.74 TBq (20 Ci) must have the contents in solid form for shipment. [§71.63]

4.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Containment Evaluation Chapter of the SARP. These procedures correspond to the *Areas of Review*, listed above in Section 4.1.

4.3.1 Description of the Containment Design

4.3.1.1 General Considerations for Containment Evaluations

4.3.1.1.1 Fissile Type A Packages

This is not applicable to the submittal.

4.3.1.1.2 Type B Packages

The 9977 is a Type B Package and must satisfy the quantitative release rates specified in §71.51(a)(1) for Normal Conditions of Transport (NCT) and 71.51(a)(2) for Hypothetical Accident Conditions (HAC). For both NCT and HAC, the applicant has elected to adopt the American National Standards Institute (ANSI) N14.5-1997[4-4] definition of *leaktight* (See Section 8.2.2.1 of the SARP). (*Leaktight* is defined as being a leakage rate of air that is less than or equal to 1×10^{-7} reference cm^3/sec (i.e., ref-cm^3/sec), at an upstream pressure of 1 atmosphere and a downstream pressure of 0.01 atmosphere or less.) By adopting the ANSI N14.5 definition of *leaktight*, the applicant is not required to show any calculations to justify their position. 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 applicant.

The entire containment boundary is tested to be *leaktight* under ANSI N14.5 during acceptance testing, annually, or after any component in the containment boundary is repaired or replaced, as described in Section 8.2.2.2 of the SARP. Replacement of the Cone-Seal Gland Nut i.e., the Leak-Test Port Plug Gland Nut), Leak-Test Port Plug, or O-rings with equivalent items does not constitute a structural modification, and hence, does not require pressure testing of the Containment Vessel (6CV).

The review also verified that the 9977 does not incorporate a feature intended to allow continuous venting during transport.

4.3.1.1.3 Combustible-Gas Generation

There is no specific subsection for *Combustible Gas Generation* in Chapter 4 of the SARP. In this case the applicant has followed the format specified in the Nuclear Regulatory Commission's (NRC's) Regulatory Guide (Reg. Guide) 7.9.[4-5] In Section 1.2.2.2.1 of the SARP, however, the applicant has noted that partial pressures will develop from thermal degradation of plastics. The applicant has also noted that the amount of plastic (as low-density polyethylene, nylon, and/or polyvinyl chloride tape) is limited to 100 grams for food-pack can configuration, or 100 grams of polyurethane plastic for the Radioisotope Thermoelectric Generator configuration. The applicant has further noted the plastics do not directly contact nuclear material.

Additional discussion on the generation of gases from the decomposition of materials is provided in Section 3.3.2 of the SARP, *Maximum Normal Operating Pressures*, and Section 3.4. 3 of the SARP, *Maximum Temperatures and Pressures*, in Appendix 3.7 of the SARP, *Pressure*

Contribution Due to Plastic Bags, and in Appendix 3.8 of the SARP, *MNOP and Maximum Operating Pressure in 9977 Package GPFP*. This is in accordance with Reg. Guide 7.9 where these topics are covered in Section 3.3.2, *Maximum Normal Operating Pressure*, and in Section 3.4.3, *Maximum Temperatures and Pressure*, respectively, and in their associated appendices.

4.3.1.2 Design Features

The 6CV Closure Assembly consists of a male-female cone joint with surfaces that have been machined to matching 10-degree angles so that the Cone-Seal Plug mates with the Containment Vessel body with a maximum radial clearance of 0.0007 inch, as described in Drawing R-R2-G-00042 of Appendix 1.1 of the SARP. The sealing surfaces are machined to a 32-micro-inch finish. Two grooves are machined on the face of the Cone-Seal Plug for the O-rings. The male-closure is a two-piece assembly, made up of a conically shaped plug (the Cone-Seal Plug) and a male-threaded ring (the Cone-Seal Nut). The two components are designed to be loosely joined using a snap-type retaining ring. The loose fit permits a lubricant (e.g., KRYTOX®) to be applied between mating surfaces which minimizes friction-induced rotation between the male-female cone surfaces as torque is applied. The retaining ring ensures removal of the Cone-Seal Plug as the Cone-Seal Nut is removed.

The Cone-Seal Nut is fabricated from Nitronic® 60 stainless steel alloy, and is cut with 6½-inch 12UNS-2A external threads. The CV weldment is made from 304L stainless steel, and is cut with 6½-inch inch 12UNS-2B internal threads. The use of dissimilar metals between the Cone-Seal Nut and the vessel weldment reduces the potential for galling.

The Cone-Seal Nut is tightened to the prescribed torque value of 100 (+20/-0) ft-lb. A maximum radial clearance of 0.0007 inch, as described in Section 4.1.4 and Appendix 2.2 of the SARP, exists between the Cone-Seal Plug and the vessel body. This close fit prevents the O-rings from extruding from the grooves under high pressure. The prescribed torque prevents the Containment Vessel from opening during NCT and HAC. The 6CV assembly is pressure tested to 1,235 ± 10 psig for acceptance, as described in Section 8.1.3 of the SARP, proving that the closures cannot be opened inadvertently by internal pressure.

The containment boundary is formed by the Containment Vessel body, the Cone-Seal Plug, the Leak Test Port Plug, and the outer O-ring seal. The inner O-ring forms a test volume to qualify the outer O-ring. The inner O-ring is not credited as part of the containment boundary, but it has been verified to be as equal a barrier to the release of material as is the outer O-ring. The same size O-ring is used for both inner and outer seals. Figures 4.1 and 4.2 of the SARP define the containment boundary for the Containment Vessel Assembly and the Containment Vessel Closure Assembly, respectively. Section 1.2.1.3 and Figure 1.4 also of the SARP define the containment boundary.

Fabrication of the Containment Vessel includes two circumferential, full-penetration, complete-fusion welds as shown in Figure 4.1 and Drawing R-R2-G-00042 of Appendix 1.1. The upper, circumferential weld joins the Containment Vessel stayed head to the Schedule 40 pipe section. The lower, circumferential weld joins a standard weight pipe cap to the pipe section to close the lower end of the vessel. Since the Package can contain up to 63,300 A_2, the radioactive material packaging design is Category I under Reg. Guide 7.11.[4-6] As a result, the Containment Vessel fabrication welds comply with Section III, Subsection NB, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.[4-7] The welds are examined with liquid penetrant, and are fully radiographed.

The 6CV incorporates a static seal design using concentric elastomer O-rings fit within circumferential grooves machined into the Cone-Seal Plug as illustrated in Figure 4.2, and Drawing R-R2-G-00042 in Appendix 1.1 of the SARP. The O-rings are made of Viton® GLT or GLT-S, and have a continuous service temperature of -40°F to 400°F.[4-8], [4-9] The respective Parker Compounds are V0835-75 and VM835-75. These O-rings have been tested to heat-induced failure at 783°F. Also, the O-rings and the Containment Vessel have been shown to be *leaktight* following a 4-hour test with nitrogen at 600°F and 1,000 psig.[4-10]

No valves or pressure relief devices are incorporated into the package design. The Maximum Normal Operating Pressure (MNOP) is 41.2 psig. The 6CV design pressure is 800 psig. The corresponding content gas temperature for MNOP is 535°F as listed in Table 2.20 of the SARP. The maximum pressure developed under HAC is 35.0 psig for the 6CV. The corresponding, calculated Containment Vessel temperature for HAC is 400°F, as given in Table 2.28 of the SARP.

The review also verified that the containment system cannot be opened unintentionally, or by a pressure that might arise within the package.

4.3.1.3 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, the Structural, and the Thermal Evaluation Chapters of the SARP. The review determined that these codes or standards specified 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 Structural and Thermal Chapters of the SARP.

The review confirmed that the performance of leakage testing is in accordance with ANSI N14.5.

4.3.1.4 Special Requirements for Plutonium

The requirement of 10 CFR 71.63 is met by requiring the contents to be in solid form.

4.3.1.5 Special Requirements for Spent Fuel

This is not applicable to the submittal.

4.3.2 Containment under Normal Conditions of Transport

4.3.2.1 Containment Design Criteria

Containment under NCT is addressed in Section 4.2 of the SARP. The applicant has elected to adopt the ANSI N14.5 definition of *leaktight* for the containment boundary.

4.3.2.2 Demonstration of Compliance with Containment Design Criteria

The results of the evaluation in Section 2.6 of the SARP and the analysis in Section 3.3 of the SARP demonstrate that the containment system is not damaged, and will remain *leaktight,* during all NCT events. Consequently, the requirements of 10 CFR 71.51(a)(1) are met, and the system satisfies the containment requirement for NCT.

The contents of the 6CV include actinide oxides limited to the quantities specified in Section 1.2.2 of the SARP. To ensure the containment of the radioactive material, a *leaktight* containment criterion is established for the 6CV.

The regulatory limit for the release of radioactive material during NCT (10^{-6} A_2/h) is met by measuring the 6CV leak rates relative to the *leaktight* criterion defined by ANSI N14.5. Prototypical 6CVs used in the NCT/HAC tests were modified to include a ¼-inch threaded pipe tap in the end cap. For the pre- and post-NCT/HAC leak-rate tests, each 6CV was evacuated and backfilled with helium to one atmosphere absolute pressure and then an “evacuated envelope–gas detector” leak test (described in ANSI N14.5, Section A.5.4) was performed. For these tests, the Leak-Test Port Plug was removed from the 6CV Closure Assembly to allow for the detection of helium leakage across the Inner O-ring seal. Section 2.7, Table 2.25 and Table 2.26 of the SARP list the leak-rates that were measured before the NCT and after the HAC testing of the prototype 6CVs.

Based on the *leaktight* performance of the O-ring seals, and by comparison to the past success of the same CV design in the 9965, 9966, 9967, and 9968,[4-11] and 9975[4-12] packagings, it is concluded the *leaktight* performance is demonstrated, and satisfies the containment criterion for NCT.

4.3.3 Containment under Hypothetical Accident Conditions

The review procedures for containment under Hypothetical Accident Conditions are similar to those under Normal Conditions of Transport. Differences relevant to Hypothetical Accident Conditions are noted below.

4.3.3.1 Containment Design Criteria

Containment under HAC is addressed under Section 4.3 of the SARP. The applicant has elected to adopt the ANSI N14.5 definition of *leaktight* for the containment boundary.

4.3.3.2 Demonstration of Compliance with Containment Design Criteria

The regulatory limit for HAC is “…no escape of radioactive material exceeding a total amount A_2 in one week.”

The 6CV is tested, as described in Sections 8.1 and 8.2 of the SARP, to be *leaktight* per the ANSI N14.5 definition, and, therefore, meets the requirement of 10 CFR 71.51 for the containment of radioactive material. Because the Containment Vessel is *leaktight*, there is no calculation of an allowable leakage rate based on an Effective A_2 value.

The results of the testing and analyses described in Sections 2.7 and 3.4 of the SARP demonstrate that the containment system is not damaged and remains *leaktight* during and after the HAC. Consequently, requirements of 10 CFR 71.51(a)(2) are met, and the system satisfies the containment requirement for HAC.

4.3.4 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., $\leq 1 \times 10^{-7}$ ref cm^3/sec, under reference air leakage test conditions. This was also verified in the Acceptance Test and Maintenance Program of the SARP, in Chapter 8.

The pre-shipment leakage rate test criterion is 10^{-3} ref cm^3/sec, which is consistent with ANSI N14.5. This was also verified in the Operating Procedures of the SARP, in Chapter 7.

4.3.5 Appendices

There is one appendix associated with Chapter 4, Appendix 4.1, *Determination of A_2 for the 9977 Fissile Package Contents*.

4.4 Evaluation Findings

4.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above with the following exceptions:

- In a number of places throughout the SARP, the applicant has referred to the *Leak-Test Port Gland Nut* as the *Cone-Seal Gland Nut*. In Chapter 4, the terminology used is simply *gland nut.* The terminology should be standardized for the Leak-Test Port Gland Nut, throughout the SARP, in order to more clearly differentiate between the *Cone-Seal Nut* and *Cone-Seal Gland Nut.*

- There is no specific subsection for *Combustible-Gas Generation*, Chapter 4 of the SARP. While this may be acceptable for the contents specified in this particular application, it may/or may not be acceptable for future applications. It is recommended, therefore, that the applicant revise their submission to provide a more complete description of Combustible-Gas Generation issues, in accordance with the requirements specified in Section 4.3.1.1.3 of the Department of Energy's *Packaging Review Guide*.[4-13]

- The applicant has noted that, for the pre- and post-NCT/HAC leak-rate tests, each 6CV was evacuated and backfilled with helium to one atmosphere absolute pressure and then an "evacuated envelope–gas detector" leak test … was performed. The applicant has also noted that, for these tests, the Leak-Test Port Plug was removed from the 6CV Closure Assembly to allow for the detection of helium leakage across the Inner O-ring seal. The applicant goes on to note that Section 2.7, Table 2.25 and Table 2.26 of the SARP list the leak-rates that were measured before the NCT and after the HAC testing of the prototype 6CVs. While this may have been a valid description for prototypical containment vessels for the General Purpose Fissile Material Packaging, where the *Inner* O-ring was defined as being part of the containment boundary, it is *not* an appropriate description for the 9977 packaging, where the *Outer* O-ring is now defined as being part of the containment boundary. It is recommended, therefore, that the applicant revise their submission to provide a more appropriate description of the leakage testing performed for the pre- and post-NCT/HAC leak-rate tests for the 9977 packaging.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, the Staff has concluded that, based on their review of the statements and representations in the SARP, the Containment Evaluation described meets the requirements of 10 CFR 71 and of IAEA Safety Standards Series No. TS-R-1.

4.4.2 Conditions of Approval

There are no additional Containment-related Conditions of Approval required for this submittal.

4.5 References

[4-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[4-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[4-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[4-4] American National Standards Institute, *American National Standard for Radioactive Materials-Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, New York, New York, 10036.

[4-5] U.S. Nuclear Regulatory Commission, Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*, Revision 2 (March 2005).

[4-6] *Fracture Toughness Criteria of Base Material Ferritic Steel Shipping Cask Containment Vessels with Maximum Wall Thickness of 4 inches (0.1 m)*, Regulatory Guide 7.11, U.S. Nuclear Regulatory Commission, Washington, DC (June 1991).

[4-7] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components*, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NB, ASME, New York, NY (2004).

[4-8] Dan Ewing, *Re:V083575*, Parker Seals, O-Ring Division, Memorandum, Lexington, KY, May 18, 2007.

[4-9] *T-3 Viton O-Ring Material Evaluation and Recommendation*, EES-2006-00056, Revision 0, Savannah River National Laboratory, Savannah River Site, Aiken, SC, September 25, 2006.

[4-10] Chalfant, Gordon, *Test Summary Report-Specification 2R-Primary and Secondary Containment Vessel-High Temperature Leakage Test (U), Fall 1980*, M-TSM-A-00001, Revision 0, Westinghouse Savannah River Company, September 12, 2003.

[4-11] Westinghouse Savannah River Company, *Safety Analysis Report-Packages, USA/9965/B(U)F (DOE), USA/9966/B(U)F (DOE), USA/9967/B(U)F (DOE), and USA/9968/B(U)F (DOE),*

Packaging of Fissile and Other Radioactive Materials, DPSPU 83-124-1, Revision 2, Savannah River Site, June 1984 (Revised February 1992).

[4-12] Westinghouse Savannah River Company, *Safety Analysis Report for Packaging, Model 9975, B(M)F-85*, WSRC-SA-2002-00008, Revision 0, Savannah River Packaging Technology, Savannah River National Laboratory (December 2003).

[4-13] Department of Energy, DOE Packaging Certification Program, *Packaging Review Guide for Reviewing Safety Analysis Reports for Packaging*, Revision 3, Draft (2007).

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5.0 Shielding Evaluation

This Safety Evaluation Report (SER) documents the review of Chapter 5, Shielding Evaluation, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[5-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[5-2] and in IAEA Safety Standards Series No. TS-R-1.[5-3]

5.1 Areas of Review

The following elements of the Shielding Evaluation Chapter were reviewed. Details of the review are provided in Section 5.3.

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 Conversions
- External Radiation Levels

5.1.5 Appendices (as applicable)

5.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Shielding Evaluation Review of the 9977 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 Normal Conditions of Transport (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 Hypothetical Accident Conditions (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)]
- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).

5.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Shielding Evaluation Chapter of the SARP. These procedures correspond to the *Areas of Review*, listed above in Section 5.1.

5.3.1 Description of Shielding Design

5.3.1.1 Design Features

The 9977 is a single containment drum type package with a bolted flange closure and a right circular cylindrical Containment Vessel (CV) enclosed by Fiberfrax® and Last-A-Foam® insulation. Major materials of construction include a stainless steel (SS) overpack drum, Last-A-Foam® polyurethane insulation, aluminum Load Distribution Fixtures (LDFs), and a SS 6-inch diameter Containment Vessel (6CV). The double O-ring sealed 6CV is removable for loading and unloading. The 9977 is designed to ship radioactive contents (Content Envelope C.1, as provided in Table 1.2 of the SARP) as assemblies of Radioisotope Thermoelectric Generators or nested food-pack cans.

Neither the package geometry nor its materials of construction are specifically designed to provide neutron or gamma shielding. Dose rate attenuation is provided primarily by the distance between the source and points external to the package, with some additional attenuation provided by the materials of the CV, Fiberfrax®, Last-A-Foam®, and the drum under NCT. However, for the HAC condition, the shielding model conservatively assumed the loss of all packaging material, leaving only the CV material to provide some attenuation.

The Staff confirms that the text and sketches describing the overall design features are consistent with the engineering drawings and the models used in the shielding evaluation. The Staff concludes that the 9977 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).

5.3.1.2 Codes and Standards

The flux-to-dose-rate conversion factors are listed in Appendix 5.1, and are consistent with the American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard, ANSI/ANS 6.1.1-1977.[5-4]

5.3.1.3 Summary Table of Maximum Radiation Levels

Table 5.1 of the SARP shows the maximum radiation levels for NCT and HAC. All dose rates are within the regulatory limits for non-exclusive use.

5.3.2 Radiation Source

The contents analyzed are based on Content Envelope C.1, presented in Table 1.2 of the SARP. The source terms were conservatively calculated based on 100 grams of ^{238}Pu, which is well in excess of the allowed 30 grams of ^{238}Pu consistent with a decay heat limit of 19 W.

The SARP used a combination of the ORIGEN-S code,[5-5] and the RASTA code,[5-6] to calculate source terms. The ORIGEN-S code is part of the NRC-sponsored SCALE code package,[5-7] available through the Radiation Safety Information Computational Center. The RASTA code is a proprietary code of Westinghouse Safety Management Solutions, a subcontractor to the applicant.

5.3.2.1 Gamma Source

The gamma source term was calculated by the applicant using the RASTA code. In addition, the gamma source term from 1 gram of ^{232}U was separately calculated using ORIGEN-S at the peak 10-year decay time to independently determine the maximum allowed impurity level of this isotope in the C.1 contents.

The Staff used ORIGEN-S to determine the gamma source term. A comparison of the gamma source spectrum is presented in Figure 5.3.1.

The gamma source from the SARP compares reasonably well (within ~8%) with that calculated by the Staff in the low energy region. However, there are large discrepancies in the region between 0.6 and 0.7 MeV. There are also two peaks around 5 and 8 MeV that are not seen in the ORIGEN-S source term generated by the Staff. Despite these differences, the total source term generated by the Staff is about 7.5% higher than that in the SARP, mainly due to the higher number of gammas in the two lowest groups in the ORIGEN-S calculations that dominate the source term.

5.3.2.2 Neutron Source

The neutron source term was calculated by the applicant using the RASTA code. The Staff used a combination of ORIGEN-S and SOURCES[5-8] (which is also used in RASTA) to determine the neutron source term. All of the non-actinide impurity content associated with the proposed payload was taken to be beryllium in determining the neutron source term. This is a conservative assumption, since beryllium would bound all other impurities for the production of neutrons via the (α, n) reaction. The contribution to the neutron source term from ^{232}U is negligible when compared to its contribution to the gamma source term.

A comparison of the neutron source spectrum is presented in Figure 5.3.2.

The neutron source term generated by the Staff compares well with that in the SARP generated by the applicant, except at low energies and at very high energies (between 6 and 12 MeV). The SARP data shows larger populations at these energies. However, the discrepancy in the low energy region is not significant. In the most important region of the spectrum, between 0.1 MeV and 6 MeV, the agreement between the two independent calculations is good (within 10%). The overall source strength calculated by the Staff is 1.86×10^6 neutrons/s, compared with 2.24×10^6 neutrons/s calculated by the applicant, leading to a 20% conservatism in the applicant's neutron source term.

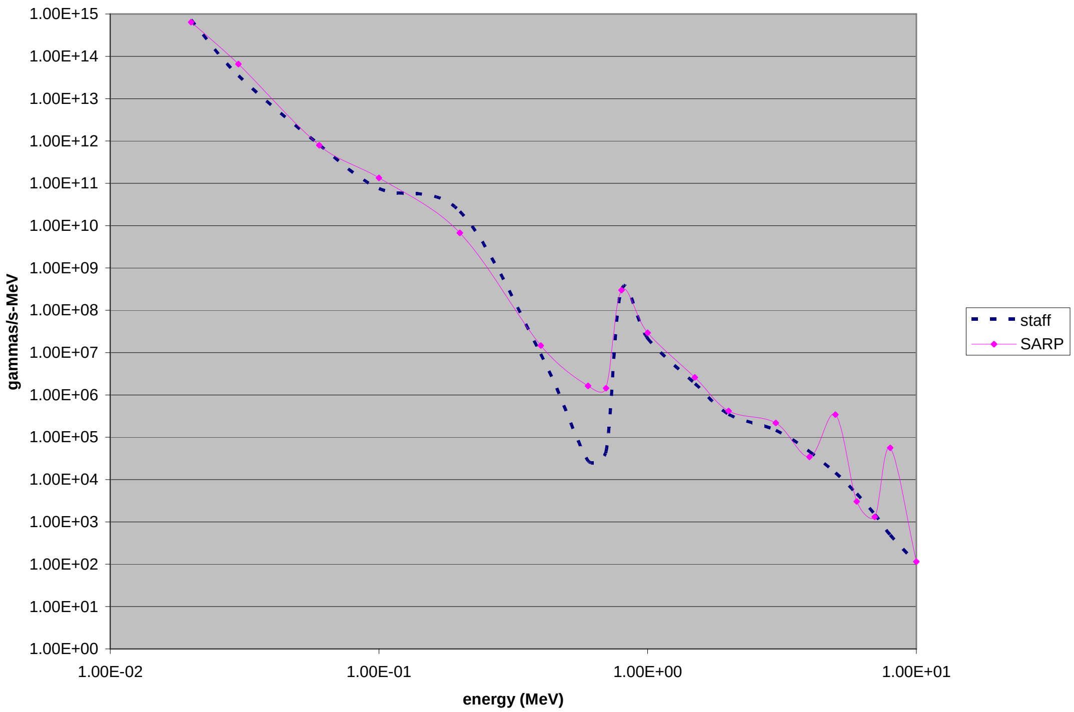

Figure 5.3.1 Gamma Source Spectrum

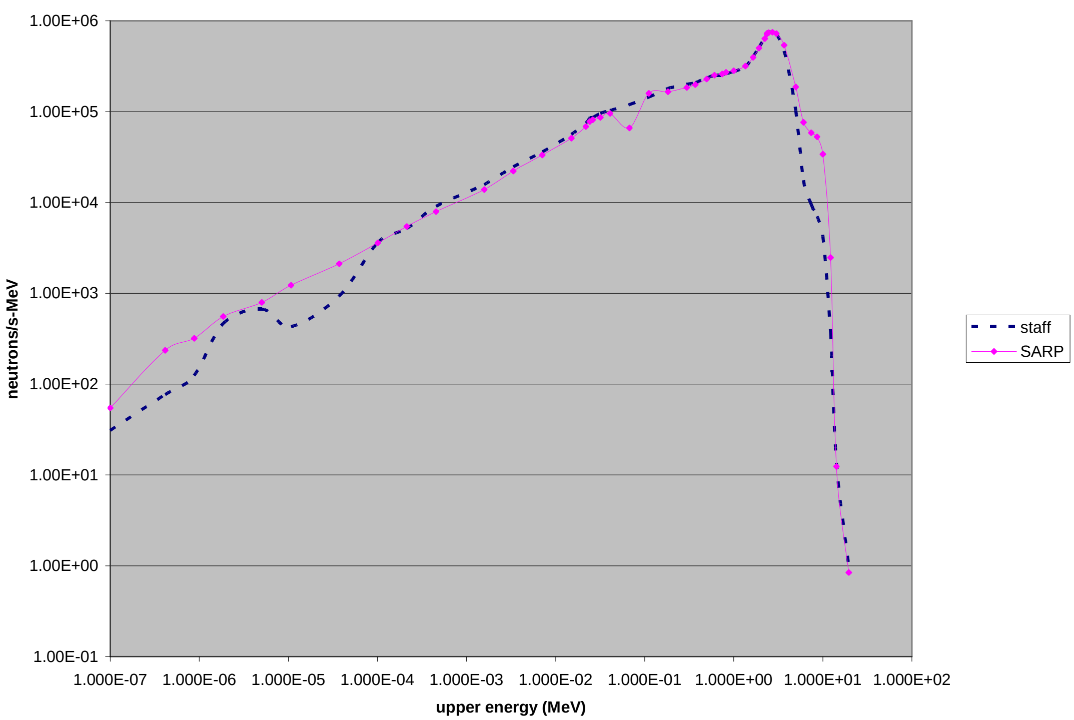

Figure 5.3.2 Neutron Source Spectrum

5.3.3 Shielding Model

The Staff concurs that the models used for the contents in Table 1.2 of the SARP in the shielding calculations are consistent with the effects of the NCT and HAC tests on the 9977.

5.3.3.1 Configuration of Source and Shielding

The dimensions of the source and packaging used in the shielding models generally 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 §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 §71.51(a)(2). All voids, streaming paths, and irregular geometries are treated in an adequate manner. Deviations from specifications in the drawings were represented in manner such that they would add to the conservatism of the model.

Surface dose rates on the package were averaged over a segment of the surface. The Staff notes that not all the surface segmentation used by the applicant's computational models capture the peak dose rates on the surfaces. Further discussions on this issue are presented in Section 5.3.4.4 of this SER.

5.3.3.2 Material Properties

Accepted values for the density of all package materials are used in the SARP. Accepted values for the source-material densities are used in the shielding calculations in the SARP. The shielding model of Content Envelope C.1 considers a homogenous source region. The Staff considers that such an approach is justified, and that the mass densities used are correct. Small deviations from the actual material properties of the Last-A-Foam® as used in the shielding analyses will have no significant effect on the final results.

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 some would survive. This is a conservative assumption, as used in the shielding analyses.

5.3.4 Shielding Evaluation

5.3.4.1 Methods

All dose rates on the 9977 for Content Envelope C.1 were determined using the three-dimensional Monte Carlo transport code MCNP.[5-9] This is a widely used code for such applications and has been referenced in the SARP. The cross sections used in MCNP were based on a combination of ENDF/B-VI and ENDF/B-V data.[5-10] The older ENDF/B-V cross section data were used for elemental iron, nickel, and chromium found in different materials in the package. Cross sections for all other materials were taken from ENDF/B-VI data.

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

Streaming paths do not play a significant role in the dose rates determined in this SARP. Although streaming paths could potentially arise in the 9977 for HAC conditions, the SARP HAC shielding model excludes all packaging materials outside the CV, essentially rendering these irrelevant.

The Staff performed calculations with the ENDF/B-VI Release 8 data to determine the sensitivity of the dose rates to the data set. A comparison of this set of calculations to the one using a combination of ENDF/B-VI and ENDF/B-V data as set forth in the SARP produced very little difference in the final results. Thus, the approach taken by the applicant is an acceptable one.

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 5.1 of the SARP. The shielding model input parameters were properly entered into the MCNP input listings in Appendix 5.1 of the SARP. No output listings are included in the SARP. However, confirmatory calculations generally verified the dose rates listed in the SARP, and established that proper convergence was achieved in obtaining these dose rates.

5.3.4.3 Flux-to-Dose-Rate Conversion

The SARP properly converts the gamma and neutron fluxes to dose rates. The flux-to-dose rate conversion factors (from ANSI/ANS 6.1.1-1977) used in the shielding calculation are properly tabulated as a function of the energy group structure in Table 5.6 of the SARP.

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 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 damage to this extent.

The limiting dose rate on the package surface was determined to be at the bottom when the source was placed at the bottom of the CV. The total of the neutron and gamma dose rates at this location was calculated by the applicant to be 141.18 mrem/hr. Confirmatory analyses were performed by the Staff for this limiting case. A comparison of the limiting bottom dose rate from the SARP and that obtained by the Staff is presented below in Table 5.3.1.

The large difference in the Staff's value and that in the SARP for the gamma dose rate is likely due to the differences in the spectra in the ~0.6–0.7 MeV region, as well as 5–8 MeV region, as discussed above in Section 5.3.2.1. Although high, this difference is not significant since the gamma dose rate is small compared to the neutron dose rate. In addition, the SARP value is more conservative than that obtained by the Staff. The neutron dose rate calculated by the Staff compares well with the SARP value.

Table 5.3.1 Limiting Dose Rate at the Bottom of the Package

Particle	Staff (mrem/hr)*	SARP (mrem/hr)	Ratio of SARP/Staff
Neutron	1.321E+02	1.283E+02	9.710E-01
Gamma	7.570E+00	1.290E+01	1.704E+00
Total**	1.397E+02	1.412E+02	1.011E+00

* Total source terms (gammas/s and neutrons/s) used to obtain the Staff's values are the same as those provided in the SARP.

** The 10 CFR 71 limit at this location is 200 mrem/hr.

As noted above in Section 5.3.3.1, the applicant's dose rate value was obtained at the bottom surface by averaging the gamma or neutron flux over a 5 cm radius disk. The Staff performed calculations to determine the sensitivity of the dose rate to averaging over different size disks, ranging from 1 mm to 5 cm, as well as calculating the dose rate at the center point of the bottom of the package. It is also noted from the Staff's calculations that the point detector estimate and the estimate from averaging over a 1 mm radius disk produced statistically the same results.

The Staff estimated that the point detector estimate for the neutron dose rate at the center bottom was about 9% higher than the estimate obtained by averaging over a 5 cm radius disk. The Staff's neutron dose rate, presented in Table 5.3.1, is that at a point in the center of the bottom of the package obtained with a point detector estimate. In the case of the gammas, a ring detector dose rate at a point at 0.25 cm from the center of the bottom gave the peak dose rate. However, this estimate of the gamma dose rate compared to the one averaged over a 5 cm disk, as is the case in the SARP, produced a ratio of 1.04. Since the neutron contribution to the dose rate is much larger than that of the gammas, the overall dose rate goes up by ~9%.

As a result of these sensitivity studies, the appropriate neutron and gamma factors were applied to the SARP values, resulting in the limiting bottom dose rate increasing to a value of 153.5 mrem/hr. This value is approximately 10% larger than the Staff's bounding value of 139.7 mrem/hr, based on the same source strengths. The Staff notes that the SARP bounding dose rate (including the upward correction applied to capture the peak dose rate) has a 23% margin to the 10 CFR 71 limit of 200 mrem/hr. The dose rates at 1 m from the bottom of the package under NCT and HAC, as well as all the relevant dose rates on the side and top, have much larger margins to the 10 CFR 71 limits. Confirmatory calculations by the Staff for the side dose rates confirm this. The Staff did not perform confirmatory calculations for the top dose rate, since these were bounded by the dose rates on the other two sides.

In addition, the Staff concurs that a limit of 0.14 mg of ^{232}U present in the contents of the package would restrict the bounding dose rate from exceeding the regulatory limits.

The 9977 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) for the Content Envelope C.1 in Table 1.2 of the SARP. The 9977 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) and §71.51(a)(1).

The 9977, with the payload given in Table 1.2 of the SARP, 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 9977, with Content Envelope C.1 in Table 1.2 of the SARP, has been shown to meet the requirements of §71.51(a)(2) of 1 rem/h at one meter from the surface of the 9977, under the tests specified in §71.73 for HAC.

The Staff agrees that the 9977 meets the requirements prescribed by 10 CFR 71, §71.43(f), §71.47(a), and §71.51(a)(2).

5.3.5 Appendices

The SARP Appendix 5.1 provides supplementary information, including selected inputs used in the analyses.

5.4 Evaluation Findings

5.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following exceptions:

- Some of the calculations by the applicant did not capture the peak dose rate on a surface, owing to averaging over less than optimal surface segments. Evaluations by the Staff have indicated that this can potentially result in an approximate 9% underestimation of the peak dose rate. The Staff also notes, however, that the models used in the applicant's evaluations have several conservatisms built into them, and the additional application of a 9% upward correction still results in a peak dose rate with sufficient margin to the regulatory limit.

- Appendix 5.1 of the SARP, in addition to analyses pertinent to Content Envelope C.1 in Table 1.2 of the SARP, contains extraneous information for other content envelopes that will not be shipped in the 9977.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, based on a review of the statements and representations in the application, and confirmatory evaluations by the Staff, 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 and IAEA Safety Standards Series No. TS-R-1.

5.4.2 Conditions of Approval

There are no additional Shielding-related Conditions of Approval required for this submittal.

5.5 References

[5-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[5-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[5-3] Regulations for the Safe Transport of Radioactive Material, 1996 Edition (As Amended 2003), Safety Requirements, Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (July 2004).

[5-4] American National Standards Institute/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.

[5-5] Hermann, O. W and Westfall, R. M., *ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms*, Version 5.0.0, Oak Ridge National Laboratory (April 2004).

[5-6] Frost, R.L., *RASTA: Radiation Source Term Analysis—Users Guide*, Westinghouse Savannah River Company Report, WSMS-CRT-97-0013 (November 1997).

[5-7] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, ORNL/TM-2005/39, Version 5, Vols. I–III (April 2005).

[5-8] 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).

[5-9] *MCNP—A General Monte Carlo N-Particle Transport Code*, Version 4C, Judith F. Briesmeister, ed. Los Alamos Report, LA-13709-M, RSICC Computer Code Collection CCC-701 (June 2001).

[5-10] *Evaluated Nuclear Data Files*, Versions B-V and B-VI, National Nuclear Data Center, Brookhaven National Laboratory (several revisions from 1973–2000).

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6.0 Criticality Evaluation

This Safety Evaluation Report (SER) documents the review of Chapter 6, Criticality Evaluation, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[6-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[6-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[6-3]

6.1 Areas of Review

The following elements of the Criticality Evaluation Chapter were reviewed. Details of the review are provided in Section 6.3, below.

6.1.1 Description of Criticality Design

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

6.1.2 Fissile Material and Other 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 Criticality Safety Index for Nuclear Criticality Control

6.1.8 Benchmark Evaluations

- Applicability of Benchmark Experiments
- Bias Determination

6.1.9 Appendices (as applicable)

6.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Criticality Evaluation Review of the 9977 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 that 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 normal conditions of transport. [§71.43(f), §71.51(a)(1), §71.55(d)(4)]
- A fissile material packaging design to be transported by air must meet the requirements of §71.55(f).
- 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 criticality safety index for nuclear criticality control to limit the number of packages in a single shipment. [§71.59(b), §71.59(c), §71.35(b)]

6.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Criticality Chapter of the SARP. These procedures correspond to the *Areas of Review,* listed above in Section 6.1.

6.3.1 Description of Criticality Design

6.3.1.1 Design Features

The 9977 is a single containment, stainless steel drum type package with a bolted flange closure. Major components of the package inside the drum are a cylindrical stainless steel 6-inch inside diameter Containment Vessel (6CV), Fiberfrax® insulation, and Last-A-Foam® insulation impact absorber. The design features of the 9977 that are important to criticality control are the loading of fissile material contents into the 6CV, which prevents their release, and the drum, which remains intact maintaining spacing between contents.

There are no other design features (e.g., neutron absorbers, flux traps, spacers) that provide criticality control in the 9977.

6.3.1.2 Codes and Standards

No specific calculations were performed by the applicant and criticality safety was established based on the following American National Standards Institute/American Nuclear Society standards: ANSI/ANS 8.15-1981,[6-4] ANSI/ANS 8.1-1998,[6-5] and ANSI/ANS 8.7-1998.[6-6]

6.3.1.3 Summary Table of Criticality Evaluation

Content Envelope C.1, given in Table 1.2 of the SARP, is dominated by ^{238}Pu, with small amounts of americium, neptunium, thorium, uranium, and other plutonium isotopes. Per Chapter 1 of the SARP, actinides, fission products, decay products, and neutron activation products, not explicitly listed, are allowed in small concentrations of less than 1000 parts-per-million. The contents are limited to a maximum fissile plutonium mass of 41 grams (40 grams of ^{239}Pu and 1 gram of ^{241}Pu) and a maximum fissile uranium mass of 40.2 grams (40 grams of ^{235}U and 0.2 gram of ^{233}U). Per ANSI/ANS-8.15-1981, the critical masses for ^{239}Pu and ^{241}Pu exceed 450 grams and 200 grams, respectively. Per ANSI/ANS-8.1-1998, the critical masses for ^{235}U and ^{233}U exceed 700 grams and 500 grams, respectively. Per ANSI/ANS-8.15-1981, the critical mass of ^{238}Pu exceeds 3 kg. The total fissile/fissionable mass allowed in Content Envelope C.1 is 100 grams, which is substantially less than the minimum critical mass of the most reactive isotope in the Content Envelope.

Therefore, for Content Envelope C.1 with a maximum of 100 grams of heat source material, criticality is not credible. A CSI of 0 has been assigned to the package.

6.3.2 Fissile Material Contents

Content Envelope C.1 can be present in the package, either as a Radioisotope Thermoelectric Generator with a maximum of 100 grams of polyurethane present, or food-pack can configuration with a maximum of 100 grams of plastic, e.g., low-density polyethylene, nylon, and/or polyvinyl chloride tape. The applicant has stated that explicit modeling of the contents for criticality safety evaluations was not necessary since criticality was not credible as discussed above in Section 6.3.1.3.

6.3.3 Considerations for Criticality Evaluations

6.3.3.1 Model Configurations

This is not applicable since no calculations were performed by the applicant.

6.3.3.2 Material Properties

This is applicable since no calculations were performed by the applicant.

6.3.3.3 Demonstration of Maximum Reactivity

Specifically, the objective of the evaluation is to demonstrate compliance with the performance requirements for Content Envelope C.1, as specified in

- 10 CFR 71.55, General requirements for fissile material packages,
- 10 CFR 71.59, Standards for arrays of fissile material packages.

The evaluation must demonstrate subcriticality based on credible scenarios, including dry and flooded conditions for a single package, Normal Conditions of Transport (NCT) arrays, and Hypothetical Accident Condition (HAC) arrays.

6.3.3.4 Computer Codes and Cross-Section Libraries

This is not applicable since no calculations were performed by the applicant.

6.3.4 Single Package Evaluation

The Staff concurs that the 9977 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

Analyses were not performed since the package contents were well below the subcritical limits.

6.3.4.2 Results

A single package will remain subcritical based on the limits of fissile materials, as listed in Table 1.2 of the SARP.

6.3.5 Evaluation of Undamaged-Package Arrays (NCT)

Per discussions in Chapters 2 and 3 of the SARP, the NCT tests did not result in any water inleakage into the containment system or damage that significantly affected the criticality safety functions of the packages. Based on independent investigations, the Staff concurs that the 9977 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 outer diameter of the 9977 under NCT is 46.6 cm. The subcritical mass of ^{239}Pu metal, in a 1,000-unit array with 45.7 cm spacing, is 3.7 kg.[6-6] The applicant has stated that, since the moisture content of Content Envelope C.1 is limited to 0.5 weight % (Table 1.2 of the SARP), the subcritical metal mass is less than the subcritical oxide mass.

The Staff notes that the subcritical mass of 3.7 kg, as noted above, is valid for the given conditions only if the H/Pu ratio is ≤ 0.01. This would not be the case if 0.5 weight % moisture is present, since this would amount to an H/Pu ratio of 0.13. However, the Staff further notes that Table 1.2 of the SARP has stated that the contents are *dry*, satisfying the requirement that the H/Pu ratio be ≤ 0.01 in order for the stated subcritical mass to be valid.

6.3.5.2 Results

The applicant has stated that, since a maximum of 100 grams of material in oxide form is present per 9977, an infinite array of 9977s with Content Envelope C.1 would be subcritical. While the Staff independently confirmed that this would be the case, it is noted that the argument presented by the applicant, based on a 1,000-unit array, does not directly lead to such a conclusion.

6.3.6 Evaluation of Damaged-Package Arrays (HAC)

Based on independent evaluations, the Staff concurs that the 9977 conforms to the HAC criticality requirements for all packages as prescribed by 10 CFR 71, §71.59(a)(2), and §71.59(a)(3).

6.3.6.1 Configuration

The outer diameter of the 9977 under HAC is 38.6 cm. The subcritical metal mass of ^{239}Pu in a 1,000-unit array with 38.1 cm spacing is 3.1 kg.[6-6] The applicant has stated that, since the material of Content Envelope C.1 is dry (as noted in Table 1.2 of the SARP), the subcritical metal mass is less than the subcritical oxide mass.

The Staff agrees with these statements, while noting that the statement that the contents will be dry, per Table 1.2 of the SARP, is in direct contradiction to the statement that up to 0.5 weight % moisture is permitted under NCT conditions (see Section 6.3.5.1 above).

6.3.6.2 Results

The applicant has stated that, since a maximum of 100 grams of material in oxide form is present per 9977, an infinite array of 9977s with Content Envelope C.1 would be subcritical under HAC. While the Staff independently confirmed that this would be the case, it is noted that the argument, based on a 1,000-unit array presented by the applicant, does not directly lead to such a conclusion.

6.3.7 Criticality Safety Index

The Staff, via independent evaluations, concurs that a CSI of 0 can be assigned to the 9977.

6.3.8 Benchmark Evaluations

Benchmark evaluations were not necessary, since no calculations were performed by the applicant.

6.4 Evaluation Findings

6.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following clarification:

> The applicant provided justification for the lack of a need to perform specific NCT and HAC array calculations by providing information on two calculations made with a 1,000-unit array model containing ^{239}Pu. The applicant did not perform any infinite array calculations to show that the arrays are subcritical for NCT or HAC, or justify the CSI value of zero. Instead, the applicant came to this conclusion from data presented in ANS Standard 8.7[6-6] for the 10×10×10 array. It should be noted that, based on a

10×10×10 array, the maximum CSI would be 0.3 (i.e., 5N = 1000, or CSI = 50/200 = 0.25, and, rounded up, this is equal to 0.3). The applicant did not explicitly provide any logical justification in extrapolating from these 1,000-unit array results to an infinite array that would lead to a CSI of 0.

However, the Staff performed independent calculations with 100 grams of ^{239}Pu (no other plutonium isotopes were included) in the package, to demonstrate that an infinite array would, indeed, remain subcritical under both NCT and HAC conditions. Based on these independent evaluations, the Staff concluded that a CSI of 0 is valid for the 9977.

The above issues notwithstanding, based on review of the statements and representations in the application, combined with their own independent evaluations, the Staff has concluded that the 9977 meets the criticality safety requirements of 10 CFR 71 and of IAEA Safety Standards Series No. TS-R-1.

Based on review of the statements and representations made in the SARP and independent evaluations by the Staff, the 9977 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 9977 has also 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.

The Staff concludes that the 9977 with Content Envelope C.1 in Table 1.2 of the SARP would meet the requirements of §71.55(b), §71.55(d), and §71.55(e), under which a single package must be subcritical, and 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.

6.4.2 Conditions of Approval

There are no additional Criticality-related Conditions of Approval required for this submittal.

6.5 References

[6-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[6-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[6-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[6-4] American National Standards Institute/American Nuclear Society, *Nuclear Criticality Control of Special Actinide Elements*, ANSI/ANS-8.15-1981, La Grange Park, IL (1981).

[6-5] American National Standards Institute/American Nuclear Society, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, ANSI/ANS-8.1-1998, La Grange Park, IL (1998).

[6-6] American National Standards Institute/American Nuclear Society, *Nuclear Criticality Safety in the Storage of Fissile Materials*, ANSI/ANS-8.7-1998, La Grange Park, IL (1998).

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7.0 Package Operations

This Safety Evaluation Report (SER) documents the review of Chapter 7, Package Operations, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[7-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[7-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[7-3]

7.1 Areas of Review

The following elements of the Package Operations Chapter were reviewed. Details of the review are provided in Section 7.3, below.

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 Preparation of Empty Package for Transport

7.1.4 Other Operations

7.1.5 Appendices (as applicable)

7.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Package Operations Review of the 9977 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)]
- The transport index of a package in a nonexclusive-use shipment must not exceed 10, and the sum of the Criticality Safety Indices (CSIs) of all packages in the shipment must not exceed 50. [§71.47(a), §71.59(c)(1)]
- 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 sum of the CSIs for nuclear criticality control of all packages in an exclusive-use shipment must not exceed 100. [§71.59(c)(2)]
- The application must include Package Operations 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 that will credibly result in the highest neutron multiplication. [§71.83]
- A package must be conspicuously and durably marked with the model number, serial number, gross weight, and package identification number. [§71.85(c), §71.19(a)(2), §71.19(b)(3)]
- 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]
- Each type B(U) or Type B(M) package design must have on the outside of the outermost receptacle a fire resistance radiation symbol in accordance with 49 CFR 172.310(d).

7.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Package Operations Chapter of the SARP. These procedures correspond to the *Areas of Review,* listed above in Section 7.1.

7.3.1 Package Loading

7.3.1.1 Preparation for Loading

The preparatory procedures for loading the package are contained in Section 7.1.1 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

- It was verified 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, the serial number, the gross weight, the package identification number, and the radiation trefoil is included.
- 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 Certificate of Compliance (CoC).

7.3.1.2 Loading of Contents

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

- Special handling equipment was specified, where appropriate.
- Special controls and precautions for loading were specified, where appropriate.
- The method of loading the contents was specified.
- Although there is no requirement to ensure that moderators or neutron absorbers are present and in proper condition, such a requirement is not necessary for shipments under the specifications for Content Envelope C.1.
- Although there is no description of the method used to remove water from the package, such a requirement is not necessary for this package.
- Although there are no descriptions of the methods used to vent excess gases or add fill gases during the loading of the 6CV, such requirements are not necessary for shipments under the specifications for Content Envelope C.1.
- 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.
- Specific requirements are in place to note that the primary containment closure (the Closure Assembly) will be torqued to 100 (+20/-0) ft-lbs, and that the Gland Nut for the Leak-Test Port Plug see — Section 7.4.1, below — will be torqued to 25±5 ft-lbs.
- 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 Appendix D of 10 CFR 835.[7-4] (Note: Since this is a Department of Energy (DOE) package, and since the shipments made under the purview of the DOE are more restrictive with respect to surface contamination limits, the requirements specified in 49 CFR 173.443[7-5] do not apply).
- 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 requirements are 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 the requirements specified in the American National Standards Institute (ANSI) document ANSI N14.5-1997.[7-6]

- 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 or set, the design of the 9977 does not incorporate the use of pressure relief devices.
- It is specifically noted that the 1-inch diameter holes located in the drum flange can be used as lifting or tie down devices, and that standard industry practice and equipment may be used.
- A specific requirement is in place to ensure that a tamper-indicating device (TID) 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 9977 does not incorporate the use of such features.
- Although there are no specific requirements that describe, for fissile material shipments, any special controls and precautions for transport, loading, unloading, handling, and any appropriate actions in case of an accident or delay, which should be provided to the carrier or consignee, such requirements are provided indirectly by the additional requirements specified in DOE Order 470.4A[7-7] and/or in site-specific/facility-specific procedures, as appropriate. (See Section 7.4.1, below.)
- 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), such requirements are provided indirectly by the additional requirements specified in DOE Order 470.4A and/or in site-specific/facility-specific procedures, as appropriate. (See Section 7.4.1, below.)
- 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), such requirements are provided indirectly by the additional requirements specified in DOE Order 470.4A and/or in site-specific/facility-specific procedures, as appropriate. (See Section 7.4.1, below.)
- A specific requirement is in place to ensure that any special instructions that should be provided to the consignee for opening the package are in place.
- The CSI for the 9977 has been determined to be 0 for Content Envelope C.1.

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 7.2.1 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 TID, 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 TID 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, such requirements are not necessary for this package.
- Specific procedures are in place that 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 7.2.2 of the SARP. The following were identified, either directly or indirectly, as being part of the operating procedures:

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

7.3.3 Preparation of Empty Package for Transport

The procedures for the preparation of an empty package 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:

- Specific procedures are in place to verify that the package is empty.
- Specific procedures are in place to ensure that the external surface contamination levels meet the requirements specified in Appendix D of 10 CFR 835. 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.
- Specific procedures are in place that describe the packaging closure requirements.
- Specific requirements are in place to note that an empty package will be shipped in accordance with the requirements specified in 49 CFR 173.428.
- Specific requirements are also in place to note that, in the case of an empty Fissile Material package, the labels and the nameplate are to be covered with tape.

7.3.4 Other Operations

The *Other Operations* described in the SARP include:

- Section 7.4.1, a section on *Packaging Storage* (for packagings that may not be used for shipment for prolonged periods of time), and

- Section 7.4.2, a section on *Records and Reporting* (that specifies that the Package Loading Record for each package will be prepared in accordance with the requirements of 10 CFR 71.91, and that they will be maintained in accordance with Section 9.17 of the SARP).

7.3.5 Appendices

There are two (2) appendices associated with Chapter 7 of the SARP:

- Appendix 7.1, entitled, *Special Tools*, provides a list of required equipment, needed for the operation of the 9977. Appendix 7.1 also provides a list of non-commercial equipment that the user may find useful for the operation of the 9977. (The applicant has also noted that illustrations and design drawings are available for all such non-commercial equipment.)
- Appendix 7.1, entitled, *EM-24 Radioactive Material Package User-Registration Form*, was provided by the applicant to register directly with EM-24, as a potential user of the 9977.

7.4 Evaluation Findings

7.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following clarifications:

- In a number of places throughout the SARP, the applicant has referred to the *Leak-Test Port Gland Nut* as the *Cone-Seal Gland Nut*. Although this may not be directly applicable to the terminology used in Chapter 7, the terminology should be standardized for the Leak-Test Port Gland Nut, throughout, in order to more clearly differentiate between the *Cone-Seal Nut* and the *Cone-Seal Gland Nut*. (Note: In Chapter 7, the terminology used is simply, *Gland Nut*.)

- Except as noted in the three (3) items in Section 7.3.1.3, above, the Staff has concluded that the applicant has met all of the regulatory requirements specified in Section 7.2 of this SER. With respect to the aforementioned items, however, the Staff has further concluded that compliance with all three of the noted items will be met through the additional requirements specified in DOE Order 470.4A, and/or in site-specific, facility-specific requirements, as appropriate.

- The EM-24 designation noted above with respect to Appendix 7.1 has been changed to EM-60.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, the Staff has concluded that, based on their review of the statements and representations in the SARP, the Package Operations described meet the requirements of 10 CFR 71 and IAEA Safety Standards Series No. TS-R-1, and are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

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 has concluded that the generic operating procedures, delineated in Chapter 7 of the SARP, should be incorporated, in their entirety, into the CoC as a Condition of package Approval.

7.5 References

[7-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[7-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[7-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[7-4] Department of Energy, 10 CFR Part 835, *Occupational Radiation Protection*; Final Rule, 63 FR 59688, pp. 59662–59689, November 4, 1998, as amended.

[7-5] Department of Transportation, 49 CFR Parts 171, 172, 173, 174, 175, 176, 177, and 178, *Hazardous Materials Regulations; Compatibility With the Regulations of the International Atomic Energy Agency; Final Rule*, 69 F.R. 3632, pp. 3632–3896, January 26, 2004, as amended.

[7-6] American National Standards Institute, *American National Standard for Radioactive Materials-Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, New York, New York, 10036.

[7-7] Department of Energy, *Safeguards and Security Program*, DOE Order 470.4A, May 25, 2007.

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8.0 Acceptance Tests and Maintenance Program

This Safety Evaluation Report (SER) documents the review of Chapter 8, Acceptance Tests and Maintenance Program, of the Safety Analysis Report for Packaging, 9977, B(M)F-96 (the SARP).[8-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[8-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[8-3]

8.1 Areas of Review

The following elements of the Acceptance Tests and Maintenance Program Chapter were reviewed. Details of the review are provided in Section 8.3, below.

8.1.1 Acceptance Tests

- Visual Inspections and Measurements
- Weld Examinations
- Structural and Pressure Tests
- Leakage Tests
- Component and Material Tests
- Shielding Tests
- Thermal Tests
- Miscellaneous Tests

8.1.2 Maintenance Program

- Structural and Pressure Tests
- Leakage Tests
- Component and Material Tests
- Thermal Tests
- Miscellaneous Tests

8.1.3 Appendices (as applicable)

8.2 Regulatory Requirements

The regulatory requirements of 49 CFR 172[8-4] and 10 CFR 71 applicable to the Acceptance Tests and Maintenance Program Review of the 9977 are as follows:

8.2.1 Acceptance Tests

- The applicant shall identify the location, on the outermost receptacle (i.e., on the outside of the package), where the package has been plainly marked with a trefoil radiation symbol that is resistant to the effects of fire and water. [49 CFR 172.310(d)]

- 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 applicant shall describe the quality assurance program for the design, fabrication, assembly, testing … and use of the proposed package. [§71.37(a)]
- The applicant shall identify any specific provisions of the quality assurance program that are applicable to the particular package design under consideration, including a description of the leak testing procedures. [§71.37(b)]
- 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)]
- Before first use, the fabrication of each packaging must be verified to be in accordance with the approved design. [§71.85(c)]
- The applicant must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]

8.2.2 Maintenance Program

- 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 applicant shall describe the quality assurance program for the … testing, maintenance, repair, modification, and use of the proposed package. [§71.37(a)]
- 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 applicant must perform any tests deemed appropriate by the certifying authority. [§71.93(b)]
- Each type B(U) or Type B(M) package design must have on the outside of the outermost receptacle a fire resistance radiation symbol in accordance with 49 CFR 172.310(d).

8.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Acceptance Tests and Maintenance Program Chapter of the SARP. These procedures correspond to the *Areas of Review,* listed above in Section 8.1.

8.3.1 Acceptance Tests

Chapter 8 of the SARP indicates that Acceptance Tests are performed prior to the first use of each package. Where applicable, sections of the Quality Assurance Program (Chapter 9 of the SARP), Package Operations (Chapter 7 of the SARP), and the appendices associated with Chapter 8 of the SARP have also been referenced.

8.3.1.1 Visual Inspections and Measurement

The applicant has noted that visual inspections and dimensional measurements are performed throughout the fabrication process to assess and verify compliance with all materials and component dimensional requirements given in the drawings. The applicant has further noted that, while all components must be fabricated of the materials and to the dimensions specified, the inspections and documentation detailed in Appendix 8.1 of the SARP, *Visual Inspection and Fabrication Verification Requirements for the 9977 Packaging*, and the independent inspection verification detailed in Appendix 8.2 of the SARP, *Packaging Independent Verification Items*, ensure that newly fabricated 9977 Packagings are complete and operable upon receipt.

8.3.1.2 Weld Examinations

The applicant has noted that a certified weld examiner shall examine specific welds in accordance with an employer's written practice, as defined in the American Society for Nondestructive Testing requirements document, SNT-TC-1A.[8-5] The applicant has further noted that the inspection methods, weld procedures, personnel qualifications, and weld reports shall meet the requirements of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (B&PV) Section V[8-6] and of ASME B&PV Code Section III,[8-7] Subsection NB or NF, as appropriate. (See Section 8.4.1, below.)

8.3.1.3 Structural and Pressure Tests

The applicant has noted that each 6-inch diameter Containment Vessel (6CV) shall be hydrostatically proof tested at an internal pressure of 1,235 ±10 psig. The applicant has further noted that: 1) at the specified test pressure, the joints, connections, and external areas of the 6CV shall have no leaks; 2) no permanent deformation of any of the 6CV parts is permitted; and 3) any 6CV not in compliance with the acceptance criteria shall be dispositioned in accordance with Section 9.15 of the SARP. Finally, the applicant goes on to note that the test pressure for the 6CV is based on the design conditions specified in Chapter 2 of the SARP, and that it meets the criteria specified in ASME Section III, Subsection NB-6200 and 10 CFR 71.85(b).

With respect to the drum and drum liner, the applicant has noted that both will be pressure tested as specified on the Drum and Liner Subassembly drawing (R-R2-G-00017). (See Appendix 1.1 of the SARP.)

8.3.1.4 Leakage Tests

The applicant has noted that the containment boundary of the 6CV will have been leak-rate tested, with helium, in accordance with the evacuated envelope method specified in Section A.5.4 of the American National Standards Institute (ANSI) document, ANSI N14.5-1997.[8-8] The testing requires that the entire Containment Vessel and its Closure Assembly be leak-tested, along with the closure for the Leak-Test Port Plug. The specifications further require that the test results must demonstrate that the leak-rate is less than 1×10^{-7} reference cm^3/s (i.e., 1×10^{-7} ref cm^3/s, which is $\sim 2 \times 10^{-7}$ cm^3/s helium), or less, in accordance with the ANSI N14.5 definition of *leaktight*. The specifications also require that the test sensitivity be at least 5×10^{-8} reference cm^3/s, which is also in keeping with the requirements specified in ANSI N14.5.

The applicant goes on to note that an example procedure for helium leak-rate testing a 9977 6CV is provided in Appendix 8.4 of the SARP. Upon completion of the helium leak-rate test, the applicant further requires that the old data label shall be removed from the drum, the applicable data shall be written onto a new label, and the new label shall be affixed to the drum adjacent to the Identification Plate. The applicant has added a final requirement that states that the drum shall display only a single label that contains all of the current data.

8.3.1.5 Component and Material Tests

<u>8.3.1.5.1 Component Tests</u>

There is no specific subsection for Component Tests. In this case, the applicant has followed the format specified in the NRC's Regulatory Guide (Reg. Guide) 7.9,[8-9] where *Component and Materials Tests* are combined in a single subsection. The applicant did, however, subdivide their subsection on *Component and Materials Tests* into three subsections: the first of these, provided in Section 8.1.5.1 of the SARP, is entitled *Valves, Rupture Discs and Fluid Transport Devices*; the second, provided in Section 8.1.5.2 of the SARP, is entitled *Gaskets*; the third, provided in Section 8.1.5.3 of the SARP, is entitled *Miscellaneous*.

Under the subheading of *Valves, Rupture Discs and Fluid Transport Devices*, the applicant has noted that the 9977 Packaging does not incorporate valves, rupture discs, or fluid transport devices.

Under the subheading of *Gaskets*, the applicant has noted that the 6CV O-ring seals are the only gaskets used in the 9977. The applicant goes on to note that performance testing of these O-rings is described in Section 8.1.4 of the SARP.

Under the subheading of *Miscellaneous*, the applicant has noted that the acceptance of fabricated packagings does not require additional testing.

<u>8.3.1.5.2 Material Tests</u>

There is no specific subsection for Material Tests. (See the related discussion above, in Section 8.3.1.5.1. See also, the related discussion in Section 8.4.1, below.)

8.3.1.6 Shielding Tests

The applicant has noted that acceptance of fabricated packagings does not require shielding integrity testing. The applicant has further noted that the packaging design does not include any features specifically credited with shielding.

8.3.1.7 Thermal Tests

The applicant has noted that performance of thermal testing is not required for acceptance of fabricated packagings. The applicant has further noted that the packaging design does not incorporate active heat transfer features, nor are passive heat transfer mechanisms particularly sensitive to normal variations in the materials of construction or fabrication methods.

8.3.1.8 Miscellaneous Tests

See the discussion above in Section 8.3.1.5.1.

8.3.2 Maintenance Program

The applicant has noted that the 9977 Packaging shall be subjected to inspections and tests annually or prior to use. The applicant has further noted that these annual activities ensure the continued and proper functioning of the packaging. The applicant has also noted that the User shall verify by direct inspection, or confirm through QA records, that the inspection and testing requirements presented in this Section are satisfied, prior to submitting a loaded package for shipment.

The applicant goes on to note that packaging subassemblies may be repaired, refurbished, or replaced using procedures prepared and approved in accordance with the QA requirements given in Section 9.15 of the SARP, and the applicable requirements of the ASME B&PV Code, Section III, Subsection NB or NF. The applicant further requires that all such repairs shall be documented in accordance with the requirements of Section 9.6 of the SARP.

8.3.2.1 Structural and Pressure Tests

The applicant has noted that the maintenance program for the 9977 Packaging does not require recurring structural or pressure tests. The applicant goes on to note, however, that pressure testing of the Containment Vessel, as specified in Section 8.1.3 of the SARP, shall be repeated after any structural modifications to, or rebuilding of, the vessel weldments, the Cone-Seal Nut or the Cone-Seal Plug. The applicant also goes on to note that replacement of the Cone-Seal Gland Nut (i.e., the Leak-Test Port Gland Nut — see Section 8.4.1, below), the Leak-Test Port Plug or the O-rings with equivalent items does not constitute a structural modification, and, hence, does not require pressure testing of the Containment Vessel.

8.3.2.2 Leakage Tests

The applicant has subdivided this subsection into two subsections. The first of these, provided in Section 8.2.2.1 of the SARP, is entitled *Pre-shipment (Post Load) Leak-Rate Test*; the second, provided in Section 8.2.2.2 of the SARP, is entitled *Maintenance Leak-Rate Test*.

Under the heading of the *Pre-shipment (Post-Load) Leak-Rate Test*, the applicant has noted that:

> "After the Containment Vessel is loaded, leak-rate tests of the Outer O-ring seal and the Leak-Test Port Plug are required to verify that the Closure Assembly has been installed

properly. The acceptance criterion is a measured leak rate less than 1×10^{-3} ref cm^3 air/sec, and the leak tests shall be capable of indicating a leak rate of 5×10^{-4} ref cm^3 air/sec or less. The leak-rate tests shall implement the pressure-rise (A.5.2) method for both the O-ring and Leak-Test Port Plug, in accordance with Section 7.6 of ANSI N14.5.

"An example of a procedure for the pre-shipment (post-load) leak-rate test is provided in Appendix 8.3 [of the SARP]."

Under the heading of the *Maintenance Leak-Rate Test*, the applicant has noted that:

"The Containment Vessel shall be leak-rate tested as specified in Section 8.1.4 [of the SARP] after any of the following events.

- Structural modifications described in Section 8.2.1 [of the SARP]
- Replacement of Containment Vessel components, including:
  - Outer O-ring seal
  - Leak-Test Port Plug
  - Leak-Test Port Gland Nut"

The applicant also noted:

"NOTE: Major modifications to or replacement of the Cone-Seal Plug, Cone-Seal Nut, or the CV weldment are not considered "Maintenance". These components may be repaired, refurbished, or replaced using procedures prepared and approved in accordance with the QA requirements given in Section 9.15 [of the SARP] and applicable requirements of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NB. All such repairs shall be documented in accordance with the requirements of Section 9.6."

The applicant then noted that:

"An annual leak-rate test is also required within the 12 months prior to presenting a loaded package for shipment. As described in Section 8.1.4 [of the SARP], the annual leak-rate test measures the rate that helium leaks through the Outer O-ring closure seal of the CV and the Leak-Test Port Plug via the evacuated-envelope method (A.5.4) of ANSI N14.5.

"Upon completion of the helium leak-rate test, applicable data shall be written onto a label, as shown in Figure 8.1 [of the SARP], and affixed to the drum as described in Section 8.1.4 [of the SARP].

"The annual leak-test report shall include the following data for each replacement O-ring:

- Material
- Size
- Date of manufacture."

(Note: Under the broader heading of the *Maintenance Program*, the applicant has noted that the annual maintenance need not be performed if the packaging is not to be placed in service within the next year.)

8.3.2.3 Component and Material Tests

8.3.2.3.1 Component Tests

There is no specific subsection for *Component Tests*. In this case, the applicant has followed the format specified in the Reg. Guide 7.9, where *Component and Materials Tests* are combined in a single subsection. The applicant did, however, note that the 9977 drum, the drum liner, and the drum closure lid are stainless steel weldments that do not require annual maintenance. The applicant also noted that the drum closure bolts are not susceptible to fatigue and do not require periodic replacement. And, the applicant noted that the Pre-loading inspection requirements described in Section 7.1.1 of the SARP will segregate out the units that need repair.

8.3.2.3.2 Material Tests

There is no specific subsection for Material Tests. (See the related discussion above, in Section 8.3.2.3.1. See also, the related discussion in Section 8.4.1, below.)

8.3.2.4 Thermal Tests

The applicant has noted that annual thermal performance is not required for the 9977 Packaging.

8.3.2.5 Miscellaneous Tests

The applicant has subdivided this section into three (3) specific subsections. The first of these, provided in Section 8.2.5.1 of the SARP, is entitled *Visual Inspection*; the second, provided in Section 8.2.5.2 of the SARP, is entitled *Shielding*; the third, provided in Section 8.2.5.3 of the SARP, is entitled *Closure Assembly Maintenance*.

Under the basic subheading of *Visual Inspection*, the applicant goes on to note that all visual inspections shall be performed with at least five (5) power magnification and bright light. The applicant also goes on to note that visual inspection of welds must be performed by an American Welding Society Certified welding inspector, with current certification, and must meet the requirements of the ASME B&PV, Section III, Subsection NB-5000 or NF-5000, as appropriate.

The applicant has further subdivided the basic subheading of *Visual Inspection*, into two additional subheadings. The first of these, provided in Section 8.2.5.1.1 of the SARP, is entitled *Sealing Surfaces*; the second, provided in Section 8.2.5.1.2 of the SARP, is entitled *O-rings*.

Under the additional subheading of *Sealing Surfaces*, the applicant notes that, prior to 6CV closure, the sealing surfaces (Figure 8.2 of the SARP) shall be visually inspected for gouges, nicks, cuts, cracks, or scratches that could affect containment performance. The applicant goes on to note that, if surface damage is found, the vessel shall be set aside and the condition documented via a Non-Conformance Report (NCR), in accordance with Section 9.15 of the SARP. The applicant further notes that, the 6CV shall not be returned to service until the NCR has been dispositioned (i.e., “reworked”, “repaired”, or “used as is”), and its closure performance has been proven acceptable by leak-rate testing described in Section 8.1.4 of the SARP.

Under this same subheading, the applicant reiterates that the Leak-Test Port Plug is part of the 6CV containment boundary. As such, the applicant goes on to note that, if the Leak-Test Port Plug is replaced, a Maintenance Leak Test must be performed per Section 8.2.2.2 of the SARP, with the new Leak-Test Port Plug installed before the 6CV can be used for shipping.

The applicant then goes on to discuss the actual maintenance requirements for the *Sealing Surfaces*, i.e., the contacting surfaces between the cone-seal nut and the cone-seal plug shall be cleaned with an approved solvent (e.g., isopropyl or ethyl alcohol, Ecolink Vortex®, or other solvent approved by the Design Agency), dried and lubricated with a thin film of KRYTOX®, or equivalent fluorinated grease as approved by the Design Agency.

Under the additional subheading of *O-rings*, the applicant notes that, prior to closure of the 6CV, the two 6CV O-ring seals shall be inspected visually for gouges, nicks, cuts, cracks, or scratches that could affect containment performance. The applicant also notes that the O-rings shall then be cleaned with ethyl or isopropyl alcohol and lubricated with a thin film of silicone high-vacuum grease as described in Section 7.1.1.1 of the SARP.

The applicant also goes on to note that, prior to the annual leak-rate test, new O-rings shall be installed in the O-ring grooves of the Cone-Seal Plug. The applicant also notes that new O-rings shall be installed when visual inspection or the post-load leak-rate tests indicate that replacement is needed. The applicant also notes that the O-rings shall be as specified on Drawing R-R2-G-00042, *6-Inch Diameter Containment Vessel Subassembly*.

The applicant then goes on to require that the certification of the O-ring material, and the size and date of manufacture, shall be furnished by the vendor with each new O-ring. The applicant also notes that the O-rings shall be individually wrapped to prevent damage in shipment and shall be labeled to ensure traceability, also noting that spare part Viton® GLT or GLT-S O-rings shall be received and stored by the shipper in accordance with SAE ARP5316,[8-10] and the *Parker O-Ring Handbook*.[8-11] The applicant finally notes that O-rings shall be no more than 30 years past their cure date when installed, and that the shipper shall be responsible for traceability of each O-ring.

The applicant then goes on to reiterate that the Outer O-ring is part of the 6CV containment boundary, and that, if the Outer O-ring is replaced, a Maintenance Leak Test must be performed per 8.2.2.2 of the SARP with the new O-ring installed before the 6CV can be used for shipping.

Finally, the applicant goes on to note that the Inner O-ring is for assembly verification testing, only, and that it may be replaced whenever it is found to be damaged because it is not part of the 6CV containment boundary.

Under the basic subheading of *Shielding*, the applicant has noted that the design of the 9977 does not incorporate shielding, and, therefore, no annual maintenance of shielding integrity is required.

Under the basic subheading of *Closure Assembly Maintenance*, the applicant has noted that, prior to the annual leak-rate test described in 8.2.2.2 of the SARP, the threaded surfaces of the Cone-Seal Nut, shall be cleaned with a suitable solvent (e.g., ethyl alcohol or Vortex®, Organic, Semi-

Aqueous Solvent or other solvent approved by the Design Agency), dried and lubricated with a thin film of KRYTOX® or equivalent fluorinated grease, as approved by the Design Agency (see Section 9.1.2 of the SARP).

The applicant also goes on to note that, the contacting surfaces between the Cone-Seal Nut and the Cone-Seal Plug shall be cleaned with a suitable solvent (e.g., ethyl alcohol or Vortex®, Organic, Semi-Aqueous Solvent or other solvent approved by the Design Agency), dried and lubricated with a thin film of KRYTOX® or equivalent fluorinated grease as approved by the Design Agency.

8.3.3 Appendices

There are five (5) appendices associated with Chapter 8 of the SARP:

- Appendix 8.1, entitled, *Visual Inspection and Fabrication Verification Requirements for the 9977 Packaging*, provides the individual visual inspection and fabrication verification criteria for the Drum Assembly, the Drum Lid Subassembly, the Drum Liner, the Drum Weldment, the Drum and Liner Weldment, the Drum Foam Installation Subassembly, the Containment Vessel Weldment, the Cone Seal Nut, the Cone Seal Plug, the Containment Vessel Subassembly, and the Containment Vessel Subassembly.
- Appendix 8.2, entitled, Packaging Independent Verification Items, provides an itemized list of Category A "Q" items, in accordance with Section 9.2.3 of the SARP.
- Appendix 8.3, entitled, *Example Pre-Shipment (Post-Load) Leak-Rate Test Procedures for the 9977 Packaging*, provides an example of a Pre-Shipment Leak-Rate Test Procedure, along with schematics, a suggested hardware list, and a set of calculational worksheets.
- Appendix 8.4, entitled, *Example Annual Leak Test Procedure for the 9977 Packaging*, provides an example of an Annual Leak Test Procedure for the 9977 Packaging.
- Appendix 8.5, entitled, *Acceptance Tests for Polyurethane Foam in the 9977 Packaging*, provides the detailed acceptance test criteria for the polyurethane foam.

8.4 Evaluation Findings

8.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above, with the following exceptions or clarifications:

- In Section 8.3.1.2, where the applicant has stated "…inspection methods, weld procedures, personnel qualifications, and weld reports…" , the ASME B&PV Code Section V and Section III, Subsection NB or NF, also includes base and weld metal characterization and weld acceptance criteria. In addition, "…weld procedure qualifications…" should replace "…weld procedures."
- In Section 8.3.1.2, the applicant has stated that the annual maintenance need not be performed if the packaging is not to be placed in service within the next year. While this may be technically correct with respect to leakage testing, it is recommended that, at a

minimum, owners/users perform a visual inspection to verify that the packaging and its components are not deteriorating over time.

- In a number of places throughout the SARP, the applicant has referred to the *Leak-Test Port Gland Nut* as the *Cone-Seal Gland Nut*. (See, for example, Section 8.1.3 of the SARP and Section 8.3.2.1 of this SER.) In order to more clearly differentiate between the *Cone-Seal Nut* and the *Cone-Seal Gland Nut*, the terminology should be standardized for the Leak-Test Port Gland Nut.

- In Sections 8.3.1.5.2 (for Acceptance Tests) and 8.3.2.3 (for the Maintenance Program), that there are no specific subsections for *Material Tests*. In Section 8.3.1.5.1 (under the heading of *Gaskets*), the applicant has stated that "…performance testing of these O-rings is described in Section 8.1.4 of the SARP." The statement is not entirely correct, in that *performance* testing of the O-rings is *not* described in Section 8.1.4 of the SARP. Additional text should be added to the SARP to differentiate between short-term characteristics, such as visual inspections and leak testing, and long-term characteristics, such as radiation resistance, compression set, etc.

- In Section 8.2.2.1 of the SARP, the applicant noted that, for the Pre-Shipment Leak-Rate test, "The acceptance criterion is a measured leak rate less than 1×10^{-3} ref cm^3 air/sec, and the leak tests shall be capable of indicating a leak rate of 5×10^{-4} ref cm^3 air/sec or less." The statement is not entirely correct, in that Section 8.4 of ANSI N14.5-1997 states that the sensitivity of the Pre-Shipment Leak-Rate test *need not be more sensitive than* 1×10^{-3} *ref·cm³/s*.

- In Chapter 9 of the SARP, the applicant has reclassified the Top and Bottom Load Distribution Fixtures (LDFs) to be Q Category A items. Should the applicant choose to retain the LDFs as Q Category A items, the LDFs should be added to Table 8.2.1 of Appendix 8.2, *Packaging Independent Verification Items*, of the SARP.

- In the example Annual Leak Test Procedure provided as Appendix 8.4 of the SARP, some of the *Prerequisite Actions* noted in Section 4.0 appear to refer to a packaging other than the 9977 Packaging.

- In the example Annual Leak Test Procedure provided as Appendix 8.4 of the SARP, a step needs to be added to the procedure to verify that the system is functioning properly *before* the leak detector is vented, and *before* the bell jar is removed from the test stand.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The above issues notwithstanding, the Staff has concluded that, based on their review of the statements and representations in the SARP, the Acceptance Tests and Maintenance Program described for the 9977 are adequate to assure that the package will be accepted and maintained in a manner consistent with its evaluation for approval. The Staff has further concluded that the Acceptance Tests and Maintenance Program described are adequate to assure packaging performance throughout its service life, and that they meet the requirements of 10 CFR 71 and of IAEA Safety Standards Series No. TS-R-1.

8.4.2 Conditions of Approval

The commitments specified in the Acceptance Tests and Maintenance Program Chapter of the SARP are typically included in the Certificate of Compliance (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 should be incorporated, in its entirety, into the CoC as a Condition of package Approval.

8.5 References

[8-1] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[8-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[8-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[8-4] Department of Transportation, 49 CFR Parts 171, 172, 173, 174, 175, 176, 177 and 178, *Hazardous Materials Regulations; Compatibility With the Regulations of the International Atomic Energy Agency; Final Rule*, 69 F.R. 3632, pp. 3632–3896, January 26, 2004, as amended.

[8-5] American Society for Nondestructive Testing, *Recommended Practice No. SNT-TC-1A — Non-Destructive Testing* (January 2001).

[8-6] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section V*, "Nondestructive Examination," New York, NY (2004).

[8-7] American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code, Section III*, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB, ASME, New York, NY (2004).

[8-8] American National Standards Institute, *American National Standard for Radioactive Material—Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, New York, New York, 10036.

[8-9] U.S. Nuclear Regulatory Commission, Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material*, Revision 2 (March 2005).

[8-10] Society of Automotive Engineers (SAE), *Storage of Elastomer Seals and Seal Assemblies which Include an Elastomer Element Prior to Hardware Assembly*, SAE ARP5316, Rev. A, SAE International, Warrendale, PA, http://www.sae.org. (2002).

[8-11] Parker Hannifin Corporation, *Parker O-ring Handbook*, ORD-5700A, Cleveland, OH, The Parker Seal Group, http://www.parker.com/o-ring (2001).

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9.0 Quality Assurance

This Safety Evaluation Report (SER) documents the review of Chapter 9, Quality Assurance, of the Safety Analysis Report for Packaging (the SARP), 9977, B(M)F-96.[9-1] The review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[9-2] and in International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1.[9-3]

9.1 Elements Reviewed

The following elements of the Quality Assurance Chapter were reviewed. Details of the review are provided in Section 9.3, below.

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

9.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Quality Assurance Review of the 9977 Packaging are as follows:

- The application must describe the QA 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 QA program. [§71.31(c)]
- Package activities must be in compliance with the QA requirements of Subpart H (§71.101–§71.137). A graded approach is acceptable. [§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(d). Records must be retained for three years after the life of the packaging. [§71.91(b)]
- 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(c)]
- Any significant reduction in the effectiveness of a packaging during use must be reported to the certifying authority. [§71.95(a)(1)]
- 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(a)(2), [§71.95(c)(4)]
- Instances in which a shipment does not comply with the conditions of approval in the Certificate of Compliance (CoC) must be reported to the certifying authority. [§71.95(a)(3)]

9.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Quality Assurance Chapter of the SARP. These procedures correspond to the *Areas of Review*, listed above in Section 9.1.

9.3.1 Description of Applicant's QA Program

9.3.1.1 Scope

Chapter 9 of the SARP was reviewed to confirm that it explicitly states that the QA program complies with 10 CFR 71, Subpart H, and is applied to package-related activities, including procurement activities consistent with the applicable regulatory requirements. The introductory text to Chapter 9, *Purpose and Scope*, describes the QA requirements for the design, procurement, fabrication, handling, shipping, storage, cleaning, assembly, use, inspection, acceptance testing, maintenance, repair, and modification of the 9977 Packaging that comply with 10 CFR 71, Subpart H, and that are important to safety. Section 9.1 of the SARP describes the applicant's organization, including the QA organizations and their responsibilities relative to implementation of the QA program. The applicant purchases 9977 Packaging materials, equipment, and services from suppliers that have been evaluated and approved to meet the applicable elements of American Society of Mechanical Engineers (ASME) NQA-1-2004 (NQA-1).[9-4]

9.3.1.2 Program Documentation and Approval

As required by §71.31(a)(3) and §71.37, Section 9.2.1 of the SARP identifies that the Westinghouse Savannah River Company (WSRC) Quality Assurance Management Plan,[9-5] documents the QA program that complies with 10 CFR 71, Subpart H, as well as 10 CFR 830, Subpart A,[9-6] DOE O 414.1.C,[9-7] DOE O 460.1B,[9-8] and NQA-1. (The WSRC Quality Assurance Manual (WSRC 1Q Manual),[9-9] identifies the procedures for implementing the WSRC QA Management Plan. Additional information on the hierarchy and relationship of requirements documents, the WSRC QA Management Plan, and implementing procedures is provided in Figure 9.2 of the SARP. The current revision and date of the applicable WSRC QA documents are provided in the references section in Chapter 9 of the SARP.

9.3.1.3 Summary of 18 Quality Assurance Requirements from 10 CFR 71, Subpart H

The twenty WSRC 1Q Manual sections (that include the quality implementing procedures) implementing each of the 18 QA requirements of 10 CFR 71, Subpart H are listed and summarized in Table 9.1 of the SARP. Chapter 9 describes the provisions in the WSRC 1Q Manual sections, as they apply to the scope of the applicant's responsibilities, identified in Section 9.3.1.1, above.

9.3.1.4 Cross-Referencing Matrix

Table 9.1 of the SARP provides a cross-referencing matrix that links each of the WSRC 1Q Manual sections to the corresponding QA requirement(s) in 10 CFR 71 Subpart H. A direct correlation exists between the 18 QA requirements of Subpart H and the sections of WSRC 1Q Manual, with the exception of WSRC 1Q Manual Sections 19 and 20. Section 19, *Quality Improvement* is identified as an extension of Section 15, *Control of Nonconforming Items*, and Section 20, *Software QA*, is identified as an extension of Section 3, *Design Control*.

9.3.2 Package-Specific QA Requirements

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

Per §71.101(b), Section 9.2.3 of the SARP describes the graded application of the WSRC Quality Assurance Manual to package structures, systems, and components (SSCs) that are important to safety. Safety-related "Q" package components are categorized as A, B, or C, with Category A items having the largest impact on safety. Table 9.2 of the SARP correlates the WSRC Safety Designations for "Q" and "non-Q" (not related to safety) for the 9977 Packaging to the safety designations in the NRC's Regulatory Guide 7.10.[9-10]

Packaging SSCs and their WSRC safety categories, functions, and drawing number are provided in Table 9.3 of the SARP.

Table 9.4 of the SARP identifies the graded level of QA controls that apply to the WSRC A, B, and C safety categories, consistent with the requirements in §71.101(b) and the guidance in Reg. Guide 7.10.

9.3.2.2 Package-Specific Quality Criteria and Package Activities

Per §71.31(a)(3) and §71.37, the SARP describes the QA controls in each section of the WSRC 1Q QA Manual listed in Table 9.1, and describes how these controls are applied to WSRC 9977 activities related to the design, procurement, fabrication, handling, shipping, storage, cleaning, assembly, use, inspection, acceptance testing, maintenance, repair, and modification of the 9977 Packaging. The graded approach, described in Section 9.3.2.1 above, is used to selectively apply the QA controls to package SSCs based on their importance to safety.

As required by §71.31(a)(3) and §71.37, Table 9.5 of the SARP details the materials, design, fabrication, testing, examination, QA program and records requirements for the Containment Vessel that conform to Section III, Division 1, Subsection NB, of the ASME Boiler and Pressure Vessel Code (B&PV).[9-11] Table 9.6 of the SARP details the materials, design, fabrication, examination, QA program and records requirements for the drum's bolted closure that conform to Section III, Division 1, Subsection NF, of the ASME B&PV Code.[9-12]

Section 9.6 of the SARP identifies documents that are controlled to ensure correct documents are used, and that records requirements are met. Controlled documents include operating procedures (SARP Chapter 7), procurement documents (SARP Section 9.4), and the inspection (SARP Section 9.10), testing, and maintenance documents (SARP Chapter 8 and Section 9.11 of the SARP).

Section 9.15 defines the controls for documenting, resolving, and preventing the recurrence of package-related nonconformances. Section 9.15 also includes provisions for obtaining WSRC Design Authority and Design Agency approval of nonconformance dispositions, and reporting package defects that significantly reduce safety performance of the package to the DOE Certifying Authority in accordance with §71.95.

Section 9.17 summarizes the provisions for ensuring sufficient written records are maintained to furnish evidence of the quality of the 9977 Packaging. The records and their retention requirements, identified in Section 9.17 and Table 9.7 of the SARP, are consistent with §71.85, §71.91(b), and §71.91(d).

Section 9.19 of the SARP includes a list of references used in Chapter 9.

9.3.3 Appendices

There are no appendices associated with Chapter 9 of the SARP.

9.4 Evaluation Findings

9.4.1 Findings

The Staff is in general agreement with the statements and conclusions for each of the sections noted above with the following clarification:

> The applicant has reclassified the Top and Bottom Load Distribution Fixtures (LDFs) to be Q Category A items. If the applicant chooses to retain the LDFs as Q Category A, then the LDFs should be added to Table 8.2.1 of Appendix 8.2, *Packaging Independent Verification Items*, of the SARP.

The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.

The aforementioned issue notwithstanding, based on review of the statements and representations in the SARP, the Staff concludes that the applicant's QA program has been adequately described and meets the QA requirements of 10 CFR 71 and IAEA Safety Standards Series No. TS-R-1. 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.

9.4.2 Conditions of Approval

Any organization involved in the design, procurement, fabrication, handling, shipping, storage, cleaning, assembly, operation, inspection, testing, maintenance, repair, modification, and use of the 9977 Packaging shall maintain and follow an appropriate QA program that is compliant with the requirements specified in 10 CFR 71, Subpart H. For non-WSRC users, this shall include compliance with the package-specific QA requirements specified in Chapter 9 of the SARP.

The package-specific QA requirements, specified in the SARP, should therefore be incorporated into the CoC as a Condition of package Approval.

9.5 References

[9-1] Westinghouse Savannah River Company, *Safety Analysis Report for Packaging, Model 9977, B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River Packaging Technology, Savannah River National Laboratory (August 2007).

[9-2] Nuclear Regulatory Commission, 10 CFR Part 71, *Compatibility with IAEA Transportation Standards (TS-R-1) and Other Transportation Safety Amendments; Final Rule*, 69 F.R. 3698, pp. 3698–3814, January 26, 2004, as amended.

[9-3] *Regulations for the Safe Transport of Radioactive Material—2005 Edition—Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, International Atomic Energy Agency, Vienna, Austria (April 2005).

[9-4] American Society of Mechanical Engineers, *Quality Assurance Program Requirements for Nuclear Facilities*, ASME NQA-1-2004, ASME, New York, NY (December 2004).

[9-5] *Quality Assurance Management Plan*, WSRC-RP-92-225, Revision 13, Westinghouse Savannah River Company, Aiken, SC (August 2004).

[9-6] Department of Energy, 10 CFR Part 830, *Nuclear Safety Management*, 66 F.R. 1818, pp. 1818-1827, January 10, 2001, as amended. See, in particular, Subpart A, *Quality Assurance Requirements*, pp. 1820–1821, as amended.

[9-7] *Quality Assurance*, DOE O 414.1C, U.S. Department of Energy, Washington, DC (June 2005).

[9-8] *Packaging and Transportation Safety*, DOE O 460.1B, U.S. Department of Energy, Washington, DC (April 2003).

[9-9] *Quality Assurance Manual*, WSRC-1Q, Westinghouse Savannah River Company, Aiken SC (October 2005).

[9-10] *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*, Regulatory Guide 7.10, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC (March 2005).

[9-11] *Rules for Construction of Nuclear Facility Components*, Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NB, "Class I Components," American Society of Mechanical Engineers (2004).

[9-12] *Rules for Construction of Nuclear Facility Components*, Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, "Supports," American Society of Mechanical Engineers (2004).

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