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LAWRENCE LIVERMORE NATIONAL LABORATORY

Technical Review Report for the Model 9978-96 Package Safety Analysis Report for Packaging (S-SARP-G-00002, Revision 1, March 2009)

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Disclaimer

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Prepared for
Packaging Certification Program
Office of Safety Management and Operations
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OVERVIEW

This Technical Review Report (TRR) documents the review, performed by Lawrence Livermore National Laboratory (LLNL) Staff, at the request of the Department of Energy (DOE), on the *Safety Analysis Report for Packaging (SARP), Model 9978 B(M)F-96*, Revision 1, March 2009 (S-SARP-G-00002). The Model 9978 Package complies with 10 CFR 71, and with *Regulations for the Safe Transport of Radioactive Material—1996 Edition (As Amended, 2000)—Safety Requirements*, International Atomic Energy Agency (IAEA) Safety Standards Series No. TS-R-1. The Model 9978 Packaging is designed, analyzed, fabricated, and tested in accordance with Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PVC).

The review presented in this TRR was performed using the methods outlined in Revision 3 of the 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 TRR, where appropriate.

The Model 9978 Packaging is a single containment package, using a 5-inch containment vessel (5CV). It uses a nominal 35-gallon drum package design. In comparison, the Model 9977 Packaging uses a 6-inch containment vessel (6CV). The Model 9977 and Model 9978 Packagings were developed concurrently, and they were referred to as the General Purpose Fissile Material Package, Version 1 (GPFP). Both packagings use General Plastics FR-3716 polyurethane foam as insulation and as impact limiters. The 5CV is used as the Primary Containment Vessel (PCV) in the Model 9975-96 Packaging. The Model 9975-96 Packaging also has the 6CV as its Secondary Containment Vessel (SCV). In comparison, the Model 9975 Packagings use Celotex™ for insulation and as impact limiters.

To provide a historical perspective, it is noted that the Model 9975-96 Packaging is a 35-gallon drum package design that has evolved from a family of packages designed by DOE contractors at the Savannah River Site. Earlier package designs, i.e., the Model 9965, the Model 9966, the Model 9967, and the Model 9968 Packagings, were originally designed and certified in the early 1980s. In the 1990s, updated package designs that incorporated design features consistent with the then-newer safety requirements were proposed. The updated package designs at the time were the Model 9972, the Model 9973, the Model 9974, and the Model 9975 Packagings, respectively. The Model 9975 Package was certified by the Packaging Certification Program, under the Office of Safety Management and Operations.

The Model 9978 Package has six Content Envelopes: C.1 (^{238}Pu Heat Sources), C.2 (Pu/U Metals), C.3 (Pu/U Oxides, Reserved), C.4 (U Metal or Alloy), C.5 (U Compounds), and C.6 (Samples and Sources). Per 10 CFR 71.59 (Code of Federal Regulations), the value of *N* is 50 for the Model 9978 Package leading to a Criticality Safety Index (CSI) of 1.0. The Transport Index (TI), based on dose rate, is calculated to be a maximum of 4.1.

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Contents

Acronyms and Abbreviations

5CV	5-inch Inside Diameter Containment Vessel
6CV	6-inch Inside Diameter Containment Vessel
9977	General Purpose Fissile Package
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing Materials
B&PVC	Boiler & Pressure Vessel Code
BD	Bottom Down
CFR	Code of Federal Regulations
CG	Center of Gravity
CGOT	Center of Gravity over Top
CoC	Certificate of Compliance
CSI	Criticality Safety Index
CV	Containment Vessel
DOE	U.S. Department of Energy
DOT	U.S. 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
Ft-lb	ft-pound
GPFP	General Purpose Fissile Package (working name for the 9977)
HAC	Hypothetical Accident Conditions
IAEA	International Atomic Energy Agency
ICE	Isentropic Compression Experiment
ID	Inside Diameter
k_{eff}	effective multiplication factor
kPa	kiloPascals
k_{safe}	multiplication factor
LDF	Load Distribution Fixture
LLNL	Lawrence Livermore National Laboratory
MCNP	Monte Carlo n-Particle Transport Code
MNOP	Maximum Normal Operating Pressure
mPa	megaPascals
mrem	milliroengten equivalent man
M/PT	MSC/PATRAN-THERMAL®

Acronyms and Abbreviations (cont.)

MSM	Minimum Subcriticality Margin
mSv	millisieverts
NCR	Nonconformance Report
NCT	Normal Conditions of Transport
NRC	U. S. Nuclear Regulatory Commission
OD	Outer Diameter
OST	Office of Secure Transportation
PCV	Primary Containment Vessel
ppb	parts-per-billion
ppm	parts-per-million
PRG	Packaging Review Guide
psi	pressure in pounds per square inch
psia	absolute 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
rem	roentgen equivalent man
SAE	Society of Automotive Engineers
SARP	Safety Analysis Report for Packaging
SCALE	Standardized Computer Analyses for Licensing Evaluations
SGT	Safe-Guards Transporter
SCV	Secondary Containment Vessel
SNT	American Society for Nondestructive Testing
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
SS	SS
SSC	Systems, Structures, and Components
SST	Safe Secure Trailer
TBq	Terabequerels
TD	Top Down
TI	Transport Index
TID	Tamper-Indicating Device
TIL	Temperature Indicating Labels
TRR	Technical Review Report
UN	United Nations
WSMS	Washington Safety Management Solutions
WSRC	Washington Savannah River Company, LLC

1.0 General Information Review

1.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[1-1] and in IAEA Safety Standards Series No. TS-R-1.[1-2] The description and engineering drawings in Chapter 1, General Information Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[1-3] were reviewed. The review also addresses Content Envelopes C.1 through C.6, excluding C.3 (reserved), as described in Table 1.2 of the SARP.

The following elements of the General Information Review 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 requirements of 10 CFR 71 applicable to the General Information review of the Model 9978 Package include:

- An application for package approval must be submitted in accordance with Subpart D of 10 CFR 71. [§71.0(d)(2)]
- An application for modification of a previously approved package is subject to the provisions of §71.19 and §71.31(b). All changes in the conditions of package approval must be approved. [§71.19, §71.31(b), §71.107(c)]
- 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)]
- An application for renewal of a previously approved package must be submitted no later than 30 days prior to the expiration date of the approval to assure continued use. [§71.38]
- The 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 for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 1.1 of this TRR.

1.3.1 Introduction

1.3.1.1 Purpose of Application

The Model 9978 Package is docketed as a new package under the current submittal. The purpose of the application is to document that the Model 9978 Package under Revision 1 of S-SARP-G-00002 satisfies the regulatory requirements of 10 CFR 71 and the *Regulations for the Safe Transport of Radioactive Material—1996 Edition—Safety Requirements*, IAEA Safety Standards Series TS-R-1.

The Model 9978 Package is designed to ship radioactive contents in arrangements of food-pack cans, DOE-STD-3013[1-4] containers, site-specific engineered containers, or Isentropic Compression Experiment (ICE) container assemblies. Some of these container configurations are designed and tested to provide confinement during handling and storage, but their ability to maintain containment during transport is not credited in this SARP. The Model 9978 Package has six Content Envelopes: C.1 (^{238}Pu Heat Sources), C.2 (Pu/U Metals), C.3 (Pu/U Oxides, Reserved), C.4 (U Metal or Alloy), C.5 (U Compounds), and C.6 (Samples and Sources).

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 Model 9978 Package is designed to transport actinide oxides, metals, alloys, or sources in excess of Type A quantities. The package is designed for an internal pressure of 900 psig at 300°F. The package type and model number, i.e., Model 9978 B(M)F-96, is provided on Drawing R-R2-G-00044, Revision 0, 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 DOE Order 470.4A.[1-5] Package users may also ship Model 9978 Packages in accordance with *exclusive use* dose-rates via the SST or the SGT as long as they have prior written approval from the DOE Office of Secure Transportation (OST). The Model 9978 is not authorized for shipments by air.

The Model 9978 Package is designated as “B(M)” package because the design pressure of the containment vessel is greater than 100 psig (700 kPa). Package contents include actinide oxide and metals in Type B quantities, and because the contents can exceed 3,000 A_2, the Model 9978 is considered a Category I package.[1-6, 1-7]

Section 1.2, *Package Description*, of the SARP includes a summary of the design criteria for the package. The American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Section III, Subsection NB was used to design, analyze, fabricate, and examine the 5CV.[1-8] The drum is designed, analyzed, and fabricated in accordance with Section III, Subsection NF of the ASME B&PVC.[1-9] General Plastics FR-3716 polyurethane foam (also known as Last-A-Foam®, with a density of 16 lb/ft^3) fills the volume between the drum wall and the Fiberfrax® insulation protecting the drum liner. The polyurethane foam acts both as insulation and an impact limiter. Additional discussion of applicable codes and standards is included in Chapters 2 through 9 of this TRR.

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 CSI for the package is 1.0. The TI, based on the estimate dose rate, is calculated to be a maximum of 4.1, and shall be established for each package at the time of shipment.

1.3.1.3 Statement of Compliance

There is a statement in Section 1.2.5, *Compliance with 10 CFR 71*, of the SARP indicating that the Model 9978 Package 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 Model 9978 Package with the regulatory safety requirements of 10 CFR 71. Criticality requirements are covered in Section 1.2.5.6, *General Requirements for and Standards for Arrays of Fissile Material Packages–10 CFR 71.55 and 71.59*. General requirements for all packages are covered in Section 1.2.5.2, *General Requirements for All Packages–10 CFR 71.43*. Structural requirements for lifting and tie-down devices are given in Section 1.2.5.3, *Structural Requirements for Lifting and Tie-Down Devices–10 CFR 71.45*. External radiation requirements are in Section 1.2.5.4, *External Radiation Requirements–10 CFR 71.47*. Requirements for Type B packages are in Section 1.2.5.5, *Requirements for All Type B Packages–10 CFR 71.51*. Special requirements for plutonium-containing packages are covered in Section 1.2.5.7, *Special Requirements for Plutonium Packages–10 CFR 71.63*. Structural and thermal performance is given in Section 1.2.5.1, *Structural and Thermal Performance Under Testing for NCT 10 CFR 71.71 and HAC 10 CFR 71.73*. Requirements for operating controls and procedures are listed in Section 1.2.5.8, *Requirements for Operating Controls and Procedures–10 CFR 71 Subpart G*. Requirements for quality assurance are located in Section 1.2.5.9, *Requirements for Quality Assurance–10 CFR 71 Subpart H*.

1.3.2 Package Description

1.3.2.1 Packaging

The Model 9978 Packaging is shown in Figures 1.1 through 1.4 and Drawing R-R1-G-00021, *Assembly with Five Inch Diameter Containment Vessel*. The packaging outer container is a 35-gallon drum, meeting the performance requirements of 49 CFR 178[1-10] for an open-head drum, modified with a bolted-flange closure. The drum shell and liner are fabricated of 18-gauge (0.048-inch) Type 304L SS (SS). The drum is designed, analyzed, and fabricated in accordance with Section III, Subsection NF of the ASME B&PVC. 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 ¼-inch-diameter vent holes are located on the bottom of the drum. All of the holes are fitted with appropriately sized Caplug® fusible plastic plugs.

The top portion of the drum incorporates a 3/16-inch-thick reinforcing rim (vertical flange) that 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, as well as drain holes. The drum shell includes a “sanitary” style drum bottom, which incorporates a radiused edge butt welded to the side wall. The drum bottom includes a rolled “wear ring,” 0.060-inch-thick (16 gauge) by ¾-inch inside diameter, attached by welds that are external to the drum shell. All external dimensions of the package are greater than four inches.

The drum closure lid is fabricated from 1/8-inch-thick Type 304L SS plate. Eight 5/8-inch by 1¼-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 receive Tamper-Indicating Devices (TIDs). The Drum Lid Top and Drum 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. In an HAC fire event, the plugs combust or melt, allowing the lid to vent heated gases from the Lid Top and Lid Bottom chambers.

1.3.2.1.1 Shielding Features

Dose-rate attenuation is provided primarily by the distance between the source and points external to the package.

1.3.2.1.2 Criticality Control Features

The Model 9978 Package does not incorporate materials specifically for the purpose of poisoning or moderating neutron radiation.

1.3.2.1.3 Insulation

Two layers of insulation material fill the volume between the drum liner and shell. 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 two 1-inch-diameter fill holes in the drum bottom and foamed in place. 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 inches thick by 8 inches diameter. When installed, this disk compresses two 8-inch-diameter by ½-inch-thick blankets of Fiberfrax® insulation to a total thickness of ½ inch.

1.3.2.1.4 Load Distribution Fixtures

The Top and Bottom Load Distribution Fixtures (LDFs) are made from 6061-T6 aluminum round bar, which fit within the Drum Liner cavity above and below the 5CV. Aluminum honeycomb Spacers fill the volume between the 5CV and the LDFs. The LDFs and Spacers stiffen the package in the radial direction and distribute loads away from the 5CV.

1.3.2.1.5 Primary Containment Vessel

The Model 9978 Packaging is designed with a containment vessel with a nominal inner diameter (ID) of five inches. The 5CV is an SS pressure vessel designed, analyzed, fabricated, and examined in accordance with Section III, Subsection NB of the ASME B&PVC. The 5CV is fabricated from 5-inch, Schedule 40 Type 304L SS pipe (0.258-inch nominal wall). A corresponding standard Schedule 40 Type 304L SS pipe cap (also 0.258-inch nominal wall) is welded to the pipe to form a blind end. A stayed head is machined from a Type 304L SS bar and welded to the open end of the pipe segment, completing the vessel body weldment. The stayed head is machined to include 5½-12UN-2B internal threads and an internal cone-seal surface with a 32-micro-inch finish. A support skirt for supporting the 5CV vertically is formed from a short segment of 4-inch, Schedule 40 Type 304L SS pipe welded to the convex side of the cap.

Both vessel body joints are Category B with full-penetration, complete fusion, circumferential welds.

The 5CV 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 with a snap-ring that also ensures unseating of the closure seal during disassembly. Both internal and external sealing surfaces are machined to the same angles, with the same surface finishes, and with matched 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 finish. Viton® GLT/GLT-S O-rings fit into these grooves to complete the *leaktight* closure assembly. A 0.094-inch-diameter vent 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 the 5CV 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 the Leak-Test Port Plug. The vessel containment boundary is formed by the vessel body, the Cone-Seal Plug, the 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 Model 9978 Package. There are six Content Envelopes allowed for the Model 9978 Package, although one of these is reserved for future use, as described in Table 1.2 of the SARP and its associated notes. The Content Envelopes, designated C.1 through C.6, are defined as follows: C.1 (^{238}Pu Heat Sources), C.2 (Pu/U Metals), C.3 (Pu/U Oxides, Reserved), C.4 (U Metal or Alloy), C.5 (U Compounds), and C.6 (Samples and Sources). The maximum allowable radioactive decay heat is 19 W. Small concentrations (<1000 ppm) 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 50 lb. The Maximum Normal Operating Pressure (MNOP) for the 5CV is 56.3 psig.

Contents containers are detailed in Section 1.2.2.1, *Contents Containers*, of the SARP. The following containers are allowed for the Model 9978 Package:

- 3013 storage containers.
- Food-pack cans.
- Engineered containers.
- ICE container assemblies.

Table 1.3 of the SARP provides a summary of requirements by Content Envelope and container configuration.

1.3.2.3 Special Requirements for Plutonium

All Model 9978 Package plutonium contents must be in solid form, regardless of whether they exceed 20 Ci.

1.3.2.4 Operational Features

All components are handled manually. Rectangular notches are located in the base of the 5CV support skirt. The notches can be used to prevent rotation of the vessel while installing/removing the closure assembly. Four 0.31-inch-diameter by ¼-inch-deep holes are located in the square drive on top of the Cone Seal Nut to facilitate lifting the Closure Assembly or the fully assembled 5CV. Special tools that may be useful for lifting and other package operations are discussed in the SARP Appendix 7.1, *Special Tools*.

1.3.3 Appendices

1.3.3.1 Drawings

Drawings of the Model 9978 Packaging are provided in Appendix 1.1 of the SARP as follows:

Drawing Number	Revision	Title
R-R5-G-00003	0	*9978—General Purpose Fissile Packaging Drawing Tree*
R-R1-G-00021	4	*9978—General Purpose Fissile Packaging Assembly with 5-inch Diameter Containment Vessel*
R-R4-G-00033	4	*9978—General Purpose Fissile Packaging Spacer Part Details for Five Inch Containment Vessel*
R-R4-G-00034	0	*9978—General Purpose Fissile Packaging Load Distribution Details*
R-R2-G-00043	3	*9978—General Purpose Fissile Packaging Five Inch Diameter Containment Vessel (CV) Subassembly*
R-R2-G-00044	1	*9978—General Purpose Fissile Packaging Drum and Liner Subassembly*
R-R2-G-00045	0	*9978—General Purpose Fissile Packaging Drum Lid Subassembly*
R-R2-G-00046	0	*9978—General Purpose Fissile Packaging Insulating Blanket Subassembly*

Drawing R-R2-G-00044, Revision 0, Section 1.2.1.8, and Figure 7.4 of the SARP indicate that the Model 9978 Package is compliant with §71.85(c).

1.3.3.2 Other Information

Appendix 1.2, *DOE-STD-3013 Storage Container Configurations*, is the other Appendix found in Chapter 1. A list of references is found in Section 1.3, References, of the SARP.

1.4 Evaluation Findings

1.4.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the design of the Model 9978 Package, as described in Revision 1 of S-SARP-G-00002, has been adequately described to meet the requirements of 10 CFR 71. By meeting the requirements of 10 CFR 71, the Model 9978-96 Package also meets the requirements of IAEA Safety 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 Model 9978 Package:

- The maximum allowable radioactive decay heat is 19 W.
- The maximum weight of the payload is not to exceed 50 lb.
- Contents are restricted to those described in Table 1.2 of the SARP, with additional restrictions provided by the applicable Table 1.2 Notes.
- Content Envelope loading configurations are further restricted to those described in Sections 1.2.2.1 and 1.2.2.2 of the SARP and in Table 1.3 of the SARP.
- The Model 9978 Package must be shipped in a closed conveyance.
- The Model 9978 Package was not evaluated to the requirements of §71.55(f); therefore, transport by air of fissile material is not authorized.
- Small concentrations (<1000 ppm) of other actinides, fission products, decay products, and neutron-activation products are permitted, except as noted in Table 1.2 of the SARP.
- 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 CSI for the Model 9978 Package is 1.0.

Drawings that define the packaging design for the Model 9978 Package SARP are delineated above, in Section 1.3.3.1.

1.5 References

[1-1] NRC, 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.

[1-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000) Vienna, Austria (2000).

[1-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[1-4] DOE, *Stabilization, Packaging, and Storage of Plutonium-Bearing Materials*, DOE Standard, DOE-STD-3013-2004, Washington, DC (April 2004).

[1-5] DOE, *Safeguards and Security Program*, DOE Order 470.4A, Washington, DC, May 25, 2007.

[1-6] NRC, *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, Washington, DC (June 1991).

[1-7] NRC, *Fabrication Criteria for Shipping Containers*, L.E. Fischer and W. Lai, NUREG/CR-3854 (UCRL-53544) (March 1985).

[1-8] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

[1-9] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, Division I, Subsection NF, Supports*, New York, New York (2004).

[1-10] U.S. Department of Transportation (DOT), 49 CFR Parts 171, 172, 173, 174, 175, 176, 177, and 178, *Hazardous Materials Regulations; Compatibility With the Regulations of the IAEA; Final Rule*, 69 F.R. 3632, pp. 3632–3896, January 26, 2004, as amended.

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2.0 Structural Evaluation

2.1 Areas of Review

This TRR documents the review of Chapter 2, Structural Evaluation, of the Safety Analysis Report for Packaging, Model 9978, 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 IAEA Safety Standards Series No TS-R-1.[2-3]

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

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- Lifting Devices.
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2.1.6 Structural Evaluation for Normal Conditions of Transport

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- Cold.
- Reduced External Pressure.
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- Vibration.
- Water Spray.
- Free Drop.

- Corner Drop.
- Compression.
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- Structural Requirements for Fissile-Material Packages.

2.1.7 Structural Evaluation for Hypothetical Accident Conditions

- Free Drop.
- Crush.
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- Thermal.
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- Immersion—All Packages.

2.1.8 Structural Evaluation for Special Conditions

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

2.1.9 Appendices

2.2 Regulatory Requirements

The regulatory requirements of 10 CFR 71 applicable to the Structural Evaluation review of the Model 9978 Package 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 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 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 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 chapter of the SARP for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 2.1 of this TRR.

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 Model 9978 Package is a single-containment, drum-type package. Its primary structural components are:

- A 35-gallon drum overpack (the overpack), which 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 protection for the containment vessel.
- A 5CV that contains the radioactive contents (i.e., a 3013 storage container, food-pack can, engineered container, or ICE container assembly. The allowable radioactive contents are described in the Content Envelopes section in the SARP.).
- Three aluminum honeycomb spacers that center the 5CV within the LDFs and form a protective volume around the 5CV while distributing impact loads to the 5CV and stiffening the liner to resist collapse.
- Two aluminum spacer assemblies, or LDFs, one at each end of the 5CV and honeycomb assembly, that hold the 5CV and honeycomb assembly in place within the drum cavity and spread the transmitted loads.

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 liner bottom, a circular drum 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. The drum shell, liner, and their bottoms are fabricated of 18-gauge (0.048-inch) Type 304L 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-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 (12 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 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-00044 and R-R2-G-00045 in the SARP. As applicable, the drum is designed, fabricated, analyzed, and accepted in accordance with Section III, Subsection NF of the ASME B&PVC.[2-5]

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, two 1-inch-diameter fill holes and three 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 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 (0.075-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 provide 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 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 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 TIDs. Details are shown on Drawing R-R1-G-00021 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 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 an 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 to 10 lb/ft^3 and 15 to 17 lb/ft^3, respectively. The thermal-physical properties of Fiberfrax® and of FR-3716 are listed in Tables 2.10, 2.11, 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-00021, R-R2-G-00044, and R-R2-G-00046 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-inches thick by 8 inches diameter. The TR-19 disk was installed over two ½-inch-thick blankets of Fiberfrax® insulation of the same diameter. The total thickness of the two blankets under compression is about ½ inch; therefore, 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-00045 in the SARP.

2.3.1.1.3 Containment Vessel (the 5CV)

The Model 9978 Package is designed with a single-containment vessel (the 5CV) with a nominal ID of five inches. As illustrated in Figure 1.4 of the SARP, the 5CV is an SS pressure vessel that was designed, fabricated, and analyzed in accordance with the requirements specified in Section III, Subsection NB of the ASME B&PVC.[2-6] The design condition for the 5CV is 900 psig at 300°F, as listed in Tables 2.21 and 3.1 of the SARP.

The 5CV is fabricated from 5-inch, Schedule 40, seamless, Type 304L SS pipe (0.258-inch nominal wall). A standard Schedule 40 Type 304L SS pipe cap (also 0.258-inch nominal wall) is welded to the pipe segment to form a blind end. A stayed head is machined from a 6-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 5½-12UN-2B internal threads and an internal cone-seal surface with a 32-micro-inch finish. Both vessel body joints are Category B, full-penetration, complete fusion, circumferential welds, in accordance with the ASME B&PVC, Section III, Article NB-3350. A support skirt to stand the 5CV vertically is formed from a short segment of 4-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 5CV 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 5CV 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 *leaktight* 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 5CV 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 Leak Test Port Gland Nut. The vessel containment boundary is formed by the 5CV body weldment, the Cone-Seal Plug, the Leak-Test-Port Plug, and the outer O-ring.

The internal volume of a closed 5CV is approximately 313 cubic inches. The nominal assembly weight is 32 lb, and the nominal overall length is 18.64 inches. The usable cavity of the 5CV is a minimum of 15 inches deep, with a minimum diameter of 5.02 inches. Details are shown in Drawing R-R2-G-00043.

2.3.1.1.4 Honeycomb Assembly and LDFs

Three aluminum honeycomb spacers center the 5CV between two LDFs. The Top and Bottom LDFs are made from 6061-T6 aluminum round bar, and fit within the Drum Liner cavity. The LDFs center the assembly in the liner, stiffen the package in the radial direction, and distribute the loads away from the 5CV and honeycomb spacer assembly (see Figures 1.1 through 1.3 of the SARP). Details are shown on Drawing R-R4-G-00033 and Drawing R-R4-G-00034.

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 5CV 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&PVC, 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 material specifications and fabrication requirements are listed in Tables 2.15 and 9.6 of the SARP and follow the requirements of ASME B&PVC, 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 that 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 8.3 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 2.16. 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 2.17 of the SARP identifies the dissimilar materials in contact in the Model 9978 Packaging, while Table 2.18 identifies the dissimilar materials in contact within the 5CV. Table 2.19 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:

- SS 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 SS 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.
- No significant galvanic cell interactions exist among any package materials and the SS interior of the 5CV.
- 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] The primary containment vessel of the Model 9975 Package, which is identical to the Model 9978 Package primary-containment vessel, has contents that bound the dose rate of the Model 9978 Package. Using the Model 9975 Package conservative dose of 1.8×10^4 rads over two years, the limit of one-year service life for the O-rings specified by the SARP is adequate for mitigating any radiation-related effects on the O-rings.

No degradation or activation of the SS 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 Model 9978 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, manufacturers 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 5CV and the overpack. The description of the overpack includes the drum body, the drum-closure lid, the Fiberfrax® insulation, and the polyurethane foam.

The 5CV body is constructed of Type 304L SS by butt welding a seamless pipe cap to one end of a five-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&PVC, Section III, Subsection NB-3350. A short segment of 4-inch pipe is skip-welded to the convex side of the cap, forming a skirt to support the 5CV vertically. The stayed head is rough-cut machined from a 6-inch-diameter by 2¼-inch-long bar, and is finish-machined to include 5½-12UN-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 material specifications and fabrication requirements for the overpack are listed in Tables 2.6 and 9.6, and follow the requirements specified in the ASME B&PVC, 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-G00046 in Appendix 1.1, the SARP specifies the procedure for the installation of Fiberfrax® insulation on the drum liner. Basically, two 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 one half of their original thicknesses when the overpack foam is installed. The overpack lid insulation also uses two ½-inch-thick Fiberfrax® blankets in the bottom chamber.

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 1-inch fill holes located in the bottom of the drum, and sets to a form rigid, closed-cell, intumescent polyurethane foam. Three ¼-inch holes are also present to vent the air being displaced. The twelve ¾-inch vent holes in the drum side-wall are closed with Caplugs® during foam emplacement. Installation is done by General Plastics personnel using a proprietary process and procedure. (See Appendix 8.3, *Acceptance Tests for Polyurethane Foam in the 9978 Packaging*.)

2.3.3.2 Examination

The surfaces and volume of the 5CV are examined in accordance with ASME B&PVC, Section III, Subsection NB, as described in Table 9.5 of the SARP. In addition, the closed 5CV assembly shall accept, without binding, a 5.02-inch-diameter by 15-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-00044 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 the liner's attachment to the drum shell are examined visually per ASME B&PVC, 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. Therefore, the SARP specifies that 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 that 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 intumescence, leachable chlorides, free-rise density, and compressive strength (parallel and perpendicular to rise). Specific foam installation and acceptance testing requirements are stipulated in Appendix 8.3 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 Model 9978 Package under NCT and HAC is predominately demonstrated by comparison to results of tests on the currently certified Model 9977 Package, which is of very similar design.[2-10]

2.3.4.2 Evaluation by Analysis

The non-linear plastic dynamic analyses presented in the SARP are used to augment the comparison to Model 9977 Package NCT and HAC test results. Also, the structural analyses results are used for demonstrating compliance with codes and regulations.

2.3.5 Lifting and Tie-Down Standards for All Packages

The Model 9978 Package 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 5CV during handling operations. Second, the overpack includes a series of eight 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 of the SARP and Section 2.5.1 of the SARP, it can be concluded that:

- The lifting devices for the 5CV and for the package meet 10 CFR 71 requirements for strength.
- 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 2.5.2 of the SARP concludes:

- Based on the stress analysis in Savannah River National Laboratory (SRNL) Document M-CLC-A-00317,[2-11] 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).
- 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 Model 9978 Package, 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 hour), no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging. The Model 9978 Package is also shown to meet the requirements in 10 CFR 71.55(d).

2.3.6.1 Heat

Section 2.6.1 of the SARP shows that the stresses generated by the Heat condition in the Model 9978 Package components are insignificant. For the 5CV, the SARP concludes that the stresses are acceptable, based on the following considerations:

- While the 5CV temperature caused by the Heat condition is similar to the 5CV design temperature (300°F), the corresponding 5CV pressure (56.3 psig) is significantly lower than the 5CV design pressure (900 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 5CV. (See Section 2.6.1.1 of the SARP.)
- Other than the pressure stresses in the 5CV, 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 5CV are within the corresponding allowable limits, as specified by ASME B&PVC, Section III, Subsection NB, for Level A service loads. (See Sections 2.6.1.3 and 2.6.1.4 of the SARP.)
- The 5CV and its closure design have been hydrostatically tested to greater than 7,000 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 Model 9978 Packaging to meet the regulatory requirements:

- The SS has adequate toughness to preclude brittle fracture at -40°F, based on Regulatory Guides (Reg. Guides) 7.11 and 7.12.
- The Viton® GLT elastomeric O-rings of the 5-inch-diameter Model 9965, which has an identical closure design as that used by the 9978 5CV's, have been leak tested at –40°F. The O-rings remained *leaktight* to sensitivity less than 10^{-7} ref·cm^3/s.
- The effect of *Cold* on FR-3716, the TR-19, Fiberfrax®, and Min-K® 2000 has been accounted for in the package analyses, and investigated through package tests. A prototypical Model 9977 Package was tested under exposure to an ambient temperature of –20°F (Appendix 2.10 of the Model 9977 Package SARP). 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 5CV pressure to 67.5 psig, including the Maximum Normal Operating Pressure, which is negligible in comparison to the 5CV design pressure of 900 psi. Therefore, the effects of this NCT condition are minimal.

2.3.6.4 Increased External Pressure

In Section 2.6.4, the SARP dismisses the significance of this NCT condition on the following basis:

- An ASME Code, Section III buckling analysis in Appendix 2.2 of the SARP has demonstrated that the 5CV 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; therefore, 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 SARP) of the random vibration and bolt-loosening effects, Section 2.6.5 of the Model 9977 Package SARP is referenced. It describes the details of a vibration test and the results for the Model 9977 Package. Because both the Model 9977 Package and the Model 9978 Package have the same overpack design, and because the 5CV is better secured against the effects of vibration with the surrounding honeycomb assembly, by comparison, it is concluded that the Model 9977 Package results demonstrate that the Model 9978 Package would be unaffected by vibration normally incident to transportation. The test specimen, identified as SN-2, is a prototypical Model 9977 Packaging with 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 any visible damage to the foam or any other packaging internals.
- The aluminum LDFs had noticeable scuffings and had produced some aluminum shavings. The result appeared to be due to 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.

Therefore, the Model 9978 Package 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 the Model 9977 Package, which has the identical overpack of the Model 9978 Package. 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.
  - No indication of absorbed water in the foam or 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 of the SARP and were found to be insignificant. The Model 9978 Packaging, with its honeycomb spacers, which are skinned, i.e., a fiberglass coating, has even less volume for water to accumulate.

Therefore, the SARP concludes that the Model 9978 Package meets the regulatory requirements for this NCT condition.

2.3.6.7 Free Drop

Neither testing nor analysis of the Model 9978 Package was performed for this requirement. However, because the Model 9978 Package is lighter and more robust, the Model 9977 Package bounds the free-drop results. Compliance is demonstrated through comparison to prototype testing and analysis of the Model 9977 Packaging. The NCT 4-ft drop, 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 Model 9977 Package. The SARP compares this unyielding target to the IAEA example unyielding target, which is described in IAEA Advisory Material paragraph 717.2,[2-12] 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.

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-ft 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 Model 9977 Packaging prototype, was also dropped 4 ft top-down. As with SN-2, no significant damage was observed to the lid or the drum rim.[2-13]

Finite-element analysis of the 4-ft 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, center-of-gravity-over-corner, 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 with a Model 9978 Packaging.)

Based on the previously 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 Model 9978 Package because its nominal weight of 267.3 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 by comparison to a Model 9977 Packaging test, which has the same overpack as the Model 9978 Packaging. Section 2.6.9 of the SARP describes the test, in which a compressive load of 1,750 lb was applied to the Model 9977 Packaging prototype specimen, SN-2, for 24 hours. There was no observable deformation to the Model 9977 Packaging at the end of the 24-hour test. Because the Model 9977 and Model 9978 Packaging overpacks are identical, and the compressive load is less for the Model 9978 Packaging, it was concluded that the Model 9978 Package satisfies the regulatory requirements.

2.3.6.10 Penetration

This regulatory requirement is satisfied by testing a Model 9977 Packaging overpack, which has the same design as the Model 9978 Packaging overpack. As described in Section 2.6.10 of the SARP, the hemispherical end of a vertical steel cylinder of 1.25-inch 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 9978 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 Model 9978 Package is shown to meet the performance requirements of 10 CFR 71.73 by a comparison to the physical testing of the Model 9977 Package and other similar packages. Analysis results on the Model 9977 Packaging were compared against test results, and these were, in turn, compared with analytical results on the Model 9978 Packaging. Analysis of the Model 9978 Packaging in conjunction with the Model 9977 Packaging was used to augment comparison to testing. Appendix 2.8 presents a detailed comparison of the Model 9978 and Model 9977 Packagings to earlier development designs. Physical testing has been performed on these 9977 development designs. Design changes made to the packages through the course of development testing are included in the final analysis models for the Model 9977 Package.

Nine prototypical Model 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 Model 9977/9978 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 Model 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-inch or 18-inch diameter and filled with 16-, 20-, or 24-lb/ft^3 polyurethane foam).

- Four Prototype Series 2 test packages (SN-2, -3, -4, and -5) that differ from the (final) Model 9977 Package, mainly in the drum bottom design. (The SN-series package design had a standard crimped bottom chime, while the final Model 9977 Package 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 *leaktight* 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 Model 9977 Package. The 6CV remained *leaktight* after the series of HAC tests.

Nonlinear plastic dynamic analyses with a calibrated finite-element model of the Model 9977 Package 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 requirement is satisfied by referencing immersion testing on the Model 9966 Packaging containment vessel, which is identical to the 5CV.

The 50-ft immersion-test requirement was also satisfied by referencing the immersion testing performed on the containment vessels of the Model 9966 Packaging. Test results for the Model 9966 Package showed no inleakage of water or structural degradation to its containment vessels following either the 3-ft or the 50-ft immersion test.

Section 5.1 of the SARP shows that calculated dose for a damaged Model 9978 Package is less than 2.56 mrem/h at 1 m from the package surface, meeting the regulatory requirement of less than 1 rem/h at 1 m (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 in Section 2.3.6.7 above for the testing of the Model 9977 Packaging.

As described in the introduction to Section 2.3.7, nine test packages (i.e., DP-1, -2, -3, -5, and –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 (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 atm. 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 the most challenging for both the Model 9977 and 9978 Packages. None of the drops was a "shallow-angle" drop, i.e., a drop with the package axis oriented nearly horizontally or nearly vertically.

Sections 2.7.1.1 through 2.7.1.5 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 expanded 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 ft.
- 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.
  - DR 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 in the foam.

— Package rebound height about 3.5 ft.

- 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 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 Model 9977 Package 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, analyses were also used to justify the omission of oblique (slap-down and shallow-angle) drops. Based on the results and the high similarity between the Model 9978 and Model 9977 Packaging deformation plots, it is concluded that the Model 9978 Packaging performs adequately for this HAC event.

2.3.7.2 Crush

Section 2.7.2 of the SARP points out that the HAC crush test needs to be performed on the Model 9978 Package in accordance with 10 CFR 71.73(c)(2). However, the Model 9978 Package design was not tested—the crush test was performed on the Model 9977 Package and supplemented with Model 9978 Package analyses. Appendix 2.6 provides the nonlinear dynamic analyses of the Model 9978 Package drop and crush conditions, which are also referenced in Section 2.7.2. Analysis of the Model 9977 Package is also referenced in Section 2.7.2 of the SARP.

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 inches thick, is floated on approximately ¾-inch of grout, and is anchored to the building footer by five 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 lifting eyes, located at the four plate corners. During testing, the plate is magnetically released.

Four Model 9977 Packaging prototypes were crush tested after the 30-drop test. 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° 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 BD from 30 ft and then crushed CGOT, as is shown in Figure 2.40 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 BD 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, there was no other apparent damage inside the overpack.

Nonlinear dynamic analyses of the Model 9978 Packaging drop-and-crush conditions are provided in Appendix 2.6. Analysis of both the Model 9977 and the Model 9978 Packaging is referenced in Section 2.7.2. Both the Model 9977 and the Model 9978 Packaging simulations resulted in stresses below the allowable ASME, Section III, Appendix F, Article F-1341.2 stress limit. For each side-by-side comparison, the two models showed little or no appreciable difference in package crush results. Given the crush tests and comparisons with analytical results, it is concluded that the Model 9978 Packaging demonstrates structural adequacy in this HAC event.

2.3.7.3 Puncture

Section 2.7.3.1 of the SARP describes the puncture testing of the Model 9977 Packaging. Because the Model 9978 Packaging has the identical overpack, the Model 9977 Packaging tests and analyses bound the Model 9978 Packaging. The puncture bar used for the test was 6 inches in diameter by 40 inches 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-ft drop and the HAC 30-ft drop.

A horizontal (side) drop over the package center of gravity (CG) 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 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 BD puncture test in an attempt to exploit a rip in the drum chime that occurred during the crush test, described in Section 2.3.7.2 above. 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 Model 9977 Package SARP presents the results of a set of nonlinear plastic dynamic analyses of the puncture tests over the CG at two orientations, i.e., at 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 SS. The analyses also indicated that the puncture test can produce plastic strain of more than 2% in the 6CV. Section 2.7.3.2 of the Model 9978 Package SARP also references the puncture analysis. Given the analysis and testing of the Model 9977 Package, and the fact that the mass of the Model 9978 Package is 65 lbs less, it is concluded that the testing and analyses are bounding to the Model 9978 Package.

2.3.7.4 Thermal

Section 2.7.4 of the SARP addresses HAC thermal-test conditions. Compliance with the thermal requirements of HAC was shown by comparison to Model 9977 Packaging testing and Model 9978/9977 Packaging thermal numerical 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 is 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 undegraded 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. Therefore, the thermal protection of the drum liner and 5CV 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 this requirement was met by assuming water inleakage. Analysis of a single 5CV per 10 CFR Part 71.55(e), which demonstrated a k_{eff} below the k_{safe} value of 0.931 for both plutonium and uranium, thus showed that the 10 CFR 71.55(b) requirements related to a flooded single package were satisfied. In addition, it should be noted that other similar packages have passed the immersion test under a three-foot head of water.

2.3.7.6 Immersion—All Packages

The Model 9978 Packaging did not undergo immersion testing. Section 2.7.6 of the SARP states that this is shown by comparison to be compliant with 10 CFR 71.73(c)(6) on the basis of identical design of the CV closure designs for the Model 9978 Package and the Model 9966 Package. Inspection of the tested Model 9966 Package 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.[2-14]

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 Model 9978 Package as being 130,000 A_2, which exceeds the 10^5 A_2 limit. Therefore, the package must be designed to withstand an external water pressure of 290 psi for a period of not less than 1 hour without collapse, buckling, or inleakage of water. Testing was not performed on the Model 9978 Packaging, but analysis was performed. Buckling analysis in Appendix 2.2 of the SARP shows stresses from the external pressure on the 5CV will not cause buckling. Also, criticality analysis shows that even in the worst case of water inleakage, the package is critically safe.

2.3.8.2 Analysis of Pressure Test

Appendix 2.2 of the SARP provides a linear elastic finite-element stress analysis of the 5CV hydrostatic pressure test. The stress results are within the allowable limits specified in ASME B&PVC, Section III, Subsection NB, for hydrostatic test conditions. Therefore, the hydrostatic test of the 5CV is safe to perform.

The test pressure used for the 5CV is 1.5 times the design pressure of 900 psig, which is greater than the regulatory requirement of 1.5 times the MNOP. The MNOP, which is listed in Table 2.21 of the SARP, is 56.3 psig.

2.3.9 Appendices

There are eight appendices associated with Chapter 2 of the SARP:

- Appendix 2.1, *General Normal Condition Design Calculations for 9978 Packaging*.
- Appendix 2.2, *General Design and ASME Calculations for the 9978 Containment Vessels*.
- Appendix 2.3, *9978 General Purpose Fissile Packaging Weights*.
- Appendix 2.4, *9965 Cone-Seal Closure Performance at –40°F*.
- Appendix 2.5, *9978 Packaging Materials and Components of Fabrication*.
- Appendix 2.6, *Hypothetical Accident Condition Analysis Drop and Crush for the Model 9978*.
- Appendix 2.7, *SRS Drop Test Pad Infrastructure for Drums*.
- Appendix 2.8, *9978 Packaging Comparisons with the GPFP Development Prototype.*

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 2.1.4, 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&PVC does not include *Fabrication* requirements. Fabrication requirements can be found in Article NB-4000 of the B&PVC. The Staff recommends that the appropriate changes be made as part of the next revision to the SARP.
- 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 (SS, 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 SS walls of the 5CV. The Staff recommends that additional emphasis should be placed on the visual inspections for the LDFs in Chapter 7 of the SARP, and that this be part of the next revision to 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 Model 9977 Package 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. This

same design enhancement was incorporated in the Model 9978 Packaging. The Staff agrees that the design changes were necessary. However, the Staff also notes that a Model 9977 or Model 9978 Packaging incorporating this design change has not been tested. However, a finite-element analysis of the Model 9977 Packaging with a simplified drum model incorporating a welded closure indicates that the new design is adequate. The Staff does recommend that any published results of drop/crush testing on drum packagings with a welded sanitary closure of similar size/weight to the Model 9978 be referenced in the next revision to the SARP to complement the analytical results.

- For the HAC impact tests described above in Section 2.3.7 of this TRR, 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-ft drop case. This does not seem to be a reasonable result because there is much less available kinetic energy to dissipate through plastic deformation than in the 30-ft drop cases.

  The Staff still considers the nonlinear analyses useful in providing a qualitative understanding of the structural behavior of the Model 9977 and 9978 Packages. The Staff has made its affirmative conclusion on the compliance of the Model 9978 based on the testing of the Model 9977. However, the Staff still considers the nonlinear analyses useful in providing qualitative understanding of the structural behavior of the Model 9977 and Model 9978 Packages.

- The Staff reached the conclusion of compliance of the Model 9978 (5CV) Package based on the test results of the Model 9977 Package and other similar packages, and not on the supplemental Model 9978 Package analysis that accompanies the SARP. Supporting this conclusion is the lower weight of the 5CV, which will reduce loading to the packaging during NCT and HAC events. Additionally, the 5CV is more robust than the 6CV and has a protective honeycomb layer to help mitigate loadings to the CV itself.

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

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

2.5 References

[2-1] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[2-2] NRC, 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] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000), Vienna, Austria (2000).

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

[2-5] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NF, Supports*, New York, New York (2004).

[2-6] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

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

[2-8] ASME, *ASME Boiler and Pressure Vessel Code, Section VIII, Rules for Construction of Pressure Vessels,* New York, NY (2004).

[2-9] Skidmore, T. E., *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] 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, SRNL (August 2007).

[2-11] Wu, T. T., *Tie-Down Load Analysis of 9977 Shipping Package*, M-CLC-A-00317, Rev. 1, Savannah River Site, Aiken, SC (August 2007).

[2-12] IAEA, *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/, Vienna, Austria (July 2002).

[2-13] Gelder, L. G., *General Purpose Fissile Material Package Prototype Testing*, M-TRT-A-00006, Revision 0, Westinghouse Savannah River Company (May 2005).

[2-14] Westinghouse Savannah River Company, *Immersion of the 9966 Package*, M-TSM-A-00005, Rev. 0, Aiken, SC (October 2003).

3.0 Thermal Evaluation

3.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[3-1] and in IAEA Safety Standards Series No. TS-R-1.[3-2] The description and engineering drawings in Chapter 3, Thermal Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[3-3] were reviewed.

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 Table of Temperatures.
- Summary Table of Maximum Pressures.

3.1.2 Material Properties, Thermal Limits, and Component Specifications

- Material Properties.
- Temperature Limits.
- Component Specifications.

3.1.3 General Considerations for Thermal Evaluations

- Evaluation by Analysis.
- Evaluation by Test.

3.1.4 Thermal Evaluation under Normal Conditions of Transport

- Initial Conditions.
- Effects of Tests.
- Maximum and Minimum Temperatures.
- Maximum Normal Operating Pressures.
- 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 requirements of 10 CFR 71 applicable to the Thermal review of the Model 9978 Package include:

- 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 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 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).
- 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 performance of the package must be evaluated under the tests specified in §71.73 for hypothetical accident conditions. [§71.41(a)]

3.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Thermal chapter of the SARP for the Model 9978 Package. 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 Model 9978 Package to the thermal environment. These components, which primarily consist of the 5CV 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 Model 9978 Package from structural damage and overheating are:

- Last-A-Foam® (or polyurethane foam) overpack, confined in the SS drum, which acts as an impact limiter and provides insulation under hypothetical accident conditions.
- The SS pressure vessel with Cone Seal Plug and Cone Seal Nut, which provides the containment of the package contents during NCT- and 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 outer O-ring serves as part of the containment boundary.

The Model 9978 Package accommodates five types of payloads listed in Section 1.2.2. Allowable payloads are specified to be shipped in one of the following containers: 3013 storage container, nested food-pack cans, engineered container, or ICE container. The specification of payload to the corresponding container is described in Table 1.3 of the SARP.

3.3.1.2 Decay Heat of Contents

The maximum decay heat rate of payloads for the Model 9978 Package 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 Model 9978 Package. Provisions will be made to limit the maximum decay heat rate to 19 W per package.

3.3.1.3 Codes and Standards

The structural materials used in the packaging conform to Section III of the ASME B&PVC.[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 Model 9978 Package components in the events of NCT in shade and in sunlight are given in Tables 3.2 and 3.5 of the SARP. The bounding content configuration is identified as a 19-W heat source (Content Envelopes C.1 or C.2) in a food-pack can configuration. The temperatures for food-pack cans, evaluated at two elevation levels inside the 5CV, are listed in Table 3.5.

In the case of NCT without insolation, the temperature distribution of the package with 19-W decay heat rate bounding-content configuration is shown in Figure 3.13. The minimum package 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 Model 9978 Package, including a 19-W heat source, has a maximum accessible surface temperature of 114°F, per Table 3.5 of the SARP, which is below the limit of 122°F, allowed for nonexclusive-use shipments.

The applicant presents the maximum temperature in the Model 9978 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 Model 9977 Packagings, packagings that are very similar to the Model 9978 Packaging design. Table A-1 in Appendix 3.4 (M-CLC-A-00325, Rev. 0) presents the temperatures of the components of Model 9977 Packagings in the fire tests.

The post-fire analysis of undamaged Model 9978 Packagings with a simulated 19-W content decay heat rate is conducted with an initial condition of combining NCT simulation results and HAC test measurements. Table 8 in Appendix 3.4 lists the transient post-fire HAC temperatures of undamaged Model 9978 Packagings, determined by finite-element analysis (FEA) simulations for both the 5CV and the O-rings. The maximum O-ring and 5CV temperatures are 378ºF, after three hours, and 392ºF, after 30 minutes, respectively, following cessation of the fire.

3.3.1.5 Summary Tables of Pressures

The Maximum Normal Operating Pressure (MNOP) values inside the 5CV cavity of the Model 9978 Package are given in Table 3.18 of the SARP. The maximum pressure in the 5CV during HAC is 48.5 psig. The maximum pressure inside the 5CV is lower for the HAC fire event in comparison to MNOP.

3.3.2 Material Properties, Temperature Limits, and Component Specifications

3.3.2.1 Material Properties

The thermal properties for the materials used in the Model 9978 Packaging are 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 5CV; the remainder was in the form of char.
- A 2.5-inch-thick layer of foam left surrounding 5CV; the remainder was in the form of char.
- All the foam burned into char.

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

3.3.2.2 Temperature Limits

The temperature limits of the 5CV, the Viton® GLT/Viton® GLT-S O-rings, and the Last-A-Foam® are given in Table 3.1 of the SARP, and the thermal decomposition temperature of the Last-A-Foam® is described in Section 3.2.1 of the SARP. The pressure limit of the 5CV 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 5CV 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

The NCT thermal tests, described in Appendix 3.1 of the SARP, were performed on a prototype of the 6CV 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 packaging was tested for about 143 hours within a thermal chamber with the internal environmental temperature controlled to 100°F. These 6CV packaging testing data were used for the 5CV packaging because of the similarity between those packages. The thermal model of the Model 9978 Packaging with benchmarked material properties is employed for the calculations of NCT event with insolation is described in Appendix 3.3 (M-CLC-A-00255, Rev. 2), and NCT event without insolation is described in Appendix 3.4 (M-CLC-A-00325, Rev. 0) of the SARP.

For HAC tests, following preheating in the thermal environmental chamber to the prefire condition, four damaged 6CV Packagings 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 Model 9977 6CV Packagings had been subjected to the drop, crush, and puncture, 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 of the fire tests, the flame extended horizontally more than 1 m (40 in.) beyond the tested packages. TILs were utilized to track the increase in temperature of the components during the fire. The maximum, measured temperature difference in the 6CV packagings, caused by the fire, was superimposed on the analytical 5CV model for the post-fire analyses, as described in Appendix 3.4 (M-CLC-A-00325, Rev. 0) of the SARP, due to the similarity between the 6CV and 5CV packagings.

The temperatures and pressures for the 5CV 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, as given in Table 3.5. 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 Model 9978 Package simulations. The thermal properties of the packaging materials, including insulation and air, are listed in Section 3.2 of the SARP. The Staff has deemed that 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 3.6 of the SARP.

Material properties, convection coefficients, radiation-surface properties, and heat generation and insolation data used in the analyses are listed in Section 3.2 of the SARP. Analyses performed on the undamaged Model 9978 Packaging for NCT were benchmarked against the 6CV packaging experiments, as discussed in Appendices 3.3 and 3.4 of the SARP. Thermal properties, verified in NCT, were used for calculating the temperature distributions corresponding to the initial conditions before the HAC fire. The postfire analyses of the Model 9978 Packagings 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 Model 9978 Packaging, under NCT conditions, with insolation applied to the surfaces of the packaging in 100°F still air. The insolation is based on the values given in 10 CFR 71.71(c) for a 12-hour period. The absorptivity of the SS drum surface was assumed to be 1.0 for solar radiation, 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 6CV Packaging, which used a 19-W heater to simulate the content decay heat rate. 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 range of 15 to 17 lb/ft^3 by the procedure specified in Appendix 8.3 of the SARP.

3.3.4.3 Maximum and Minimum Temperatures

The minimum temperature of –40ºC in the package is assumed to occur when the decay-heat-rate load is zero watts in an environment of –40ºC. As noted in Section 3.3.1.4 above, the *Cold* condition of -40ºC ambient temperature will not result in degradation of the Model 9978 Packaging or any of its components. The 304 L austenitic SS, used for the 5CV 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 two axial positions in the 5CV of the food-pack cans, as described in Appendix 3.3 of the SARP. 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 these 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 Pressures

The MNOP in the 5CV is the result of (1) the increase in temperature of the enclosed cavity air with an initial pressure of 14.7 psia and an initial temperature of 70°F and (2) off-gassing of 100 g of plastic, e.g., polyurethane, low-density polyethylene, nylon, or polyvinyl chloride tape. The MNOP, calculated by the applicant, is given for the 5CV in Summary Table 3.18 of the SARP. This pressure, 56.3 psig, obtained for the case of food-pack can content, is an upper bound for the 5CV with other content configurations. As shown in Chapter 3 of the SARP, this pressure does not produce stresses in the 5CV 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 Model 9978 Package 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 5CV of the Model 9978 Package 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 Model 9978 Package 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 Model 9978 Packaging (5CV) for HAC were performed by fire testing on a similar 6CV packaging and by FEA simulations on a 5CV packaging. The FEA simulations of the Model 9978 packaging were, after benchmarking its material model, performed to calculate temperature distributions in the 5CV Packaging in pre-fire and post-fire events with a maximum internal heat generation rate of 19 W. The pool-fire tests were carried out to determine the temperature response of the 6CV Packagings under the conditions specified in §71.73(c)(4). The initial temperatures for Model 9978 Packaging for HAC post-fire were obtained by superimposing the temperature increases on the components during the 30-minute fire to the temperature profile obtained under NCT conditions.

3.3.5.2 Effects of Thermal Test

Both NCT and HAC thermal tests were performed on the 6CV Packagings with a mock content of a 19-W heater. It is appropriate that those test results were applied to the Model 9978 Packaging for benchmarking the numerical model and including the fire affect because of the similarity of 6CV and 5CV packagings.

The post-fire transient analyses were modeled with an undamaged Model 9978 Packaging, and used the thermal properties of the charred and uncharred polyurethane foam with the three thickness conditions (noted in Section 3.3.2.1 above). The convection coefficient correlations in Table 3.12 of the SARP demonstrate that the formulations of external boundary conditions, imposed on the analytical model, were appropriate. The analytical models with the content decay heat at different elevations inside 5CV were used to calculate the highest temperature in each key component of the package. The results are combined with the most unfavorable temperature in each case.

3.3.5.3 Maximum Temperatures and Pressures

The maximum temperatures experienced by the Model 9978 Packaging 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 5CV is due to the increase in temperature of the cavity air and off-gassing of 100 g of plastic. The maximum bounding pressure of 48.5 psig, calculated by the applicant for the food-pack can content, is given for the 5CV in Appendix of 3.8 of the SARP, as described in Section 3.3.1.5 above.

3.3.5.4 Maximum Thermal Stresses

As described in Section 3.4.3.2 of the SARP, the pressure does not produce significant stresses in the 5CV that could exceed the allowable stress limits under HAC. A review of the calculations of the pressures, produced under HAC, confirmed that the results were reasonable and conservative. Also, as described in Section 3.4.4 and in Section 2.6.1.2 of the SARP, the thermal stresses in the Model 9978 Package, due to the differential thermal expansions between the packaging components, are small under HAC.

The Staff finds that the 5CV of the Model 9978 Package remains fully effective as a containment boundary for Content Envelopes C.1 through C.6 during HAC. The resultant deformation of the vessel will not impair the containment or criticality safety functions of the 5CV. Therefore, the functions of the Model 9978 Package with Content Envelopes C.1 through C.6 payloads are not affected by HAC.

3.3.6 Thermal Evaluation of Maximum Accessible Surface Temperature

The maximum accessible surface temperatures of the Model 9978 Package with the maximum decay heat rate were determined without insolation, based on the surface-heat flow by natural convection and thermal radiation to the environment at an ambient temperature of 100°F. This 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 10 Appendices associated with Chapter 3 of the SARP:

- Appendix 3.1, *NCT Thermal Analysis Benchmarking*, S-TSM-A-00001.
- Appendix 3.2, *SRNL Package Burn Test Report*, NT-TDR-06-101.

- Appendix 3.3, *NCT Thermal Model for the 9978 Package*, M-CLC-A-00255.
- Appendix 3.4, *HAC Thermal Model for the 9978 Package*, M-CLC-A-00325.
- Appendix 3.5, *MSC/PATRAN/THERMAL Version 2003 Software Test Documentation*.
- Appendix 3.6, *PATRAN-PLUS and P/Thermal Code Descriptions*.
- Appendix 3.7, *Pressure Contribution Due to Plastic Bags*.
- Appendix 3.8, *MNOP and Maximum Operating Pressure for the 9978 Package*, M-CLC-A-00256.
- Appendix 3.9, *Determination of the Volume of LAST-A-FOAM® Remaining in the 9977 Packaging after the Hypothetical Accident Conditions (HAC) Thermal Test*, M-CLC-A-00254.
- Appendix 3.10, *Thermal Properties of Aluminum Honeycomb.*

3.3.7.1 Description of Test Facilities and Equipment

The thermal environmental chamber tests of 6CV Packagings were performed at the 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 Model 6CV Packagings, is presented in Appendix 3.2 of the SARP. During the pool-fire tests, the 6CV Packagings were tested without the 19-W heater present to simulate the content decay-heat source. Adjustments were made to the measured temperatures by adding conservatively-estimated effects of the decay heat rate. The test report includes the measured temperature histories of various components.

3.3.7.2 Test Reports

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 6CV 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 6CV Packagings after the pool-fire tests were performed is described in Appendix 3.9. By the measurement of the tested packagings 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 packaging.

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,[3-7] 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 Model 9978 Packagings 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 9978 Packagings (5CV) using the P/Thermal module were performed to simulate the packagings under post-fire conditions. The analytical model was adjusted to bring the calculated temperatures of the Model 9978 Packaging under NCT into a sufficiently accurate correspondence with the measured 6CV packaging results (Appendices 3.1, 3.3, and 3.4) and into correspondence with the calculated initial HAC post-fire temperatures after superimposing the temperature changes in the components obtained from the fire 6CV tests (Appendices 3.2 and 3.4).

The pressure change in the 5CV 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 5CV 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 Model 9975 Package, a certified package of similar design.
- 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 given 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 actual revision number for each Appendix should be documented in Section 3.6 of the SARP.
- The general formulae of natural convection heat-transfer coefficients are provided in Table 3.12 of the SARP. The actual values of the convection coefficients applied to the FEA model for NCT and HAC should be included in the SARP or a section of the FEA input file attached for verification of the proper use of the formulae.

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 TRR for the Model 9978 Package, the shipment of Content Envelopes C.1 to C.6 must conform to Table 1.2 of the SARP, and any content configurations must include a maximum decay heat rate limit of 19 W.

3.5 References

[3-1] NRC, 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.

[3-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000) Vienna, Austria (2000).

[3-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[3-4] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, New York, New York (2004).

[3-5] General Plastics Manufacturing Company, Product Manual, *General Plastics Last-A-Foam® FR-3700 for Crash & Fire Protection of Nuclear Material Shipping Containers*, Tacoma, Washington (March 2001).

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

[3-7] American Society for Testing and Materials, ASTM D 4635-08a, *Standard Specification for Polyethylene Films Made from Low-Density Polyethylene for General Use and Packaging Applications*, West Conshohocken, Pennsylvania, 19428-2959, November 1, 2008.

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4.0 Containment Evaluation

4.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[4-1] and in IAEA Safety Standards Series No. TS-R-1.[4-2] The description and engineering drawings in Chapter 4, Containment Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[4-3] were reviewed.

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

4.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Thermal review of the Model 9978 Package include:

- 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 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 chapter of the SARP for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 4.1 of this TRR.

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 Model 9978 is a Type B Package and must satisfy the quantitative release rates specified in §71.51(a)(1) for NCT and §71.51(a)(2) for 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 8.1.4, *Leakage Tests*, of the SARP). (*Leaktight* is defined as being a leakage rate of air that is less than or equal to 1×10^{-7} ref·cm^3/s, i.e., ref·cm^3/s, at an upstream pressure of 1 atm and a downstream pressure of 0.01 atm or less, at 25ºC.) By adopting the ANSI N14.5 definition of *leaktight*, the applicant is not required to show any calculations to justify its position. 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} ref·cm^3/s 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, *Maintenance Leak-Rate Test*, of the SARP. Replacement of the Leak-Test Port Gland Nut, the Leak-Test Port Plug, or the O-rings with equivalent items does not constitute a structural modification, hence does not require pressure testing of the 5CV.

The review also verified that the Model 9978 Package 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 except for a brief mention in Section 4.2.2, *Pressurization of the Containment Vessel*. In this case, the applicant has followed the format specified in the U. S. Nuclear Regulatory Commission's (NRC's) Regulatory Guide (Reg. Guide) 7.9.[4-5] The applicant has noted that the amount of plastic (as low-density polyethylene, nylon, or polyvinyl chloride tape) is limited to 100 g for food-pack can configurations for any of the five Content Envelopes currently allowed in the Model 9978 Package. For the 3013 storage container, no organic liners are allowed in any of the containers, from the product can(s), to the inner container, to the outer container. For the Model 9978 Package, only Content Envelope C.2, i.e., plutonium/uranium metal, is shipped in the 3013 storage container. For engineered containers, only 100 g of plastic, as described above, is allowed for Content Envelopes C.1 and C.6 (no other Content Envelopes are permitted in the engineered containers). The ICE Special Nuclear Material Target Container Assembly applies only to Content Envelope C.6 and can have up to 100 grams of nitryl-PCV foam (Armaflex SA foam). 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*; in 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 9978 Package GPFP,* M-CLC-A-00257. This is in accordance with Reg.

Guide 7.9, where these topics are covered in Section 3.3.2, *Maximum Normal Operating Pressures*, and in Section 3.4.3, *Maximum Temperatures and Pressures*, respectively, and in their associated appendices.

4.3.1.2 Design Features

The 5CV Closure Assembly consists of a male–female cone joint with surfaces that have been machined to matching 10° angles so that the Cone-Seal Plug mates with the Containment Vessel body with a maximum radial clearance of 0.0007 inches, as described in Drawing R-R2-G-00043 of Appendix 1.1 of the SARP. The sealing surfaces are machined to a 32-micro-inch finish. Two grooves are machined into 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, threaded 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. KRYTOX® is also applied to the surface between the Cone-Seal Nut and the Cone-Seal Plug.

The Cone-Seal Nut is fabricated from Nitronic® 60 SS alloy, and is cut with 5½-inch 12UN-2A external threads. The CV weldment is made from 304L SS and is cut with 5½-inch 12UN-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 50 (+10/–0) ft-lb. A maximum radial clearance of 0.0007 inches, as described in Section 4.1.4, *Closure*, and in Appendix 2.2, *General Design and ASME Calculations for the 9978 Containment Vessels*, 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. As described in Section 8.1.3, *Structure and Pressure Tests*, of the SARP, the 5CV assembly is pressure tested to 1,365 ± 10 psig for acceptance, 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, *5-inch Diameter Containment Vessel*, 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-00043 of Appendix 1.1 in the SARP. 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. Because the package can contain up to 130,000 A_2, as documented in Appendix 4.1, *Determination of A_2 for the 9978 Fissile Package with Contents*, the radioactive-material package 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 ASME B&PVC.[4-7] The welds are examined with liquid penetrant and are fully radiographed.

The 5CV incorporates a static seal design using concentric, elastomeric O-rings fit within circumferential grooves machined into the Cone-Seal Plug, as illustrated in Figure 4.2, and by Drawing R-R2-G-00043 in Appendix 1.1 of the SARP. The O-rings are made of Viton® GLT or Viton® 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 at pressures up to 1,000 psig. Also, the O-rings and the CV 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 MNOP is 56.3 psig. The 5CV design pressure is 900 psig. The corresponding content gas temperature for MNOP is 535°F, as listed in Table 2.21 of the SARP. The maximum pressure developed under HAC is 48.5 psig for the 5CV. 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, *Containment under Normal Conditions of Transport*, 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 SARP Section 2.6, *Normal Conditions of Transport*, and the analysis in SARP Section 3.3, *Thermal Evaluation under Normal Conditions of Transport*, 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 5CV include actinide ^{238}Pu heat sources, oxides, metals, alloys, samples, and sources 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 5CV.

The regulatory limit for the release of radioactive material during NCT (10^{-6} A_2/h) is met by measuring the 5CV 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 atm absolute pressure before 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 Closure Assembly to allow for the detection of helium leakage across the Inner O-ring seal. Section 2.7, *Hypothetical Accident Conditions*, Table 2.25 and Table 2.26 of the SARP list the leak rates that were measured before the NCT testing 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 Model 9965, 9966, 9967, 9968,[4-11] 9975,[4-12] and 9977[4-13] 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 HAC are similar to those under NCT. Differences relevant to HAC are noted below.

4.3.3.1 Containment Design Criteria

Containment under HAC is addressed under Section 4.3, *Containment under Hypothetical Accident Conditions*, 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 5CV is tested, as described in Sections 8.1, *Acceptance Tests*, and 8.2, *Maintenance Program*, of the SARP, to be *leaktight* per the ANSI N14.5 definition, thus meets the requirement of 10 CFR 71.51(a)(2) 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, *Hypothetical Accident Conditions*, and 3.4, *Thermal Evaluation under Hypothetical Accident Conditions*, 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/s, under reference air-leakage test conditions. This was also verified in the Acceptance Test and Maintenance Program of the SARP in Chapter 8.

The preshipment leakage-rate test criterion is 10^{-3} ref·cm^3/s, 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

Chapter 4 has one Appendix, Appendix 4.1, *Determination of A_2 for the 9978 Fissile Package with 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:

- There is no specific subsection for *Combustible-Gas Generation*, Chapter 4 of the SARP. It is recommended, therefore, that, in future submissions, the applicant provide a more complete description of Combustible-Gas Generation issues, in accordance with the requirements specified in Section 4.3.1.1.3 of DOE's *Packaging Review Guide*.[4-14]
- The applicant in Section 4.2.3 of the SARP has noted that, for the pre- and post-NCT/HAC leak-rate tests, each 6CV was evacuated and backfilled with helium to one atm absolute pressure before 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 and Tables 2.25 and 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 Model 9977 Packaging, where the *outer* O-ring is now defined as being part of the containment boundary. It is recommended, therefore, that the applicant provide a more appropriate description of the leakage testing performed for the pre- and post-NCT/HAC leak-rate tests for the Model 9977 Packaging.

The Staff recommends that, in future submissions, 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] NRC, 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.

[4-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000) Vienna, Austria (2000).

[4-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[4-4] ANSI, *American National Standard for Radioactive Materials-Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, New York, New York, 10036 (1977).

[4-5] NRC, 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] NRC, *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, Washington, DC (June 1991).

[4-7] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

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

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

[4-10] Chalfant, G., *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, SRNL (December 2003).

[4-13] SRNL, *Safety Analysis Report for Packaging, Model 9977 B(M)F-96*, S-SARP-G-00001, Revision 2 (August 2007).

[4-14] DiSabatino, A. A., et al., DOE Packaging Certification Program, *Packaging Review Guide for Reviewing Safety Analysis Reports for Packaging*, UCID-212218, Revision 3 (February 2008).

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5.0 Shielding Evaluation

5.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71,[5-1] and in IAEA Safety Standards Series No. TS-R-1.[5-2] The description and engineering drawings in Chapter 5, Shielding Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[5-3] were reviewed.

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

5.1.1 Description of Shielding Design

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

5.1.2 Radiation Sources

- 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

5.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Shielding review of the Model 9978 Package include:

- 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, the external radiation levels must meet the requirements of §71.47(a) for nonexclusive-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 normal conditions of transport. [§71.43(f), §71.51(a)(1)]
- Under the tests specified in §71.73 for hypothetical accident conditions, the external radiation level must not exceed 10 mSv/h (1 rem/h) at one meter from the surface of a Type B package. [§71.51(a)(2)]

5.3 Review Procedures

The following subsections describe the review methods for the Areas of Review applicable to the Shielding chapter of the SARP for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 5.1 of this TRR.

5.3.1 Description of Shielding Design

5.3.1.1 Design Features

The Model 9978 Shipping Package is a single-containment drum-type package with a bolted flange closure and a right circular cylinder CV enclosed by insulation. Major materials of construction include an SS overpack, drum, Fiberfrax®, Last-A-Foam® polyurethane insulation, aluminum honeycomb spacers, aluminum LDFs, and an SS 5CV. The double-O-ring-sealed 5CV is removable for loading and unloading.

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 Model 9978 Package conforms to the general standards for all packages as prescribed by 10 CFR 71, i.e., §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 Appendices 5.1 and 5.2 are consistent with ANSI 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

Radiation sources were derived for Content Envelopes C.1, C.2, and C.4 to C.6, with C.3 being reserved for future implementation. Content Envelope C.1 is a ^{238}Pu source with other Pu isotopes and other actinides being present in small quantities. The source used 100 g of ^{238}Pu, which is three times the mass that would be allowed to limit the decay heat rate to 19 W. The source term was evaluated in a bounding manner with only ^{238}Pu and ^{236}Pu (1 ppm) with 0.005 g of beryllium.

Content Envelope C.2 was modeled with Pu isotopes and ^{241}Am, the latter being limited to a mass equivalent to 19 W decay heat. The ^{240}Pu mass was limited to 25% of the total mass of 4,500 g (i.e., an extra 100 g over the limit of 4,400 g was used). The source term was evaluated at 153% of the maximum allowed mass of 4,400 g. The source term for the uranium metal content of C.2 was bounded by the Content Envelope C.4, which contains 13,500 g of uranium metal. The ^{236}Pu was optimally decayed to maximize the ^{208}Tl content.

The Content Envelope C.6 source term was obtained by summing up the individual contributions of the various isotopes that could be present.

The Staff concurs with this conservative approach to establish the various source terms for both gamma rays and neutrons.

5.3.2.1 Gamma Source

The SARP uses a combination of the ORIGEN-S code[5-5] and the Radiation Source Term Analysis (RASTA) code[5-6] to calculate source terms. The ORIGEN-S code is part of the NRC-sponsored Standardized Computer Analyses for Licensing Evaluation (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.

The Staff concludes that the gamma source terms have been appropriately calculated using these codes.

5.3.2.2 Neutron Source

The RASTA code was used by the SARP to obtain the neutron source terms. The effect of subcritical multiplication was not included in the source term but was accounted for in the radiation-transport calculations.

The Staff agrees with this approach and has determined that the neutron source terms for the various content envelopes were appropriately calculated.

5.3.3 Shielding Model

The Model 9978 Package shielding model uses the 6CV model used for the Model 9977 Package shielding analysis, although the actual containment vessel in the Model 9978 Package is a 5CV. However, because the original 6CV shielding model placed the bottom of the CV at a distance where the bottom of the 5CV would be in the absence of the bottom spacer, this model was used by the SARP. Because the source is placed either at the bottom of the CV or in the center of the

CV (for the side dose), this model was deemed appropriate for the Model 9978 Package shielding analyses. The slight difference in the thickness of the wall will have a negligible effect on the calculated dose rates.

The drum consists of an SS outer shell with an SS liner, aluminum spacers, aluminum LDFs, and miscellaneous other hardware. The drum is modeled as a right circular cylinder with some of the components simplified. These simplifications do not affect the analysis or add conservatism because they place the source material closer to the surface of the drum. The simplifications include:

- The bottom of the drum is modeled as flat rather than convex.
- The drum top rim and bottom wear ring are not modeled.
- The drum rolling hoops are not modeled.

The CV was modeled as a cylindrical main portion with a conical transition at the top, a short upper cylinder, a Cone-Seal Plug, and the Cone-Seal Nut. Some of the components of the CV that are easy to model (e.g., the conical transition at the top) are included exactly. Other components that are more complex are modeled as simpler shapes and are discussed below.

- The gland nut is a complex set of cones and cylinders outside the gland plug. For simplicity the gland nut and gland plug are modeled together as a single cylinder of 304L SS, with a small cylindrical cavity at the top. This does not impact the radiation-transport calculations.
- The end cap of the CV is modeled as a 2/1 ellipse with the axis of rotation (the z-axis) set to half the radius of the CV cylindrical portion and the minor axis set to the radius of the CV cylindrical portion.

Because the 6CV model was used for shielding analyses, the aluminum spacers were not included in the model. The NCT dose rates were calculated with the model representing the intact package. For the HAC dose rates, the model assumed that only the CV remains intact.

The dose-rate point locations in the shielding model are given at the package surface and at 1 m from that surface, as prescribed in §71.47(a) for NCT non-exclusive use shipments. Also, the dose-rate 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 a manner such that they would add to the conservatism of the model. Both surface-flux and point-detector estimators were used to calculate the dose rates.

The Staff finds this approach acceptable for the purposes of calculating the external radiation levels for this package.

5.3.3.1 Configuration of Source and Shielding

The Model 9978 Package does not have any materials that are designed to provide neutron or gamma shielding. Attenuation of the radiation from the source is achieved by the distances from the source to the points of dose-rate measurements external to the package. Sources are modeled as spheres of appropriate radii consistent with the mass of the payload and the density of the source material.

The Staff examined the source models used for the five Content Envelopes, with particular emphasis on the source models used for the two Content Envelopes that produced the largest external dose rates: C.2 and C.6. While the Content Envelope C.2 source dimensions were appropriately represented, the Content Envelope C.6 source dimensions used in the SARP analyses corresponded to the mass of ^{238}Pu. Because the plutonium contents in Content Envelope C.6 can potentially have all the plutonium isotopes, a larger mass of plutonium can lead to a higher subcritical multiplication factor. Further discussion of the impact of this can be found in Section 5.3.4.4 of this TRR.

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. 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.

The Staff finds the material properties used in the SARP analyses to be appropriate.

5.3.4 Shielding Evaluation

5.3.4.1 Methods

As discussed in Sections 5.3.2.1 and 5.3.2.2 of this TRR, RASTA and ORIGEN-S were used to evaluate the gamma and neutron source terms. The radiation-transport calculations were performed using Monte Carlo N-Particle Transport Code (MCNP)[5-8] and Evaluated Nuclear Data Files (ENDF)-B-VI[5-9] cross sections. The use of these codes and data is standard practice, and the Staff finds them to be acceptable.

5.3.4.2 Input and Output Data

All input and output files used in the SARP analyses have been provided to the Staff. In addition, sample input files have been included in Appendices 5.1 and 5.2 of the SARP. No output files have been included in the Appendices.

The Staff examined these files and found them to be correctly generated. The Staff also performed independent calculations for selected cases to verify the validity of these calculations.

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), 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 Staff examined the dose-rate calculations presented for the five different content envelopes. Of these content envelopes, the review of Content Envelope C.1 was already dealt with in connection with the Model 9977 Package.[5-10] Of the remaining four Content Envelopes, C.4 and C.5 have very low external dose rates. The two Content Envelopes with the largest external radiation levels are C.2 and C.6. As a result, the Staff performed alternate calculations for these two cases to confirm the validity of the SARP results.

For Content Envelope C.2, the source term was evaluated at 153% of the total allowed content, including masses of ^{238}Pu and of ^{241}Am that were individually equivalent to the total decay heat rate of 19 W. Therefore, a great deal of conservatism was present in the source term. The Staff repeated the calculations with a spherical source of radius 3.7832 cm, which is equivalent to 4,500 g total allowed mass (4,400 g + additional 100 g used in the shielding analyses) and the SARP mass limit of 25% ^{240}Pu. The comparison of the SARP calculations and the Staff calculations is given in the Table 5.1 below.

Table 5.1. Limiting Surface Dose Rates at the Bottom for Content Envelope C.2.

Calculation	Neutron Dose Rate (mrem/h)	Gamma Dose Rate (mrem/h)	Total Dose Rate (mrem/h)
SARP	174.2	22.3	198.3*
Staff	176.3	19.5	195.8

* Includes a 1 ppb contribution from ^{232}U of 1.8 mrem/h (from Table 16 of Appendix 5.1 of the SARP).

The two independent calculations agree well but are very close to the regulatory limit of 200 mrem/h. However, as stated above, the source terms used are extremely conservative, and the actual radiations levels are likely to be less than those predicted by the calculations. The Staff agrees that with the maximum 25% weight limit for ^{240}Pu in Content Envelope C.2, the package will meet regulatory limits.

The Content Envelope C.6 source term was determined by adding up the contributions of the individual items that constitute this content envelope. The SARP determined the dose rates based on a source of radius 0.34792 cm, which is derived from 3.5 g of ^{238}Pu metal. The actual contents of the source were a 50/50 mix of ^{239}Pu and ^{240}Pu. The SARP states that this is conservative because the smaller radius would have less self-shielding and would also keep the source closer to the detector location. The Staff is of the opinion that this is not necessarily valid for the side dose rate because a larger radius would actually increase the dose rate on the side, especially for neutrons that are only minimally self-shielded by the source material. The Staff is also of the opinion that potentially all of the plutonium isotopes can be present in the payload. This would increase the mass of the fissile material and consequently increase subcritical multiplication. The impact of the increased neutron dose can be offset by a decrease in the gamma dose, which is affected by self-shielding.

Table 5.2 compares the SARP estimation of the surface dose rate with the Staff's independent dose-rate calculation. Both of these estimates were obtained based on a source of mass 3.5 g.

Examining the table below, the total dose rate from the SARP is about 6% higher, while the neutron and gamma dose rates are within 10% of each other.

The Staff performed calculations to study the impact of the increased mass and of the consequent increase in the source radius. The Staff calculated the dose rates based on the total mass of the plutonium isotopes (10% increase in limit per Table 2 of Appendix 5.2), with a new radius of 1.4483 cm for the source sphere. Both the bottom and the side dose rates were calculated and compared to those for the smaller mass of 3.5 g.

Table 5.2. Limiting Surface Dose Rates at the Bottom for Content Envelope C.6.

Calculation	Neutron Dose Rate (mrem/h)	Gamma Dose Rate (mrem/h)	Total Dose Rate (mrem/h)
SARP	149.2	31.7	180.9
Staff	134.8	34.7	169.5

Table 5.3. Comparison of Dose Rates for Two Different Masses for Content Envelope C.6.

Calculation	Neutron Dose Rate (mrem/h)	Gamma Dose Rate (mrem/h)	Total Dose Rate (mrem/h)
3.5 g-bottom	134.8	34.7	169.5
252.5 g-bottom*	137.6	11.0	148.6
3.5 g-side	37.0	6.0	43.0
252.5 g-side*	47.3	4.6	51.9

* Mass of Pu isotopes represents a 10% increase over the original limits as presented in Table 2 of Appendix 5.2.

A comparison of bottom dose rates in Table 5.3 shows a small positive gain in the neutron dose rate for the higher mass due to the increased multiplication, which is partially offset by the larger average distance from the source center to the bottom of the package. In the case of the gamma rays, the larger distance and, consequently, the larger amount of self shielding, the dose rate is down by as much as a factor of three. Overall, the dose rate drops for the larger mass. In the case of the side, the neutron dose rate increases by about 30%, both because of increased multiplication and the fact that the increased radius brings the source closer to the side of the package. In the case of gamma rays, the dose rate is still lower, mainly because of self-shielding. However, now the total side dose rate is higher for the higher mass. In the case of the Model 9978 Package, the dominant dose rate is at the bottom surface; therefore, the bottom dose-rate values presented in the SARP based on the 3.5-g source mass are bounding.

The Staff performed a final calculation assuming that all of the contents of the Content Envelope C.6 are present in the source. The results from this calculation are given below in Table 5.4.

Table 5.4. Dose Rates on Contact based on all Isotopes from Content Envelope C.6.

Calculation	Neutron Dose Rate (mrem/h)	Gamma Dose Rate (mrem/h)	Total Dose Rate (mrem/h)
Bottom	64.3	2.3	66.6
Side	34.3	1.2	35.5

From the above results it is clear that the Content Envelope with the entire contents present has a much lower dose rate than that produced by the 3.5-g 50/50 ^{239}Pu/^{240}Pu source.

The HAC dose rates for all the Content Envelopes are all well below the regulatory limit of 1 rem/h at 1 m from the side and bottom of the package.

The Staff concludes that the SARP analyses have demonstrated that the external radiation levels for all five Content Envelopes would remain below regulatory limits.

5.3.5 Appendices

Appendix 5.1, *9978 Shielding Analysis*, N-CLC-G-00119, Revision 2, and Appendix 5.2, *C.6 Mass Limits Shielding Analysis*, N-CLC-G-00130, Revision 0, provide the details of the calculations performed to estimate the external radiation levels and provide selected input listings.

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. The Staff has previously pointed out that information not pertinent to the submittal is present in the Appendices. The applicant has agreed to note these at the beginning of each Appendix, and to make appropriate modifications in the next revision of these documents. The Staff recommends that the next revision be made in a timely manner.

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] NRC, 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.

[5-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000), Vienna, Austria (2000).

[5-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[5-4] 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 (1977).

[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] *MCNP—A General Monte Carlo N-Particle Transport Code*, Version 4C, J. F. Briesmeister, Ed., Los Alamos National Laboratory, LA-13709-M, RSICC Computer Code Collection CCC-701 (June 2001).

[5-9] *Evaluated Nuclear Data Files*, Versions B-V and B-VI, National Nuclear Data Center, Brookhaven National Laboratory (several revisions from 1973–2000).

[5-10] *Technical Review Report for the Safety Analysis Report for Packaging Model 9977 Package*, Lawrence Livermore National Laboratory, UCRL-TR-235385, September 30, 2007.

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6.0 Criticality Evaluation

6.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[6-1] and in IAEA Safety Standards Series No. TS-R-1.[6-2] The description and engineering drawings in Chapter 6, Criticality Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[6-3] were reviewed.

Included in the Criticality Review were the following:

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

6.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Criticality review of the Model 9978 Package include:

- 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)]
- A single package must be subcritical under the conditions of §71.55(b), §71.55(d), and §71.55(e).
- 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)]
- 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)]
- Unknown properties of fissile material must be assumed to be those that will credibly result in the highest neutron multiplication. [§71.83]

6.3 Review Procedures

Chapter 6 of the Model 9978 Package SARP includes the information essential for a criticality evaluation, including the drawings, the packaging materials and densities, and the fissile isotopic composition and mass. This criticality information in the Model 9978 Package SARP was reviewed by the Staff for completeness and compliance with regulatory requirements. Of particular importance are the subcriticality requirements per 10 CFR 71.55 and 10 CFR 71.59. The following subsections describe the detailed review for the *Areas of Review* applicable to the Criticality chapter of the SARP, as listed in Section 6.1 above.

6.3.1 Description of Criticality Design

6.3.1.1 Design Features

The Model 9978 Package is a single-containment, SS drum-type package with a bolted flange closure. Major components of the package inside the drum are a cylindrical SS 5CV, Fiberfrax® insulation, and Last-A-Foam®, which functions both as an insulation and as an impact absorber.

The fissile contents are generally placed directly in a product or convenience can. The product or convenience can may be placed in one or more low-density polyethylene bags, provided that no more than 100 g of polyethylene are involved in the case of food-pack cans for Content Envelopes C.1, C.2, C.4, C.5, and C.6. Plastics are also limited to 100 g in engineered containers and in the ICE container assembly, for Content Envelope C.6. Content Envelope C.1 (^{238}Pu heat sources) is packaged either in a food-pack can or engineered container. Plutonium/uranium metals (Content Envelope C.2) may be enclosed in 3013 or food-pack can configurations. Uranium metal (Content Envelope C.4) or uranium oxides (Content Envelope C.5) may be enclosed in a food-pack can container configuration, which, in turn, is placed within the 5CV. Samples and Sources (Content Envelope C.6) go into food-pack cans, engineered containers (such as sealed sources), or an ICE container assembly containing an ICE special nuclear material target. The design of the Model 9978 Package does not include any specific neutron-absorbing material (or spacers) for criticality control.

The package utilizes the geometry of the containment vessel and control of the quantity and composition of the fissile material to ensure that the single-package contents are subcritical under NCT and HAC. In addition to the control of the geometry and specific fissile content, interaction control is also established by the fact that the containment vessel is enclosed in a drum, ensuring a center-to-center separation of at least the diameter of the drum in the lateral direction (perpendicular to the drum axis). Furthermore, the hydrocarbon-based insulating–spacing material (foam with a nominal minimum density of 0.256 g/cm^3) acts as a neutron moderator, and acts to further isolate a package from neighboring packages. These features ensure that the arrays of packages are subcritical under NCT and HAC.

The Staff confirms that the text and sketches describing the criticality design features are consistent with the engineering drawings and the models used in the criticality evaluation. The Staff also concludes that the Model 9978 Package conforms to the general standards for all packages, as prescribed by 10 CFR 71, i.e., §71.31(a)(1), §71.31(a)(2), §71.31(c), §71.33, and §71.35(a). In addition, the Staff concludes that the SARP has assigned a proper CSI of 1.0 for all Content Envelopes, except Content Envelope C.1, for the Model 9978 Package with metal or oxide payloads, as prescribed by 10 CFR 71, i.e., §71.59, §71.35(b). Content Envelope C.1 has a CSI of 0.

The 5CV for the Model 9978 Package consists of a cylindrical structure with a nominal ~5.1-inch ID, ~18.6-inch overall height, and 313-cubic-inch internal volume. The 5CV complies with the stress criteria of the ASME B&PVC, Section III, Subsection NB. The 5CV is placed within a 35-gallon removable-head drum, constructed of Type 304L SS, with a minimum OD of ~18.35 inches and ~36.1 inches high (the drum rolling hoops are somewhat larger). The 5CV is enclosed by an approximate combined radial thickness of 4.95 inches of foam insulation and Fiberfrax® insulation materials. The combined axial thickness of insulating materials is 4.52 inches.

6.3.1.2 Codes and Standards

Please see Section 6.3.3.4, *Computer Codes and Cross-Section Libraries*, of this TRR.

6.3.1.3 Summary Table of Criticality Evaluation

The SARP summary table, Table 6.1, addresses the following cases for the Model 9978 Package: a single package, under the conditions of §71.55(b), (d), and (e); an array of undamaged packages, under the conditions of §71.59(a)(1); and an array of damaged packages, under the conditions of §71.59(a)(2). Table 6.1 includes the maximum value of the effective multiplication factor (k_{eff}) for Content Envelopes C.2, C.4, and C.5, including two standard deviations. It also lists the safe value for the multiplication factor (k_{safe}), for which the appropriate bias and bias uncertainty have been subtracted from 0.95 (which includes the accepted minimum subcritical margin of 0.05). It also lists the number of packages evaluated in the arrays. The table demonstrates appropriate subcriticality by showing that the value of k_{eff} is less than k_{safe} for that package and payload. The k_{eff} values are not explicitly calculated for Content Envelopes C.1 and C.6, as the criticality for C.1 is not credible and the Content Envelope C.6 is bounded by that of Content Envelope C.2, relative to criticality safety.

6.3.2 Fissile Material and Other Contents

The contents used in the criticality analyses are consistent with those specified in the General Information section of the SARP in Chapter 1. The density for any allowed fissile material is its maximum theoretical density. For the purpose of conservatism, the plutonium and uranium contents were assumed to be 100 weight percent ^{239}Pu and 100 weight percent ^{235}U, respectively.

The Staff notes that the limiting values of Content Envelope C.2 (4,400 g of Pu/U metal) are identical to those evaluated for the Model 9975-96 Package SARP. In addition, the limiting values of Content Envelope C.4 (13.5 kg of U metal/alloys) are also identical to a content evaluated for the Model 9975-96 Package SARP. It is also to be noted that the containment and drum sizes and compositions are also similar to those for the Model 9975-96 Package. The Model 9978 Package does not use a secondary containment vessel and associated lead shield. The Model 9978 Package also uses different types of insulating and impact-absorbing materials. However, their densities (especially the hydrogen density) are similar to that of Celotex® used in the Model 9975-96 Package. Therefore, the k_{eff} values for different scenarios are not expected to be significantly different from those reported in the Model 9975-96 Package SARP for identical contents.

6.3.3 General Considerations for Criticality Evaluations

6.3.3.1 Model Configuration

The configurations for the calculational models for a single package and for the arrays of packages used to perform the criticality evaluation of the Model 9978 Package are described in Section 6.3 of the SARP.

The criticality modeling for the Model 9978 Package makes several assumptions for the package models to be used for a single package. The SARP presents different package models for the NCT and HAC-array analyses.

The model for the single Model 9978 Package assumes that the 5CV is a simple cylinder. The nominal inner cylinder diameter is chosen for the 5CV and the drum. The calculational model assumes full water reflection of the 5CV, as required by 10 CFR 71.55(b).

For the single-package analyses, the fissile materials are treated as being spherical metal surrounded by water. Plutonium metal bounds plutonium oxide from a criticality standpoint. Also, the possible polyethylene/plastic bags surrounding the fissile material in metal form are considered by allowing a 100-g shell of C_2H_4 to surround the fissile sphere. These treatments maximize the reactivity.

For single-package calculations, the fissile materials are also treated as solution surrounded by water.

It was noted that the NCT and HAC models used the 6CV dimensions from the Model 9977 Package SARP instead of the 5CV dimensions from the Model 9978 Package SARP. The Staff agrees that this is a conservative assumption.

The NCT tests did not cause any damage to the Model 9978 Package that significantly affected criticality. Analyses reported in the SARP show that an infinite number of undamaged packages remain subcritical under the NCT conditions. The HAC tests did cause damage to the package that affected the criticality calculations. The HAC model conservatively took into account the insulation-burn-test and drop-test data. The displacements of the 5CV in neighboring packages in an array are treated to maximize their interaction and produce maximum reactivity. This is a very conservative treatment of the HAC damage.

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

For the HAC-array calculations, the fissile materials are located within the 5CV to give the closest interaction with respect to the fissile materials in other neighboring packages. This treatment maximizes the reactivity.

The closest packed array of Model 9978 Packages achievable is hexagonal in a lateral plane (perpendicular to the package axes), but square in the vertical direction for subsequent layers of packages. The Model 9978 Package SARP analyses used square arrays in both directions, but decreased the lateral pitch by 7% to account for this approximation in the lateral-plane layers.

Because the Model 9978 Package has no in-leakage occurring during HAC tests, the HAC-array calculation model assumes that the 5CV is dry. For the NCT and HAC calculations, the most reactive fissile material contents were used in the form of a dry sphere.

6.3.3.2 Material Properties

Accepted values for the density of all packaging materials are used in the Model 9978 Package SARP. The density for any allowed fissile material is its maximum theoretical density. For the purpose of conservatism, the plutonium and uranium contents were assumed to be 100 wt% of ^{239}Pu and 100 wt% of ^{235}U, respectively. One hundred grams of polyethylene material (with a density of 0.95 g/cm^3) was conservatively used for plastic materials. The SS material 304L is represented by SS 304. The effect of a slight variation in composition between 304L and 304 is acceptable because of their negligible effect on reactivity.

The Staff concludes that the fissile material properties for the Model 9978 Package conform to the requirements specified in 10 CFR 71.83.

Confirmatory calculations with an independent MCNP Code[6-4] verified the most limiting cases.

6.3.3.3 Demonstration of Maximum Reactivity

In the SARP, maximum reactivity was demonstrated for single packages with plutonium. Analyses of the configuration with the polyethylene shell give slightly more reactivity than without it. Maximum reactivity for single packages was also demonstrated with fissile material in solution. Confirmatory calculations verify these conclusions.

The most reactive individual package appropriate to the specific conditions was used for NCT and HAC-array analyses. Maximum reactivity was demonstrated for both NCT and HAC-array analyses for the mass and position of fissile material, and for internal and interspersed moderation. Confirmatory calculations verify this conclusion.

The reference 3013 configuration was found to be the most reactive configuration for the single package and was used to bound the other food-pack can type of containers. The SARP analyzed the effect of various combinations of flooding and reflection of the 5CV in determining the most reactive configuration. The trend is very similar to the earlier evaluation for the Model 9975-96 Package SARP.

The Staff confirms that the SARP has used the most reactive configuration in demonstrating subcriticality.

6.3.3.4 Computer Codes and Cross-Section Libraries

The containment vessels are leak tested to the ANSI N14.5-1997 Standard.[6-5] The ANSI/ANS 8.1[6-6] and ANSI/ANS 8.15[6-7] were also used to check the subcritical limits for a single package with plutonium and highly enriched uranium (HEU) contents. The SCALE 5[6-8]/KENO VI[6-9] code system operating on the Washington Safety Management Solutions (WSMS) Linux Workstation Cluster was used in the SARP. The criticality studies used the 238-group Evaluated Nuclear Data Files (ENDF)/B-V cross-section library with the CSAS26 driver in SCALE 5. BONAMI and CENTRM modules were used for the generation of the cross-section library, and the KENO-VI module was used to perform Monte Carlo k_{eff} calculations. These computer codes and cross-section libraries are appropriate for the criticality calculations, and are consistent with the neutron spectrum of the package. Also, these cross-section libraries properly account for resonance absorption and self-shielding effects. The benchmark evaluations and resulting biases were determined using the same codes and cross-section sets.

Independent confirmatory calculations were performed with MCNP 5 using the ENDF/B-VI cross-section set.

The SARP study used sufficient neutron histories to obtain the k_{eff} values within a statistical uncertainty of 0.002. The number of neutron histories was adequate to assure that the fissile systems analyzed were sampled in a statistically acceptable manner and that convergence was achieved.

6.3.4 Single-Package Evaluation

The Staff concludes that the Model 9978 Package conforms to the criticality requirements, as prescribed by 10 CFR 71, i.e., §71.43(f), §71.51(a), §71.55(b), §71.55(d), §71.55(e).

6.3.4.1 Configuration

The SARP determined that the maximum reactivity occurs when the 5CV in the Model 9978 Package contains a solid 4,400-g sphere of ^{239}Pu metal, with both completely surrounded by water (fully flooded), and with full water reflection of the containment vessel, as required in §71.55(b).

6.3.4.2 Results

This Content Envelope (C.2) was evaluated in great detail for the Model 9975-96 shipping container, and was approved for the Model 9975-96 Package SARP. Because the drum dimensions of the Model 9975-96 Package and those of the Model 9978 Package are similar, and the containment volumes are the same, it is expected that other differences (e.g., insulation type) will not produce a large difference in reactivity for identical contents. This is noted by the analysis results in the SARP.

For example, the k_{eff} value for the Model 9975-96 Package single-unit dry configuration with 4,400 g of plutonium in a 3013 container is 0.8509 (Case No. 7, Table 6.11 of Model 9975-96 Package SARP[6-10]), while the k_{eff} value for the Model 9978 Package SARP, single-unit dry configuration with 4,400 g of plutonium in a 3013 container is 0.8390 (Case No. 8, Table 6.14).

The Staff performed an independent confirmatory analysis with a different code (MCNP) for Case No. 1 in Table 6.14 of the Model 9978 Package SARP. The MCNP k_{eff} result is 0.8122, while the corresponding Model 9978 Package SARP result is 0.8080. The k_{eff} value of the corresponding case for the Model 9975-96 Package SARP (Case No. 6, Table 6.11) is 0.8100.

The maximum k_{eff} value for the single-unit plutonium metal case is 0.9082 (Case No. 10 in Table 6.14 of the Model 9978 Package SARP), and it bounds the solution-case k_{eff} values for the single unit.

The criticality results of the most reactive case for the single-package analysis are consistent with the information presented in the Summary Table discussed in Section 6.3.1.3 of this TRR.

ANSI-8.1 gives 5.0 kg of ^{239}Pu metal as the subcritical limit. The SARP argues that a single Model 9978 Package with a solid 4,400-g sphere of ^{239}Pu metal is subcritical because it is 600 g less than the ANSI-8.1 subcritical limit, and that the packaging surrounding the 5CV (lead, fiberboard, drum, etc.) is essentially less than 12-inches-equivalent of water.

Therefore, 4,400 g of ^{239}Pu metal in any configuration in a full, water-flooded 5CV and fully water-reflected containment vessel is appropriately subcritical. The Staff concurs with this assessment. This plutonium-metal content bounds 4,400 g of plutonium in 5.0 kg of plutonium oxide.

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

The NCT tests did not result in any water leakage into the containment system or damage that significantly affected the criticality of the packages. The Staff concludes that the Model 9978 Package is designed, constructed, and prepared for shipment so that there will be no significant reduction in the criticality safety of any package during NCT. The Staff also concludes that the Model 9978 Package conforms to the NCT criticality requirements for all packages, as prescribed by 10 CFR 71.59(a)(1), §71.59(a)(3).

6.3.5.1 Configuration

The SARP evaluated the most reactive dry fissile contents in an undamaged Model 9978 Package for the NCT analyses. The most reactive dry fissile content was a solid 4,400-g sphere of ^{239}Pu metal in a 5CV. The SARP analyses evaluated an infinite array of packages to demonstrate subcriticality. Some cases specifically modeled a 6×6×7 array of Model 9978 Packages with 30 cm of water reflection surrounding the array. It may be noted that the NCT cases were conservatively modeled with a 6CV instead of the 5CV in the Model 9978 Package SARP.

6.3.5.2 Results

The most reactive, dry individual Model 9978 Package was used for the NCT analyses. The array analyses reported in the SARP showed that an infinite array of packages, with each fissile mass located at the center of the 5CV in each package, is appropriately subcritical.

The Staff performed an independent confirmatory analysis with a different code (MCNP) for Case No. 2 in Table 6.17 of the SARP. The MCNP k_{eff} result is 0.8823, while the corresponding Model 9978 Package SARP result using the 6×6×7 array is 0.8741. The corresponding case (3013 dry, the rest flooded, infinite array) for the Model 9975-96 Package (Case No. 9, Table 6.19 in the Model 9975-96 Package SARP) is 0.8772. The confirmatory calculations used the actual hexagonal lattice packing for the lateral layers in order to confirm that the SARP results are acceptable.

The confirmatory calculation for 13.5 kg of uranium was also performed. The MCNP k_{eff} result for Case No. 1a in Table 6.17 of the SARP is 0.8025, while the corresponding Model 9978 Package SARP result is 0.7799.

Confirmatory analyses verify that the SARP conclusions are valid.

It is also noted that the 5CV remained *leaktight* following the prescribed accident case; therefore, Case No. 2 exceeds the analysis requirements for NCT conditions. All NCT cases were below the k_{safe} value of 0.931. A large reactivity margin is available for the NCT cases using a very conservative model.

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

The Staff concludes that the Model 9978 Package conforms to the HAC criticality requirements for all packages, as prescribed by 10 CFR 71.59(a)(2) and §71.59(a)(3).

6.3.6.1 Configuration

The SARP uses the most reactive contents in a damaged Model 9978 Package for the array calculations under HAC analyses. Because the Model 9978 Package has a single containment (with double O-ring seals) and did not leak during HAC tests, and because the Model 9978 Package containment vessel design for the 5CV complies with the stress criteria of the ASME B&PVC, Section III, Subsection NB,[6-11] the 5CV is assumed *leaktight*. Therefore, the contents are assumed to remain dry.

The most reactive fissile content is a solid 4,400-g sphere of ^{239}Pu metal within a 5CV. The most reactive configuration of packages in the HAC calculations is with no interspersed moderation between packages.

A 6×6×3 square pitch array was used in the HAC analysis. The drum dimensions were reduced by 7% to approximate an equivalent triangular pitch-array configuration. In addition, the drum dimensions were further reduced to incorporate the damaged drum parameters from the test results. The uniform reduction in drum dimensions was judged to be very conservative.

In addition, each 5CV moved laterally toward the vertical axis through the center of the two or four packages as much as allowed by the damaged condition of the insulation material. The two-cluster and four-cluster models allowed the movement of the 5CV, such that the distance between the fissile materials was minimized, thereby maximizing the interaction of fissile contents.

6.3.6.2 Results

The most reactive single Model 9978 Package with appropriate damage was used for the HAC. The dry configuration was found to be most reactive. This is consistent with the Model 9975-96 Package SARP results.

The Staff performed an independent confirmatory analysis with a different code (MCNP) for the most reactive case (Case No. 6 in Table 6.18 of the SARP). The MCNP k_{eff} result is 0.8990, while the corresponding SARP result is 0.8895. The corresponding case (3013 dry, the rest flooded, infinite array) for the Model 9975-96 Package (Case No. 2, Table 6.22 in Model 9975-96 Package SARP) is 0.9089. It may be noted that the HAC cases were conservatively modeled with a 6CV instead of the 5CV in the Model 9978 Package SARP. The confirmatory calculations used the actual hexagonal lattice packing for the lateral layers in order to confirm that the SARP results are acceptable. Confirmatory analyses and comparisons with the similar results from the Model 9975-96 Package SARP support that the SARP conclusions are valid.

The array analyses performed assumed that the plutonium sphere was located within each 5CV, and that each damaged 5CV-foam combination was displaced, so that the closest separation exists between fissile masses in neighboring packages. The most reactive array is, in addition, when no interspersed moderation is present between packages. This is a very conservative model.

The SARP analyses finds that a 6×6×3 array of HAC packages is appropriately subcritical. A CSI of 1.0 is determined for 108 packages being subcritical for HAC.

6.3.7 Criticality Safety Index for Nuclear Criticality Control

A minimum criticality CSI of 1.0 (for all contents except Content C.1) is assigned to the Model 9978 Packages, based on the HAC-array calculations showing that 108 packages in any configuration have a multiplication factor plus bias and bias uncertainties and a 5% minimum subcritical margin (MSM) that is less than 1.0. The CSI for the Content Envelope C.1 is 0.0. These CSI values are consistent with those reported in Chapter 1 on General Information in the SARP. The Staff concurs with these values.

6.3.8 Benchmark Evaluations

The Model 9978 Package SARP used the same criticality computer code, hardware, and cross-section library sets to determine the bias values from benchmark experiments as those used to calculate the multiplication factors for the packages.

6.3.8.1 Applicability of Benchmark Experiments

The benchmark experiments used in this study were taken from the various volumes of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*[6-12]and are appropriately referenced. This collection of benchmark experiments is the accepted standard in the criticality-safety-engineering community.

No critical experiments similar to the Model 9978 shipping package system are available. Therefore a wide range of critical experiments was chosen for validation.

The Staff concurs that the plutonium and uranium benchmark experiments are applicable to the actual packaging design and contents.

6.3.8.2 Bias Determination

The United States Nuclear Regulatory Commission (NUREG) document NUREG/CR-5661[6-13] recommends a minimum subcritical margin of 0.05 for packaging applications. Because a large number of critical experiments were used in bias determination, the additional use of the "areas of applicability margin" is not needed. Contributions from uncertainties in experimental data are included for all benchmark experiments reported in the Handbook. Also, a sufficient number of appropriate benchmark experiments are analyzed, and the results of these benchmark calculations are used to determine an acceptable bias and bias uncertainty for each fissile payload. These bias values are then used in the calculation of a k_{safe} value for the package payloads. The statistical and convergence uncertainties of the benchmark calculations and package evaluations are essentially consistent.

The SARP determined an acceptable value for the bias for plutonium metal/uranium metal/solution. The most limiting value of 0.931 was chosen as the k_{safe}. Acceptable statistical analyses demonstrate that this value is accurate and conservative.

The Staff concurs that the benchmark experiments and corresponding bias value are applicable, and conservative, as applied to the Model 9978 Package.

6.3.9 Appendices

There is one Appendix for Chapter 6, N-NCS-A-00014, Revision 1, *Nuclear Criticality Safety Evaluation: 9977 Shipping Package Analysis for SARP*. The Appendix also provides various input and output files on a compact disc.

6.4 Evaluation Findings

6.4.1 Findings

Based on review of the statements and representations made in the SARP, the Staff concludes that the nuclear criticality safety design has been adequately described and evaluated, and that the Model 9978 Package design has been shown to meet the subcriticality requirements of 10 CFR 71.

6.4.2 Conditions of Approval

Section 5(a)(3) of the CoC must contain the restriction that the Model 9978 Package must be constructed as specified on the engineering drawings in the SARP. The CoC must also contain the restriction that the contents be bounded by Table 1.2 of the SARP. Also, the package is *not* authorized for transport by air.

6.5 References

[6-1] NRC, 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.

[6-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000) Vienna, Austria (2000).

[6-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[6-4] *MCNP5—A General Monte Carlo N-Particle Transport Code, Version 5*, X-5 Monte Carlo Team, LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, New Mexico (April 2003).

[6-5] Institute of Nuclear Materials Management, *Leakage Tests on Packages for Shipment*, ANSI N14.5-1997, American National Standard for Radioactive Materials, New York (1997).

[6-6] American Nuclear Society, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, ANSI/ANS-8.1-1998 (1998).

[6-7] American Nuclear Society, *Nuclear Criticality Control of Special Actinide Elements*, ANSI/ANS-8.15-1981 (1981).

[6-8] *RISC Computer Code Collection, SCALE 5, Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, CCC-725, Oak Ridge National Laboratory, Oak Ridge, Tennessee (April 2005).

[6-9] Hollenbach, D.F.; Petrie, L.M.,; and Landers, N.F., *KENO-VI: A General Quadratic Version of the KENO Program*, ORNL/TM-2005/39, Version 5, Volume II, Book 3, Section F17, Oak Ridge National Laboratory, Oak Ridge, Tennessee (April 2005).

[6-10] Washington Savannah River Company, *Safety Analysis Report for Packaging, Model 9975*, S-SARP-G-00003, Revision 0, Savannah River Packaging Technology, SRNL (January 2008).

[6-11] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

[6-12] Nuclear Energy Agency, *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, Organization for Economic Co-Operation and Development, NEA/NSC Doc (95) 03 (September 2008).

[6-13] NRC, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages*, NUREG/CR-5661, Washington, DC (April 1997).

7.0 Package Operations Evaluation

7.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[7-1] and in IAEA Safety Standards Series No. TS-R-1.[7-2] The description and engineering drawings in Chapter 7, Package Operations Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[7-3] were reviewed.

Included in the Package Operations Review were the following:

7.1.1 Package Loading

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

7.1.2 Package Unloading

- Receipt of Package from Carrier.
- Removal of Contents.

7.1.3 Preparation of Empty Package for Transport

7.1.4 Other Operations

7.1.5 Appendices

7.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Package Operations review of the Model 9978 Package include:

- 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 (CSI) 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 for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 7.1 of this TRR.

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, *Package Loading*, 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 CoC.

7.3.1.2 Loading of Contents

The procedures for loading the contents into the package are contained in Section 7.1.2, *Loading of Contents*, 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 Envelopes C.1 through C.6.
- 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 add fill gases during the loading of the 5CV, such requirements are not necessary for shipments under the specifications for Content Envelopes C.1 through C.6.
- 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 50 (+10/–0) ft-lbs, and that the Gland Nut for the Leak-Test Port Plug 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, *Preparation for Transport*, 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: Because this is a DOE package, and because 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] are not applicable.)
- Procedures are in place to ensure that the preshipment 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 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 Model 9978 Package 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 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 Model 9978 Package does not incorporate the use of such features. The impact limiter is part of the drum.
- 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, that 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] or in site-specific/facility-specific procedures, as appropriate.
- Although there are no specific requirements that identify any special controls that 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 or in site-specific/facility-specific procedures, as appropriate.
- Although there are no specific requirements that identify any special controls that 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 or in site-specific/facility-specific procedures, as appropriate.
- 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 Model 9978 Package has been determined to be 1.0.

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, *Receipt of Package from Carrier*, 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, *Removal of Contents*, 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 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, *Preparation of Empty Package for Transport*, 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, *Limited Quantities of Radioactive Materials*, or 49 CFR 173.428, *Empty Class 7 (radioactive) Materials Packaging*, 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).

- 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, *Quality Assurance Records*, of the SARP).

7.3.5 Appendices

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

- Appendix 7.1, *Special Tools*, provides a list of required equipment needed for the operation of the Model 9978 Package. Appendix 7.1 also provides a list of non-commercial equipment that the user may find useful for the operation of the Model 9978 Package. (The applicant has also noted that illustrations and design drawings are available for all such non-commercial equipment.)
- Appendix 7.2, *EM-60 Radioactive Material Package User-Registration Form*, was provided by the applicant to register directly with EM-63 as a potential user of the Model 9978 Package. Note: The applicant can go to the Radioactive Materials Packaging (RAMPAC) website to find the on-line registration form.

7.4 Evaluation Findings

7.4.1 Findings

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

The Staff has concluded that the generic operating procedures delineated in Chapter 7 of the SARP—because they represent the framework from which the formal, site-specific operating procedures will be developed for each user/shipper—should be incorporated, in their entirety into the CoC as a Condition of package Approval.

7.5 References

[7-1] NRC, 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.

[7-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000), Vienna, Austria (2000).

[7-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

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

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

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

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

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8.0 Acceptance Tests and Maintenance Program Review

8.1 Areas of Review

This review includes an evaluation of the SARP with respect to the requirements specified in 10 CFR 71[8-1] and in IAEA Safety Standards Series No. TS-R-1.[8-2] The description and engineering drawings in Chapter 8, Acceptance Tests and Maintenance Program Review of the Safety Analysis Report for Packaging (the SARP), Model 9978 Package, B(M)F-96,[8-3] were reviewed.

The following elements of the Acceptance Tests and Maintenance Program Review 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

8.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Acceptance Tests and Maintenance review of the Model 9978 Package include:

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 chapter of the SARP for the Model 9978 Package. These procedures correspond to the *Areas of Review* listed above in Section 8.1 of this TRR.

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 9978 Packaging*, and the independent inspection verification detailed in Appendix 8.2 of the SARP, *Packaging Independent Verification Items*, ensure that newly fabricated Model 9978 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.[8-4] The applicant has further noted that the inspection methods, weld procedures, personnel qualifications, and weld reports shall meet the requirements of ASME B&PVC, Section V,[8-5] and of ASME B&PVC, Section III, Subsection NB[8-6] or Subsection NF,[8-7] as appropriate. (See Section 8.4.1 of this TRR below.)

8.3.1.3 Structural and Pressure Tests

The applicant has noted that each 5CV shall be hydrostatically proof tested at an internal pressure of 1,365 ±10 psig per the *Five Inch Diameter Containment Vessel Assembly* Drawing (R-R2-G-00043). The applicant has further noted that: (1) at the specified test pressure, the joints, connections, and external areas of the 5CV shall have no leaks; (2) no permanent deformation of any of the 5CV parts is permitted; and (3) any 5CV not in compliance with the acceptance criteria shall be dispositioned in accordance with Section 9.15, *Nonconforming Materials, Parts, or Components*, of the SARP. Finally, the applicant goes on to note that the test pressure for the 5CV 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-00044). (See Appendix 1.1 of the SARP.)

8.3.1.4 Leakage Tests

The applicant has noted that the containment boundary of the 5CV will have been leak-rate tested, with helium, in accordance with the evacuated-envelope method specified in Section A.5.4 of the 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} ref·cm^3/s (i.e., 1×10^{-7} ref·cm^3/s, which is ~2×10^{-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} ref·cm^3/s, which is also in keeping with the requirements specified in ANSI N14.5.

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

8.3.1.5.1 Component Tests

There is a specific subsection for Component Tests. The applicant did, furthermore, subdivide their subsection on *Component 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*; and 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 Model 9978 Packaging does not incorporate valves, rupture discs, or fluid transport devices.

Under the subheading of *Gaskets*, the applicant has noted that the 5CV O-ring seals are the only gaskets used in the Model 9978 Packaging. 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.

8.3.1.5.2 Material Tests

There is no specific subsection for Material Tests. (See 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 of this TRR.

8.3.2 Maintenance Program

The applicant has noted that the Model 9978 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 quality assurance (QA) records, that the inspection and testing requirements, presented in Section 8.2 of the SARP, 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&PVC, Section III, Subsection NB or NF, as appropriate. 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 Model 9978 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, *Structure and Pressure Tests*, 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 Leak-Test Port Gland Nut, Leak-Test Port Plug, or the O-rings with equivalent items does not constitute a structural modification, hence does not require pressure testing of the Containment Vessel.

8.3.2.2 Leakage Tests

The applicant has subdivided this subsection in two. 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, 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 per subclause 7.6 of ANSI N14.5 to verify that the Cone-Seal Assembly has been installed properly. The acceptance criterion is a measured leak rate not greater than 1×10^{-3} ref·cm^3/s air. The post-load test shall implement the gas pressure rise test per ANSI N14.5 method, A.5.2, for the Outer O-ring and Leak-Test Port Plug."

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/Refurbishment of containment vessel components, including:
>   - Cone-Seal Plug
>   - Cone-Seal Nut
>   - Outer O-ring Seal
>   - Leak-Test Port Plug"

The applicant goes on to note that:

> "The annual leak-rate test measures the rate that helium leaks through the Outer O-ring closure seal of the CV, the Leak-Test Port Plug, the Cone-Seal Plug, and the CV weldment via the evacuated-envelope method (A.5.4) of ANSI N14.5.
>
> "Upon completion of the annual 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]. At a minimum, the annual leak-test record shall include the label data and the following data for each replacement O-ring:
>
> - Material
> - Size
> - Date of manufacture"

(Note: Under the broader heading of the *Maintenance Program* in SARP Section 8.2, 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,[8-9] where Component and Materials Tests are combined into a single subsection. The applicant did, however, note that the Model 9978 Packaging drum, drum liner, and drum-closure lid are SS 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. In addition, the applicant noted that the Pre-loading inspection requirements described in Section 7.1.1 of the SARP will segregate the units that need repair.

8.3.2.3.2 Material Tests

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

8.3.2.4 Thermal Tests

The applicant has noted that annual thermal performance is not required for the Model 9978 Packaging.

8.3.2.5 Miscellaneous Tests

The applicant has subdivided this section into three 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 *Cone-Seal 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-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&PVC, 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 5CV 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 5CV 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 5CV 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 5CV can be used for shipping.

Under this same subheading, the applicant has added a precautionary note to remind users to avoid excessive lubrication. The note goes on to state that an excess of the high viscosity grease can promote a temporary hydro-lock condition, and hence, result in relaxation of the applied closure torque over a period of time.

Under the additional subheading of *O-rings*, the applicant notes that, prior to closure of the 5CV, the two 5CV 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-00043, *5-Inch Diameter Containment Vessel Subassembly,* Item 8.

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, noting that spare-part Viton® GLT or GLT-S O-rings shall be received and stored by the shipper in accordance with Society of Automotive Engineers (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 the traceability of each O-ring.

The applicant then goes on to reiterate that the Outer O-ring is part of the 5CV containment boundary and that, if the Outer O-ring is replaced, a Maintenance Leak-Rate Test must be performed per Section 8.2.2.2 of the SARP, with the new O-ring installed before the 5CV 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 5CV containment boundary.

Under the basic subheading of *Shielding*, the applicant has noted that the design of the Model 9978 Package does not incorporate shielding, therefore no annual maintenance of shielding integrity is required.

Under the basic subheading of *Cone-Seal Assembly Maintenance*, the applicant has noted that, prior to the annual leak-rate test described in Section 8.2.2.2 of the SARP, the threaded surfaces of the Cone-Seal Nut (Figure 8.3 of the SARP) 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, 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 three appendices associated with Chapter 8 of the SARP:

- Appendix 8.1, *Visual Inspection and Fabrication Verification Requirements for the 9978 Packaging*, provides the individual visual-inspection and fabrication-verification criteria for the Drum Assembly, Lid Weldment Subassembly, Drum Liner, Drum Weldment, the Drum and Liner Weldment, Drum Foam Installation Subassembly, CV Assembly, CV Weldments, Cone-Seal Nut, Cone-Seal Plug, CV Subassembly, and Overall Assembly.
- Appendix 8.2, *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, *Acceptance Tests for Polyurethane Foam in the 9978 Packaging*, provides the detailed acceptance-test criteria for the polyurethane foam for a specific foam formulation (compressive modulus, thermal conductivity, and specific heat), a batch (flame retardancy, intumescence, and leachable chlorides), and a pour (density and compressive modulus).

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 above (Section 8.1.2 of the SARP), where the applicant has stated "Inspection methods, weld procedures, personnel qualifications, and weld reports…," the ASME B&PVC, 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 8.3.2.2 above (Section 8.2 of the SARP), 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 Sections 8.3.1.5.2 above (for Acceptance Tests, Section 8.1.5 of the SARP) and 8.3.2.3.2 above (for the Maintenance Program, Section 8.2.3 of the SARP), it is noted there are no specific subsections for *Material Tests*. In addition, in Section 8.3.1.5.1 above (Section 8.1.5.2 of the SARP 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.

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 Model 9978 Package are adequate to assure that the packaging 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 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] NRC, 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.

[8-2] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000), Vienna, Austria (2000).

[8-3] SRNL, *Safety Analysis Report for Packaging, Model 9978 B(M)F-96*, S-SARP-G-2002, Revision 1 (March 2009).

[8-4] American Society for Nondestructive Testing, *Recommended Practice No. SNT-TC-1A — Non-Destructive Testing* (January 2001).

[8-5] ASME, *ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination*, New York, NY (2004).

[8-6] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

[8-7] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NF, Supports*, New York, New York (2004).

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

[8-9] NRC, 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).

9.0 Quality Assurance Review

9.1 Areas of Review

This Technical Review Report documents the review of Chapter 9, QA, of the Safety Analysis Report for Packaging, Model 9978, 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 IAEA Safety Standards Series No. TS-R-1.[9-3]

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.

9.1.3 Appendices

9.2 Regulatory Requirements

The requirements of 10 CFR 71 applicable to the Quality Assurance review of the Model 9978 Package include:

- The application must describe the quality assurance program for the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the package. [§71.31(a)(3), §71.37]
- The application must identify established codes and standards proposed for the package design, fabrication, assembly, testing, maintenance, and use. In the absence of any codes and standards, the application must describe the basis and rationale used to formulate the package quality assurance program. [§71.31(c)]
- Package activities must be in compliance with the quality assurance requirements of Subpart H (§71.101-§71.137). A graded approach is acceptable. [§71.101(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 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 for the Model 9978 Package. 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, inspection, testing, operation, maintenance, repair, and modification of the Model 9978 Package 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 Model 9978 Packaging materials, equipment, and services from suppliers that have been evaluated and approved to meet the applicable elements of 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] and WSRC Quality Assurance Manual (WSRC 1Q Manual)[9-6] document the QA program's compliance with 10 CFR 71, Subpart H, as well as with 10 CFR 830, Subpart A,[9-7] DOE O 414.1.C,[9-8] DOE O 460.1B,[9-9] and NQA-1. The WSRC 1Q Manual 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 Criteria

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 packaging structures, systems, and components (SSCs) that are important to safety. Safety-related "Q" packaging components are categorized as A, B, or C, 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 Model 9978 Packaging to the safety designations in the NRC's Regulatory Guide 7.10.[9-10]

Packaging SSCs and their "Q" safety categories, functions, and drawing/part nomenclature 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 "Q" categories A, B, and C, 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 Model 9978 Package activities related to the design, procurement, fabrication, handling, shipping, storage, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of the Model 9978 Package. The graded approach, described in Section 9.3.2.1 above, is used to selectively apply the QA controls to packaging SSCs based on their importance to safety.

As required by §71.31(a)(3), §71.31(c),and §71.37, Table 9.5 of the SARP details the materials, design, fabrication, testing, examination, QA program, and records requirements for the CV that conform to Section III, Division 1, Subsection NB, of the ASME B&PVC.[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&PVC.[9-12]

Section 9.6 of the SARP identifies documents that are controlled to ensure that 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 that sufficient written records are maintained to furnish evidence of the quality of the Model 9978 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

Based on our 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. Packaging-specific requirements are adequate to assure that the package is designed, procured, fabricated, handled, shipped, stored, cleaned, assembled, inspected, tested, operated, maintained, repaired, and modified 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, inspection, testing, operation, maintenance, repair, and modification of the Model 9978 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] SRNL, *Safety Analysis Report for Packaging*, Model 9978 B(M)F-96, S-SARP-G-2002, Revision 1 (March 2009).

[9-2] NRC, 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] IAEA, *Regulations for the Safe Transport of Radioactive Material, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, 1996 Edition (as amended 2000), Vienna, Austria (2000).

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

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

[9-6] Westinghouse Savannah River Company, *Quality Assurance Manual*, WSRC-1Q, Aiken SC (October 2005).

[9-7] DOE, 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.

[9-8] DOE, *Quality Assurance*, DOE O 414.1C, Washington, DC (June 2005).

[9-9] DOE, *Packaging and Transportation Safety*, DOE O 460.1B, Washington, DC (April 2003).

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

[9-11] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NB, Class I Components*, New York, New York (2004).

[9-12] ASME, *ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components*, *Division I, Subsection NF, Supports*, New York, New York (2004).

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