Department of Energy
Washington, DC 20585

SAFETY EVALUATION REPORT

FOR THE

NAC-LWT SHIPPING CASK

Docket 98-17-9225

Approved:

Michael E. Wangler

Michael E. Wangler, Director
Package Approval and Safety Program
Office of Site Operations, EM-70

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1. GENERAL INFORMATION REVIEW

1.1 Review Procedures

The following areas applicable to the General Information chapter of the SARP for the NAC International (NAC) Legal Weight Truck (LWT) package (NAC-LWT) were reviewed.

1.1.1 Introduction

1.1.1.1 Purpose of Application
The application addressed by this Safety Evaluation Report (SER) requests DOE approval of the NAC-LWT to transport tritium-bearing Lead Test Assemblies (LTAs), discussed in Section 1.1.2.2 below. The application consists of a Safety Analysis Report for Packaging (SARP), with amendments and other supplemental information listed in Section 1.3 of this SER. The NAC-LWT is certified by the Nuclear Regulatory Commission (NRC) for transport of spent nuclear fuel. This application is the initial submission of the package for DOE certification. The DOE review summarized in this SER does not address the use of the NAC-LWT for any contents other than the LTA contents, as described in more detail below.

1.1.1.2 Summary Information
The NAC-LWT is certified as USA/9225/B(U)-85 (DOE). The primary radionuclide of the contents is tritium. Because the tritium is produced by irradiation in a reactor, the contents also include activated stainless steel and crud, each having ^{60}Co as the predominant radionuclide. Other radionuclides are present in very small quantities due to activation of impurities in LTA materials. The total inventory of radionuclides meets the definition for a highway route controlled quantity of Class 7 (radioactive) materials.

The package is intended for transport by either exclusive-use or nonexclusive-use shipment. Section 1.2.3 of the SARP indicates that four packages will be shipped by highway vehicle with one package per truck/tractor. Prior to transport, the NAC-LWT is placed within an ISO container or personnel barrier, neither of which is considered a package component.

The package category is not specifically indicated in the SARP. However, because the package is intended to transport an excess of 1.1 PBq (30,000 Ci) of tritium, the guidance for Category I is applicable. (This category is consistent with that for spent nuclear fuel.)

Quality assurance is discussed in Section 9 of this SER.

The package contains no fissile material.

1.1.2 Package Description

1.1.2.1 Packaging
The packaging is described in Sections 1.2.1 and 1.2.2 of the SARP, with drawings provided in Section 1.4.

The NAC-LWT is a steel-encased, lead-shielded shipping cask designed to transport one LTA of tritium-producing burnable absorber rods (TPBARs) and a fuel assembly skeleton in a pressurized water reactor (PWR) fuel basket with a basket spacer and shield plug. The overall dimensions of the package, including impact limiters, are 589 cm (232 in.) long by 165 cm (65 in.) in diameter. The cask body is approximately 508 cm (200 in.) in length and 112 cm (44 in.) in diameter. The volume of the cavity is approximately 411 liters (14.5 cubic feet).

The cask body consists of a 1.91-cm (0.75-inch) thick stainless steel shell, a 14.6-cm (5.75-in.) thick lead gamma shield, a 3.1-cm (1.2-in.) thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded to a 10.1-cm (4-in.) thick stainless steel bottom end forging and to a stainless steel top end forging. The cask bottom consists of a 7.6-cm (3-in.) thick, 52.71-cm (20.75-in.) diameter, lead disk enclosed by an 8.9-cm (3.5-in.) stainless steel plate and bottom end forging. The cask lid is a 28.7-cm (11.3-in.) thick stainless steel stepped design, secured to a 36.20-cm (14.25-in.) thick ring forging with twelve 2.5-cm (1-in.) diameter bolts. The lid seal is a metallic O-ring, with a second (Teflon) O-ring and test port for leak testing. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and Teflon O-rings. A test port on each cover enables leak testing.

The neutron shield tank consists of a 0.61-cm (0.24-in.) thick stainless steel shell with 1.27-cm (0.50-in.) thick end plates. The shield region is 417 cm (164 in.) long and 12.7 cm (5-in.) thick. For transport of the contents addressed in this SER, the shield tank is not filled.

The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 165.7 cm (65.25 in.) and a maximum thickness of 70.6 cm (27.8 in.). The bottom impact limiter has an outside diameter of 153.0 cm (60.25 in.) and a maximum thickness of 71.9 cm (28.3 in.). Both impact limiters extend 30.5 cm (12 in.) along the side of the cask body, and each is attached with four attachment lugs.

A tamper-indicating device is looped through a hole near the end of one ball-lock pin, which connects the impact limiter attachment lugs to the mating cask lugs. This seal provides indication of any authorized opening of the lid or port covers, which are covered by the upper impact limiter during transport.

The maximum weight of the package is 23,583 kg (52,000 lbs.), and the weight of the LTA contents (including basket, spacer, and shield plug) is approximately 590 kg (1300 lbs.).

1.1.2.2 Contents

The contents are described in Section 1.2.3 of the SARP. The contents addressed by this certificate include a PWR basket, basket spacer, shield plug, and one LTA in a fuel assembly skeleton.

The LTA consists of a hold-down assembly with 16 Zircaloy-4 thimble plugs and 8 undamaged TPBARs, each mechanically attached to the 3/8-inch-thick stainless steel hold-down assembly plate.

The TPBARs are physically similar to a PWR burnable poison rod. Top and bottom end plugs are welded to each end of a 0.381-inch-diameter stainless steel tube. The tube is lined on the inside with an aluminide coating to prevent inward diffusion of hydrogen from the reactor coolant. The TPBAR tubes are loaded with annular shaped pencils, containing lithium aluminate ($LiAlO_2$) enriched with ^6Li for tritium production during irradiation. Each pencil is clad with a Zircaloy-4 (NPZ) tritium getter, which is clad on both the inside and outside surfaces with nickel to prevent oxidation. In order to limit tritium leakage through the uncoated welds of the tube end plugs, NPZ disks are held in place at the top and bottom of the TPBAR with a spring. The maximum length of a TPBAR is 152 inches, with a maximum active lithium length of 144 inches.

The maximum quantity of tritium in an LTA is limited to 9.6 g (1.2 grams per TPBAR). TPBARs are irradiated for a maximum of 500 effective full-power days in a commercial PWR and cooled for a minimum of 60 days prior to transport. The maximum heat load for an LTA is 55 watts.

The LTA is loaded into an LTA container for structural support during transport. The LTA container is essentially an unirradiated 17x17 PWR fuel assembly skeleton that has been reinforced by 44 open-ended Zircaloy 4 tubes in 44 fuel rod positions. The top and bottom nozzles are stainless steel, and the grids are Inconel.

The LTA container and LTA itself are collectively referred to as the LTA contents.

1.1.3 Appendix

The SARP does not include an appendix to Chapter 1. Drawings for the NAC-LWT packaging are included in Section 1.4 of the SARP. The following drawings are applicable to the NAC-LWT for the transport of the LTA contents:

315-40-01		Rev. 2	NAC-LWT Cask Assembly
315-40-02		Rev. 6	NAC-LWT Cask Body Assembly
315-40-03	Sheets 1-6	Rev. 9	NAC-LWT Cask Body
315-40-04		Rev. 6	NAC-LWT Cask Assembly
315-40-05		Rev. 4	NAC-LWT Upper Impact Limiter
315-40-06		Rev. 4	NAC-LWT Lower Impact Limiter
315-40-08	Sheets 1-3	Rev. 7	NAC-LWT Parts Detail
315-40-09		Rev. 2	NAC-LWT PWR Basket Spacer
315-40-10		Rev. 2	NAC-LWT Fuel Basket Assembly
315-40-093	Sheets 1-2	Rev. 0	Shield Plug, LTA Shipment, LWT Cask
315-40-095		Rev. 0	NAC-LWT Cask Assembly, LTA Shipment

1.2 Evaluation Findings

1.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the design of the NAC-LWT has been adequately described to meet the requirements of 10 CFR Part 71. This description also demonstrates that the NAC-LWT meets the minimum size limitations and contains an anti-tampering device required by the regulation.

1.2.2 Conditions of Approval

The conditions of approval include:

Specification of authorized contents,

List of drawings given in Section 1.1.3 of this SER, and

Specification that the package may be transported only by exclusive use since the package will contain a highway route controlled quantity of Class 7 (radioactive) materials.

1.3 References

NAC International. Safety Analysis Report for the NAC Legal Weight Truck Cask, Revision LWT-99H, including Attachments 1-4, June 1999.

NAC International. Letter to U.S. DOE, July 22, 1998, #ST980979. Subject: Amendment Submittal.

NAC International. Letter to U.S. DOE, December 22, 1998, #ST981376. Subject: Revised Amendment Submittal.

NAC International. Letter to U.S. DOE, April 5, 1999, #ST990543. Subject: Responses to Q1s, with Enclosures 1-5.

NAC International. Letter to U.S. DOE, June 9, 1999, #ST990543. Subject: Responses to Q2s, with Enclosure.

NAC International. Letter to U.S. DOE, June 24, 1999, #ST990543. Subject: Responses to Q3s, with Enclosure.

U.S. Department of Energy. Request for Additional Information (RAI 0) on the amendment 98G July 1998, October 14, 1998.

U.S. Department of Energy. Q1 Questions on NAC-LWT Cask Application, Docket 98-17-9225, February 18, 1999.

U.S. Department of Energy. Q2 Questions on NAC-LWT Cask Application, Docket 98-17-9225, June 4, 1999.

U.S. Department of Energy. Additional Questions for Telecon Discussion, June 16, 1999.

U.S. Department of Energy. Additional Questions for Telecon Discussion, June 28, 1999.

2. STRUCTURAL REVIEW

2.1 Review Procedures

The following areas applicable to the Structural Evaluation chapter of the SARP for the NAC-LWT were reviewed.

2.1.1 Description of Structural Design

2.1.1.1 Design Features
The NAC-LWT cask is a right circular cylindrical cask. Chapters 1 and 2 of the SARP adequately describe the major structural components of the packaging and contents as follows:

A cask body which is made of two XM-19 stainless-steel cylindrical shells welded at the cask closed end to a 304 stainless-steel circular plate forging and at the cask open end to a 304 stainless-steel ring forging (Drawings 315-40-03 Sheet 2 of the SARP)

A 304 stainless-steel closure lid that is bolted to the upper ring forging with twelve 1-in. SA-453 Grade 660 high alloy steel bolts (Drawing 315-40-04 and SARP Section 2.3.1.4)

A lead shield which is encased in the space between the inner and outer shells of the cask body and also in the bottom end forging (Drawing 315-40-03 Sheet 2)

A fill-port cover and a drain-port cover which are made of SA705, Grade 630 (Type 17-4PH) precipitation-hardened stainless steel and bolted to the cask upper ring forging with three 3/8-in.-diameter SA-193, Grade B6 (Type 410) stainless steel cap screws (Drawings 315-40-03 Sheet 4 and 315-40-08, and SARP Section 2.3.1.2)

A neutron shield tank and an expansion tank which are made of 304 stainless steel welded to the cask exterior cylindrical surface (Drawing 315-40-03 Sheets 2 and 5)

Two aluminum honeycomb impact limiters, each of which is fitted over one end of the cask and attached to the cask exterior cylindrical surface with four 1/2-in. 92384A094 McMaster Carr ball-lock pins (Drawings 315-40-01, -05 and -06)

Four 304 stainless-steel lifting lugs which are equally spaced (at 90-degree intervals) around the cask body and welded to the exterior surface of the cask upper ring forging (Drawing 315-40-03 Sheets 1 and 2)

Four 1.5 in.-deep, 1-in.-diameter threaded holes which are drilled into the outer surface of the cask closure lid for four corresponding lid-lifting bolts (Drawing 315-40-04)

Two 304 stainless-steel rotation lugs which are welded on opposite sides of the cask exterior cylindrical surface near the closed or bottom end of the cask (Drawing 315-40-03 Sheets 4 and 5)

A 90-degree shear ring segment which is welded to the cask exterior cylindrical surface near the top end of the cask (Drawing 315-40-03 Sheet 3).

In its tie-down position, the cask is supported at the shear ring by a saddle-and-strap device and at the rotation lugs by a pin-and-frame structure (Drawings 315-40-13 through 315-40-18). The shear ring helps restrain the cask in the cask axial direction while the strap contributes to the restraint in the lateral direction. The rotation lugs also support the cask when the cask is lifted into or out of the tie-down position.

The contents being reviewed herein for shipment using the NAC-LWT cask is the Lead Test Assembly (LTA) which consists of 16 thimble plugs and 8, 316 stainless-steel-cladded Tritium Producing Burnable Absorber Rods (TPBARs) attached to a stainless hold-down assembly plate (Figure 1.2-6 of the SARP). The LTA is supported by the following structural components in the cask cavity or containment vessel:

> An LTA container, which is a standard unirradiated 17 x 17 PWR fuel assembly skeleton, reinforced with 44 open-ended Zircaloy-4 thimble tubes and 24 guide thimble tubes for receiving the LTA assembly (Fig. 1.2-9 of the SARP)
>
> A PWR basket which is a 6061-T6 aluminum circular cylinder with a central hollow square cavity (Drawing 315-4-10)
>
> A 304 stainless-steel-encased lead shield plug (Drawings 315-40-093 and -095).

2.1.1.2 Codes and Standards
The drawings in Section 1.4 of the SARP clearly identify ASME B&PV Code Section III Subsection NB to be the code and standard used for the fabrication and examination of the cask containment components. However, the codes and standards for other components are not clearly specified.

Section 2.1.2 of the SARP describes the design criteria used for the structural evaluation. Consistent with the code and standard used for fabrication and examination, the structural design criteria for containment components are based on ASME B&PV Code Section III, Subsection NB and Regulatory Guides 7.6 and 7.8. Similar but less restrictive criteria are used for non-containment components. Tables 2.1.2.1 and 2.1.2.2 of the SARP tabulate the allowable stresses specified in the design criteria for containment structures and non-containment structures under normal conditions of transport (NCT) and hypothetical accident conditions (HAC) loadings.

Because of the tolerance of the non-containment components for damage, the staff considers the structural design criteria to be adequate.

2.1.2 Materials of Construction

The drawings in Section 1.4 of the SARP identify the materials of construction for the entire package except the LTA contents. These are described in Section 1.2.3.4.1 of the SARP. The list of major structural components in Section 2.1.1.1 of this SER identifies the materials of construction of each component.

2.1.2.1 Material Specifications and Properties
The drawings in Section 1.4 of the SARP also provide specifications of all materials of construction identified therein. Detailed properties of most of these materials are tabulated in Sections 2.3.1.1 and 2.3.1.2 of the SARP. The design and materials of the

LTA container are proprietary. The properties used for the proprietary materials are estimates.

2.1.2.2 Prevention of Chemical, Galvanic, or Other Reactions

Section 2.4.4 of the SARP certifies that all cask materials in direct contact do not have potential chemical and galvanic reactions.

2.1.2.3 Effects of Radiation on Materials

Section 4.5.4.1 of the SARP estimates the maximum amount of tritium in the cask cavity. In response to the staff's questions, the applicant believes that this small amount of tritium cannot adversely affect any of the materials of construction of the cask. The staff considers this belief to be justifiable, as long as the amount of tritium in the cask cavity does not exceed the SARP limit.

2.1.3 Fabrication, Assembly, and Examination

2.1.3.1 Fabrication and Assembly

The drawings in Section 1.4 of the SARP specify the weld designs and bolt torques for fabricating and assembling the packaging. The drawings also specify the codes and standards for welding containment components. All welds on the containment boundary are full penetration butt welds that must be fabricated in accordance with ASME B&PV Code Section III, Subsection NB, and qualified according to the requirements of Section IX. Section 1.4 explicitly states that the base metal and welds of the containment components are subject to the requirements of NB-2500 and NB-2400 as defined by the requirements of Section NB-4000. The drawings in Section 1.4, however, do not specify the codes and standards to be followed for welding non-containment components.

2.1.3.2 Examination

The drawings in Section 1.4 of the SARP also specify the method of weld examination and acceptance criteria for containment components. Consistent with the welding design and procedure, all containment welds will be examined per ASME B&PV Code Section III, Article NB-5000, and Section V, and with acceptance per Section III, Article NB-5300. However, as noted in Drawing 315-40-03 Sheet 2, the containment welds of four previously fabricated cask units (#2, 3, 4, and 5) do not fully comply with the general requirements and acceptance criteria for radiographic examination, as specified in Article NB-5000. The welds were accepted on the basis of the modified criteria of ASME Section III, Code Case N-493.

2.1.4 General Considerations for Structural Evaluations

2.1.4.1 Design Criteria

As stated in Section 2.1.1.2 of this SER, the design criteria used for structural evaluation of the containment components are consistent with the codes and standards used for fabrication and assembly. The criteria for the non-containment components are adequate although they may not be consistent with the codes and standards used for fabricating and assembling the components.

2.1.4.2 Evaluation by Analysis

The structural evaluation presented in the SARP is primarily by analysis. All the analyses are accomplished using one or more of three analytical tools: (1) approximate formulas from handbooks, (2) the commercial finite-element computer program ANSYS, and (3) NAC's proprietary computer program RBCUBED for evaluating force-deflection curves of honeycomb impact limiters and for analyzing free-drop dynamics. ANSYS is a widely used program for static and dynamic, linear and non-linear stress analysis. It is known to have an adequate QA program. The validity of the RBCUBED computer program is demonstrated in Sections 2.10.12.4 and 2.10.12.5 of the SARP using the test results of a quarter-scale model of the cask and impact limiters.

Simplified approximate methods and handbook formulas are used for the stress analysis of other components such as the neutron shield, expansion tank, lifting and rotation lugs, impact limiters, and closure bolts.

For the review, the staff relied on engineering judgement, simplified models, approximate formulas, and the SCANS computer program to confirm the structural evaluation presented in the SARP.

2.1.5 Structural Evaluation for Normal Conditions of Transport

2.1.5.1 Heat

In Section 2.6.1 of the SARP, the stresses caused by the heat condition specified in 10 CFR 71.7(c)(1) are analyzed and compared to the allowable stresses tabulated in SARP Table 2.1.2.1 for the containment components. The ANSYS computer program and model described in Sections 2.10.1 and 2.10.2 of the SARP are used for the analysis. The model includes the cask body (the cask inner and outer shells, the upper ring forging, the bottom plate forging, and the lead shield), the closure lid, and closure bolts. The cask body and lid are modeled using axisymmetric solid elements while the bolts are modeled with beam elements. The preload of the closure bolts is simulated using an appropriate initial strain. For the analysis, the cask is assumed to be loaded and ready for shipment in the horizontal position with an ambient temperature of 130°F. An internal pressure of 50 psig and a maximum (PWR fuel) decay heat of 2.5 kW are conservatively used for the analysis. The maximum normal operating pressure is estimated in Section 3.4.4 of the SARP to be 28.6 psia. Stresses due to the cask and contents weight are included but not the fabrication stresses, as the fabrication stresses have been shown in Section 2.6.11 of the SARP to be negligible. The combined stresses are compared to the corresponding allowable stresses tabulated in Table 2.1.2.1 of the SARP to show compliance with the design criteria for containment components. The most stressed section is in the closure lid. A margin of safety of 0.3 is shown for this section.

2.1.5.2 Cold

In Section 2.6.2 of the SARP, the same analysis procedure used for the heat condition is used to demonstrate compliance with the regulatory requirement of 10 CFR 71.71(c)(2). For this analysis, the cask is assumed to be loaded and ready for shipment in a horizontal position in an ambient temperature of -40°F. The minimum margin of safety is 1.5 and occurs in the closure lid.

2.1.5.3 Reduced External Pressure
Section 2.6.3 of the SARP explains that the reduced external pressure condition of 3.5 psia, as specified in 10 CFR 71.71(c)(3), need not be evaluated since the cask has been conservatively evaluated for an internal pressure of 50 psig.

2.1.5.4 Increased External Pressure
Using numerical results from simple calculations, Section 2.6.4 of the SARP justifies that the effect of the increased external pressure condition of 20 psia as specified in 10 CFR 71.71(c)(4) is insignificant.

2.1.5.5 Vibration
Section 2.6.5 of the SARP estimates the maximum alternating stress range that the vibration condition specified in 10 CFR 71.71(c)(5) can produce in the cask body and the rotation lugs. The alternating stresses do not exceed the fatigue endurance limits of the materials. Thus the design complies with the regulatory requirement.

2.1.5.6 Water Spray
Section 2.6.6 of the SARP states that a water spray as specified in 10 CFR 71.71(c)(6) has no adverse effect on the package because the cask is made of stainless steel.

2.1.5.7 Free Drop
In Section 2.6.7 of the SARP, the cask is evaluated for a 1-ft free drop in accordance with 10 CFR 71.71(c)(7). A quasi-static analysis method is used. The maximum impact force or deceleration g-force is first obtained using the proprietary computer program RBCUBED, then the g force is applied to the ANSYS finite element model and the resulting maximum stresses are compared to the design criteria to demonstrate compliance. The RBCUBED program determines the impact-limiter force-deflection curve and analyzes the impact process using two simplified assumptions: (1) the limiter honeycomb material has a constant crush strength independent of the impact velocity; (2) only the limiter material that lies vertically above the impact area contributes to the impact force; i.e., the material outside this volume is not deformed. The adequacy of these assumptions and the computer program has been demonstrated with results from static crush tests and free drop tests of quarter-scale models of the cask and impact limiters.

The ASME stress analysis results of three drop orientations, end, side and corner drops are presented in SARP Tables 2.6.7-1 through 2.6.7-31 for the cask body at two ambient temperatures, 130°F and -40°F. All stresses show an adequate margin of safety relative to the design criteria for containment components.

The NAC-LWT cask impact limiter has two significantly different materials or components. One of the components, called the secondary limiter in the SARP, is a relatively thin and soft (250-psi crush strength) layer of unidirectional aluminum honeycomb located at the bottom or the impact end of the impact limiter. The other limiter component or the primary limiter is made of a high-strength (3500-psi crush strength) multi-directional aluminum honeycomb that occupies the remainder of the limiter volume. The secondary limiter is designed to protect the cask during an NCT 1-ft end drop, while the primary limiter is for protection during an HAC 30-ft drop. The combined limiter has the capacity to absorb the total kinetic energy of a 30-ft drop following a 1-ft drop without bottoming out or showing excessive deformation. Thus the limiter design meets the 10 CFR 71.43(f) and 71.51(a)(1) requirement that the NCT does not cause a substantial reduction in the effectiveness of the packaging. As additional proof of compliance, Section 2.6.7.1 of the SARP analytically shows that the impact limiter attachment lugs, welds, and pins do not yield during normal handling and 1-ft free drops.

Tables 2.6.7-32 and 2.6.7-33 of the SARP summarize the maximum impact displacements (crush depths), and forces due to a 1-ft NCT drop and a subsequent 30-ft HAC drop. The maximum displacements are about 1 in. and 10in. for 1-ft and 30-ft flat side drops, respectively. These results indicate that in a side drop the lifting lugs, the neutron shield tank, and the expansion tank are protected; only the impact limiters touch the flat unyielding surface. Thus the stress analyses of the cask are justified to consider only the impact limiter reactions and the inertial effect of the material mass. Table 2.6.7-32 shows that the maximum impact forces due to the 1-ft end, corner, and side drops are about 16 g, 12 g, and 24 g, respectively. Table 2.6.7-33 shows the corresponding results for the 30-ft drops as 48 g, 60 g, and 59 g.

Section 2.6.7.7.3 of the SARP addresses the hydrodynamic load generated by the drops on the structural components of the neutron-shield and expansion tanks. Since the tanks are not filled for the LTA shipment, the evaluation of the hydrodynamic load was not reviewed.

2.1.5.8 Corner Drop
This package is not subject to the corner drop requirement of 10 CFR 71.71(c)(8).

2.1.5.9 Compression
This package is not subject to the compression requirement of 10 CFR 71.71(c)(9).

2.1.5.10 Penetration
Compliance with the penetration test requirement of 10 CFR 71.71 (c)(10) is demonstrated in Section 2.6.10 of the SARP using analyses. Sections 2.6.10.1 through 2.6.10.1.4 present the analyses for the exposed surface of all exterior cask components which include the impact limiter, the expansion tank, the neutron-shield tank, and the port covers. Simplified methods and formulas are used to show that the penetration test may produce localized permanent deformation but not a rupture of the surface of the impact limiter, the expansion tank, and the neutron-shield tank. The deformation should not degrade the normal functions of these cask components and their ability to withstand subsequent HAC.

2.1.5.11 Structural Requirements for Fissile Material Packages
The NAC-LWT cask with the LTA contents has no fissile material. Therefore, the package needs not meet the structural requirements of 10 CFR 71.55(d)(2-4).

2.1.6 Structural Evaluation for Hypothetical Accident Conditions

2.1.6.1 Free Drop

Section 2.7.1 of the SARP describes the analysis procedure used to evaluate compliance with the 30-ft HAC free-drop requirements of 10 CFR 71.73(c)(1). The procedure is essentially the same as for the NCT 1-ft drops. The analyses and results are discussed in Sections 2.7.1.1 through 2.7.1.3 of the SARP for six drop orientations: end (0 degrees), side (90 degrees) and oblique (15.74, 30, 45, and 60 degrees). For each orientation, top and bottom drops in several thermal environments are considered. In general, the thermal environments include the following: 130°F ambient temperature with maximum contents decay heat, -40°F ambient temperature without decay heat, and -40°F ambient temperature with decay heat.

The critical stresses in the cask body due to the free-drop cases are tabulated and compared to the design criteria in Tables 2.7.1-1 through 2.7.1-19 of the SARP for end and side drops and in Sections 2.7.1-24 through 2.7.1-59 for oblique drops. All tabulated results show adequate margins of safety, i.e., the critical stresses are substantially below the corresponding allowables of the design criteria. Possible buckling of the cask inner shell or containment vessel under the compressive impact load is investigated in Section 2.10.6 of the SARP using the analysis procedure of ASME Section III Code Case N-284. The analysis concludes that the cask inner shell does not buckle under the maximum impact load of a 30-ft free drop.

Similar stress and buckling analyses of the LTA contents are presented in Section 2.9.3 of the SARP. The analyses show that the LTA container does not buckle under a 60-g impact load. However, this conclusion is based on the following assumptions: (1) the thimble-tubes of the LTA container supporting the LTA assembly are perfectly straight and have no initial geometric imperfections, and (2) the thimble tubes are totally fixed in displacement and rotation by the spacer grids and bottom nozzle of the LTA container. The staff considers these assumptions to be unrealistic and nonconservative because the applicant cannot provide detailed information about the container design and valid evidence supporting the assumptions. Using the realistic assumptions commonly accepted in buckling evaluations (geometric imperfections exist and the thimble tubes are free to rotate about the spacer grid and the bottom nozzle), the staff's confirmatory analyses predicted that the thimble tubes buckle under the 60-g impact load or under a load which is only a quarter of that presented in the SARP. Thus, the staff is not convinced that the LTA contents do not buckle and result in damage to the TPBARs of the contents during an HAC 30-ft free drop.

Section 2.7.1.5 of the SARP analyzes the structural integrity of the closure bolts during the HAC drops. SARP Tables 2.7.1-60 and 2.7.1-61 present the bolt stresses due to the drops in the hot and cold conditions. The tabulated results show a positive margin of safety of 0.3 to 0.45 for the bolt stresses relative to the yield strength of the bolts.

Section 2.7.1.4 of the SARP addresses the effect of lead slump due to the HAC drops on the external radiation dose rate. The potential lead slump, including the thermal effect, is obtained using ANSYS finite element analysis for several bounding cases of combined impact and thermal effects. Section 2.10.5 of the SARP summarizes the results. The maximum lead gap due to the combined effect of lead slump and thermal expansion is 0.3216 in. However, a more conservative value of 1.63 in. is used to demonstrate that the lead slump effect on the dose rate is acceptable. The value of 1.63 in. is obtained from a simplified and conservative thermal expansion analysis.

The SARP has addressed the structural integrity of the neutron shield tank, the expansion tank and the impact limiter for NCT but not for HAC. Nevertheless, this information is not essential since the neutron shield is not filled for the LTA shipment and the impact limiter is not needed after the 30-ft drop to protect the containment seals from the HAC fire.

The SARP has not addressed the possible amplification of the impact effect due to vibrations of the cask components; i.e., the SARP always uses a dynamic amplification factor of 1 in its quasi-static free-drop impact stress analysis. However, the staff does not consider this oversight to be a serious problem because the amplification is estimated to be small and the SARP analyses and results have adequate margin to cover the amplification effect.

2.1.6.2 Crush

The NAC-LWT cask is not subject to the test requirement of 10 CFR 71.73 (c)(2).

2.1.6.3 Puncture

Similar to the penetration evaluation, the puncture evaluation presented in the SARP is by analysis using simplified methods and formulas. Sections 2.7.2.1 through 2.7.2.6 of the SARP discuss the methods used and the results obtained for the cask outer shell, the cask bottom plate, the closure lid, the port covers, and the outer shell of the neutron shield tank. Using Nelms empirical equation for predicting puncture thickness, only the outer shell of the neutron shield is shown to have the possibility of puncture. Using a plate bending analysis, the outer shell and the bottom plate of the cask body are shown to have local permanent bending deformation due to the puncture load. Using the same analysis, the deformations of the closure lid and the port covers are shown to remain elastic and small under the puncture loading. Thus the SARP demonstrates that the closure lid and port covers can maintain containment after the puncture test as required by 10 CFR 71.73(c)(3). The SARP also shows that puncture tests will not stress the cask more than the 30-ft free drops because puncture drops produce lower maximum impact loads than the 30-ft drop.

The SARP does not examine the possibility that a puncture test following a 30-ft drop can separate the damaged impact limiters from the cask body. However, since the cask does not depend on the limiters for thermal protection during the subsequent HAC fire, this possibility needs no investigation.

2.1.6.4 Thermal

Sections 2.7.3.2 and 2.7.3.4 of the SARP demonstrate that the maximum stresses in the cask inner shell or containment vessel meet the containment vessel design criteria for the fire accident. In addition, Section 2.7.3.3 shows that the closure bolts remain elastic during the fire. Thus, the containment ability of the cask is maintained during the fire and the cask containment system meets the requirement of 10 CFR 71.73(c)(4).

2.1.6.5 Immersion–Fissile Material

The NAC-LWT package has no fissile material. Therefore, the package is not subject to the requirement of 10 CFR 71.73(c)(5).

2.1.6.6 Immersion–All Packages

Section 2.7.5 of the SARP states that the cask meets the immersion requirement of 10 CFR 71.73(c)(6) if the cask containment system meets the special pressure condition of 10 CFR 71.61 for spent fuel packages. The NAC-LWT cask with the LTA contents is exempt from the special pressure conditions for spent fuel packages, and the small external pressure of 21.7 psig generated by the immersion test is too small to have a significant effect on the containment seals of the NAC-LWT cask. Therefore, the staff considers that the NAC-LWT with the LTA contents meets the immersion requirement.

2.1.7 Lifting and Tie-down Standards for All Packages

2.1.7.1 Lifting Devices

Section 2.5.1 of the SARP identifies the lifting devices and evaluates their structural capacity according to 10 CFR 71.45(a). The evaluation shows that the cask lifting lug and welds can carry three times the cask weight without yielding. The evaluation further shows that the cask upper ring forging to which the lifting lugs are attached is stronger than the lifting lugs and their welds. Thus the lifting devices meet the regulatory requirement that states that the failure of the lifting devices does not prevent the cask from continuing to meet other regulatory requirements.

2.1.7.2 Tie-down Devices

Section 2.5.2 of the SARP identifies the tie-down devices and evaluates their structural capacity according to the requirements of 10 CFR 71.45(b). The evaluation shows that the tie-down devices, namely, the rotation lugs, shear ring, and their welds meet, with positive margin of safety, the design criteria for non-containment structural components. In addition, these devices and their connections to the cask outer shell are shown to be weaker than the cask shell. Thus the requirement of 10 CFR 71.45(b)(3) is satisfied.

2.1.8 Special Pressure Conditions

2.1.8.1 Special Requirement for Irradiated Nuclear Fuel

The NAC-LWT cask with the LTA contents is not subject to the special pressure condition requirement of 10 CFR 71.61 for irradiated nuclear fuel packages.

2.1.8.2 Analysis of Pressure Test

The stress analyses in Sections 2.6.1 and 2.6.2 of the SARP demonstrate that in both cold and hot environments the containment system of the cask can withstand an internal pressure of 50 psig without yielding. Since 50 psi is greater than 1.5 times the maximum operating pressure which is determined in Section 3.4.4 to be 28.6 psia, the cask can be safely pressure tested in accordance with 10 CFR 71.85(b). However, Chapter 8 of the SARP states that a pressure of 209 psig is used for the hydrostatic test. If a stress analysis were performed for this pressure, the analysis would show the yielding of some of the cask materials.

2.1.9 Appendix

The review also included the following sections and their subsections in the appendices of Chapter 2 of the SARP, which contain information supporting the structural evaluation.

Section 2.10.1 of the SARP describes the two computer programs, ANSYS and RBCUBED, used for the structural analysis. Section 2.10.2 describes the ANSYS finite-element model used for analyzing the stresses and displacements of the cask body under various loadings. Section 2.10.3 presents the temperature contours in the cask body as obtained in Chapter 3 of the SARP. The section also outlines the procedure used to determine the critical ASME stress intensities from the finite-element structural analysis results for the cask body. Section 2.10.4 presents the method used for a slap-down analysis. Section 2.10.5 summarizes the lead-slump estimates for 30-ft end drops. The estimates include both thermal and impact effects on the lead slump. Section 2.10.6 describes the buckling design criteria and evaluation of the cask inner shell (containment boundary). The ASME Section III Code Case N-284 is used for the buckling evaluation. Section 2.10.7 tabulates for selected cask cross-sections the finite-element stress results due to various NCT and HAC loadings. Section 2.10.8 summarizes the 30-ft drop test program for a quarter-scale model of the NAC-LWT cask. The summary covers the test objective, test-model design and construction, test instrumentation and procedure, test results and interpretation. Section 2.10.9 describes the method for stress analysis of closure bolts. The section also tabulates the analysis results for the HAC loadings. Section 2.10.10 presents finite element stress results for 30-ft drops. Section 2.10.12 describes and compares the impact-limiter force-deflection curves that are experimentally measured in static crush tests to those that are analytically determined using the RBCUBED computer program.

2.2 Evaluation Findings

2.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the structural design has been adequately described and evaluated and that the package design meets the structural requirements of 10 CFR Part 71. However, the findings come with two cautions.

First, as discussed in Section 2.1.6.1 of this SER, the SARP evaluation has not adequately demonstrated the integrity of the LTA contents after a 30-ft drop because the buckling evaluation of the contents is based on uncertain design information and nonconservative assumptions. Because of this concern, Chapter 4 of the SARP addresses the containment evaluation with the assumption of damaged TPBARs, as discussed in Section 4 of this SER. Second, as discussed in Section 2.1.2.1 of the SER, the structural evaluation assumes that the tritium does not degrade the structural properties of the package materials. The staff accepts this assumption provided the amount of tritium in the cask cavity never exceeds 27.9 Ci, the limit established in Chapters 4 and 7 of the SARP. Both foregoing cautions highlight the implication and importance of the tritium monitoring and annual weld inspection programs reviewed in Chapters 7 and 8.

2.2.2 Conditions of Approval

Conditions of approval applicable to the Structural review of the NAC-LWT cask with the LTA contents include the requirement and acceptance criteria for tritium monitoring and the annual weld inspection, as discussed in Section 7 and 8 of this SER.

2.3 References

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

3. THERMAL REVIEW

3.1 Review Procedures

The following areas applicable to the Thermal Evaluation chapter of the SARP for the NAC-LWT were reviewed.

3.1.1 Description of Thermal Design

3.1.1.1 Design Features
The NAC-LWT package is passively cooled. The LTA and its container are loaded in a PWR basket within the helium-filled cavity of the NAC-LWT cask. The structural and mechanical means for the transfer of heat from the LTA to the package surface are illustrated in Figure 3.4-7 of the SARP. The package is transported within an ISO container or personnel barrier. The cask regions include the lead shielding, steel structure, and an unfilled neutron shield. The gaps between the cask regions are assumed to be air.

The text and sketches describing thermal design features in the Thermal Evaluation chapter of the SARP are consistent with the General Information chapter, the engineering drawings, and the models used in the thermal evaluation.

3.1.1.2 Decay Heat of Contents
The decay heat of the LTA contents containing eight TPBARs, described in Section 3.1 of the SARP, is 14 watts. The thermal analysis for the NAC-LWT package with the LTA contents assumes a decay heat load of 55 watts. The design basis heat load for the NAC-LWT is 2.5 kW, based on the decay heat for PWR fuel contents.

3.1.1.3 Codes and Standards
No codes and standards specific to the thermal design of the package were identified or used. The structural materials conform to the ASME Code, Volume II, Part A as described in Sections 1.4 and 2.3 of the SARP. The allowable stress limits of the structural materials are dependent on the temperature of the material as presented in the ASME Code, Volume II, Part D.

3.1.1.4 Summary Tables of Temperatures
Maximum component temperatures under normal conditions of transport with the LTA contents are presented in SARP Table 3.4-9. Maximum, minimum, and allowable temperatures of the O-rings and lead shielding under normal conditions of transport are presented in SARP Table 3.4-5. These maximum temperatures are based on the PWR contents. The staff concurs that these temperatures are bounding for the LTA contents and that temperatures under normal conditions of transport are within the allowable temperature range.

The maximum component temperatures of the NAC-LWT with LTA contents under the hypothetical accident conditions are presented in SARP Table 3.5-4. With the exception of the temperature history of the TPBAR cladding shown in SARP Figure 3.5-11, the times following the initiation of the 30-minute fire at which these temperatures occur are not presented. The maximum O-ring, port cover, and valve temperatures under hypothetical accident conditions are listed in SARP Table 3.5-1. Again, these temperatures are based on the PWR contents, and the staff agrees that these temperatures are bounding and acceptable for the LTA contents Following the initiation of the 30-minute fire, the time at which the maximum temperatures occur for some components of the NAC-LWT with PWR content are presented in the temperature histories of Figure 3.5-5 of the SARP.

While not presented in the SARP, the maximum post-fire steady-state temperatures should be the same as the temperatures for the normal conditions of transport listed in Table 3.4-9 of the SARP.

3.1.1.5 Summary Table of Maximum Pressures

Because the maximum normal operating pressure and the maximum pressure in the containment system under hypothetical accident conditions for the LTA contents are less than those for the PWR assembly contents, the SARP does not determine these pressures for the LTA contents. The staff concurs that this maximum operating pressure (28.6 psia) and maximum pressure under hypothetical accident conditions (125.8 psia) are bounding and acceptable.

3.1.2 Material Properties, Temperature Limits, and Component Specifications

3.1.2.1 Material Properties

The heat transfer properties of the applicable materials are presented in Tables 3.2-1 through 3.2-7 in the SARP. The structural properties, including the modulus of elasticity, Poisson's ratio, and coefficient of thermal expansion, are presented in Tables 2.3.1-1 through 2.3.1-3 and Tables 2.3.1-5 through 2.3.1-9 in the SARP.

3.1.2.2 Temperature Limits

The design stress intensities of the structural materials as a function of temperature are presented in Tables 2.3.1-1 through 2.3.1-3 in the SARP. The strengths (ultimate and yield) and other mechanical properties of the 6061-T6-aluminum basket material and the bolting material are presented in Tables 2.3.1-5, 2.3.1-6, and 2.3.1-7 in the SARP.

3.1.2.3 Component Specifications

The temperature limits for the lead shielding as well as the seals are presented in Section 3.3 of the SARP. These limits are also presented in Table 3.4-5 of the SARP.

3.1.3 General Considerations for Thermal Evaluations

3.1.3.1 Evaluation by Test

The thermal evaluations of the NAC-LWT package are not based on tests.

3.1.3.2 Evaluation by Analysis

The thermal analyses of the NAC-LWT package under both normal conditions of transport and under hypothetical accident conditions are performed using the general purpose ANSYS computer code.

3.1.4 Thermal Evaluation under Normal Conditions of Transport

3.1.4.1 Initial Conditions

The boundary conditions for normal conditions of transport presented in Section 3.4.1.6 of the SARP satisfy the regulatory requirements of 10 CFR 71.71. The load combinations for the hot environment with 100°F ambient temperature and the cold environment of -40°F are consistent with those presented in Table 1 of NRC Regulatory Guide 7.8. The package thermal performance is determined for the shipment in the ISO container. The neutron shield tank is specified to contain only air.

3.1.4.2 Effects of Tests

The normal conditions of transport do not degrade the heat-transfer capability or insulating capability of the package. The change in the package component temperatures under normal conditions of transport from the as-fabricated and assembled temperature do not significantly affect the structural or functional (containment, shielding, and criticality control) performance of the package.

3.1.4.3 Maximum and Minimum Temperatures

The maximum temperatures of various components of the NAC-LWT package with the LTA contents are determined for a total decay heat of 55 W. The maximum component temperatures, presented in SARP Table 3.4-9, are substantially less than those for the PWR spent fuel assembly operating with an ambient temperature of 130°F still air with full insolation, as presented in SARP Table 3.4-2. The minimum temperature of the package without contents is -40°F. The maximum and minimum temperatures of the O-rings and the lead gamma shield are within the safe operating range presented in SARP Table 3.4-5.

3.1.4.4 Maximum Normal Operating Pressure

The maximum normal operating pressure for the PWR spent fuel assembly assuming 3% failed fuel rods is 28.6 psia. The maximum pressure in the containment vessel with the LTA contents of eight intact TPBARs (including the contribution of the residual water vapor and permeated tritium) is bounded by this value.

3.1.4.5 Maximum Thermal Stresses

The thermal stresses and temperature gradients for the NAC-LWT with the 55-W LTA contents are bounded by those with PWR spent fuel assembly and are acceptable.

3.1.5 Thermal Evaluation under Hypothetical Accident Conditions

3.1.5.1 Initial Conditions

The initial temperature fields in the package prior to the hypothetical accident conditions are based on the normal conditions of transport presented in Section 3.4.1.6 of the SARP with 100°F ambient temperature and insolation. The package initial thermal performance is determined for the shipment in the ISO container. The neutron shield tank is specified to contain only air.

The impact limiters are removed and the package is removed from the ISO container at the initiation of the 30-minute, 1475°F fire. The package and fire emissivities satisfy the requirements of 10 CFR 71.73(c)(4). The convection coefficient of 0.02446

BTU/hr-in^2-°F, presented in Section 3.5.2 of the SARP, is appropriate for the forced convection in a fire with a velocity between 10 to 20 m/s.

3.1.5.2 Effects of Thermal Test

The post-fire conditions include the temperature conditions of 10 CFR 71.73(b) and post-fire cooling limitations of 10 CFR 71.73(c)(4) with appropriate insolation. To the extent that the fire evaluation is performed by analysis, combustion issues are not addressed. However, the only potential combustible portion of the package is the O-ring seal, which is contained within the package.

Temperatures of the package components important to containment and shielding do not exceed their allowable limits. Criticality is not an issue, as the LTA contents do not contain fissionable material.

3.1.5.3 Maximum Temperatures and Pressures

The maximum component temperatures under hypothetical accident conditions are determined using a two-dimensional analysis for a total decay heat of 55 watts. The maximum component temperatures, presented in SARP Table 3.5-4, are substantially less than those for PWR spent fuel assembly, presented in SARP Table 3.5-1. The maximum temperatures of the O-rings, lead gamma shield, and cladding are within the safe operating range presented in Table 3.5-1. The maximum temperature of the TPBAR cladding is 260°F, but exceeds 200°F for only a 40-hour duration, as shown in Figure 3.5-11 of the SARP.

The maximum pressure under hypothetical accident conditions is bounded by 125.8 psia, which is determined for PWR spent fuel assembly contents (assuming 100% failed fuel rods).

3.1.5.4 Maximum Thermal Stresses

The maximum thermal stresses in the NAC-LWT package caused by a hypothetical fire accident condition occur following cessation of the fire. The maximum temperature gradient occurs in the outer shell of the cask following cessation of the fire. Like the case for the normal conditions of transport, the thermal stresses and temperature gradients with the LTA contents are bounded by those for PWR spent fuel assembly and are acceptable.

3.1.6 Thermal Evaluation of Maximum Accessible Surface Temperature

While not covered in the SARP, the maximum accessible surface temperature of the NAC-LWT package (not the ISO container or personnel barrier) in 100°F still air in the shade with the LTA contents is less than 122°F. This determination is based on a heat load of 55 W, which significantly exceeds the actual heat load in an LTA with eight TPBARs. This temperature satisfies the conditions of 10 CFR 71.43(g) for both non-exclusive and exclusive use shipments.

3.1.7 Appendix

The SARP does not include an appendix to Chapter 3.

3.2 Evaluation Findings

3.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the thermal design has been adequately described and evaluated, and that the thermal performance of the NAC-LWT package meets the thermal requirements of 10 CFR Part 71.

3.2.2 Conditions of Approval

The conditions of approval for the NAC-LWT for the shipment of LTA contents include a decay heat limit of 55 watts and a requirement that the TPBARs be undamaged.

3.3 References

Nuclear Regulatory Commission, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," Regulatory Guide 7.8, Rev. 1, March 1989.

4. CONTAINMENT REVIEW

4.1 Review Procedures

The following areas applicable to the Containment chapter of the SARP for the NAC-LWT were reviewed.

4.1.1 Containment Design Features

4.1.1.1 Containment Boundary
The boundary of the containment system is described in Sections 1.2 and 4.1 of the SARP. The containment boundary for the NAC-LWT cask consists of the inner shell, the bottom end plate, the lid, the upper ring forging, the lid inner metallic O-ring, the valve port covers, and the inner O-rings on the valve port covers. The inner shell is Type XM-19 stainless steel, and the bottom end plate, the lid and the upper ring forging are Type 304 stainless steel. The valve port covers are made of SA-705, Grade 630, condition H1150 precipitation-hardened stainless steel. The O-rings on the valve port covers are made of polytetrafluoroethylene (PTFE). The components of the containment system are shown in the following drawings in the SARP: 351-40-01 (Rev. 2), 315-40-02 (Rev. 6), 315-40-03 (Sheets 1-6, Rev. 9), 315-40-04 (Rev. 6), and 315-40-08 (Sheets 1-3, Rev. 7).

4.1.1.2 Containment Boundary Penetrations
The containment boundary penetrations are adequately described in Section 4.1.1 of the SARP. The penetrations into the NAC-LWT cask containment vessel cavity are the cask lid and the fill and drain ports. No device on the NAC-LWT cask containment vessel allows continuous venting. The only valves on the package are the quick-connect valves for the fill and drain ports, which are not part of the containment boundary.

4.1.1.3 Seals and Welds
The seals and welds on the containment boundary are adequately described in Section 4.1.2 of the SARP. The inner O-rings of the cask lid and valve port covers are the seals of the containment boundary. The cask lid seal is a metallic O-ring made by United Metallic O-rings. These metallic O-rings can withstand temperatures from -425°C (-733°F) to 1800°C (3272°F) and pressures from vacuum to 100,000 psi. The metallic O-ring lid seals are appropriate for use in the NAC-LWT cask with the LTA contents, and the seal grooves are properly sized.

The seals on the port covers are polytetrafluoroethylene (PTFE) O-rings (Shamban #S11214-147) and can withstand temperatures of -55°C (-67°F) to 135°C (275°F) and operating pressures from 0 to 3000 psi continuous. All containment seals and penetrations, including drain and vent ports, can be leak tested. The fill and drain ports utilize quick-connect valves; the leak testing procedures ensure that these valves do not preclude leakage testing of their corresponding containment seals. Although PTFE O-ring seals are not typically used for containment of tritium, because of the low concentration of releasable tritium in the containment vessel and the relatively short time that the tritium is in the cask, the PTFE O-rings are adequate. The PTFE seals receive a radiation dose as a result of tritium permeating through the seals; however, the radiation damage (addressed below) that occurs under normal and accident conditions is less than the threshold given for safe use of the seals. No significant galvanic, chemical, or other reactions are expected to occur between the seal and the packaging or its contents. The seal grooves for the non-metallic O-rings are properly sized.

All containment vessel welds are full penetration bevel or groove welds.

The review confirmed that the maximum and minimum temperatures of seals under normal conditions of transport and hypothetical accident conditions are within the manufacturers recommended operating ranges.

4.1.1.4 Containment Closure

The closure of the containment system is adequately described in Section 4.1.3 of the SARP. The containment vessel lid is machined to recess into the upper ring forging when it is installed on the cask. The closure lid attaches to the cask using 12 bolts with a 1-inch diameter (1-8 UNC bolts), each tightened to 260 foot-pounds of torque. Coordination with the structural review verified that the specified bolt torque provides proper compression for containment seals. The containment lid boundary seal is achieved by a metallic O-ring captured in a groove machined on the underside of the lid. A second O-ring is provided to allow seal testing. The lid O-rings mate against a machined sealing surface of the cask upper forging. A plastic weather seal is also provided on the closure lid.

The port covers are retained by three 3/8 – UNC bolts, each tightened to 100 inch-pounds of torque. The port cover bolt material is SA-193, Grade B6 high alloy steel.

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

4.1.1.5 Codes and Standards

Based on the LTA contents, the NAC-LWT is a Type B, Category II package. However, because the package was designed to transport spent nuclear fuel, ASME Code criteria applicable to a Category I package were applied to the design of the containment system. As discussed below and in Sections 7 and 8 of this SER, ANSI N14.5 criteria are used in the containment evaluation and leakage testing for the NAC-LWT.

4.1.2 General Considerations for Containment Evaluation

4.1.2.1 Type B Packages

Because the NAC-LWT with the LTA contents is a Type B package, the containment criteria in the SARP are based on the allowable release rates specified in 10 CFR 71.51(a), converted to leakage rates in accordance with ANSI N14.5. The effective A_2 is determined according to 10 CFR 71.51(b).

4.1.2.2 Combustible Gas Generation

As discussed below, the only source of flammable gas in the NAC-LWT with the LTA contents is tritium that is released from the TPBARs under normal conditions of transport and hypothetical accident conditions. The molar fraction of tritium in the free volume of the containment vessel is less than 1%. Furthermore, prior to shipment the package is backfilled with helium and essentially no oxygen is present for combustion. Consequently, no combustible gas generation issues should be relevant for the package.

4.1.3 Containment under Normal Conditions of Transport

Containment under normal conditions of transport is addressed in Section 4.2 of the SARP.

4.1.3.1 Releasable Source Term under Normal Conditions of Transport
The bounding inventory of tritium in the package and the bounding estimation of the releasable source term are presented in Section 4.5.4.1 of the SARP. The review confirmed that the radionuclides and physical form of the contents under normal conditions of transport evaluated in the Containment chapter are consistent with those presented in the General Information chapter of the SARP. The review verified that the maximum normal operating pressure and maximum temperature under normal conditions of transport are consistent with those determined in the Thermal Evaluation chapter.

The releasable source term consists of tritium that can escape intact TPBARs in the LTA and crud that can spall off the surface of the TPBARs. The releasable source term is based on 24 TPBARs even though the contents will include only eight rods.

The total inventory of tritium in 24 TPBARs is bounded by 28.8 g, or approximately 279,000 Ci. Most tritium is contained within the TPBARs due to the getter and other TPBAR design features. Tritium that escapes the rods and accumulates in the containment vessel receptacle is estimated to be bounded by 6.57 Ci. This corresponds to a TPBAR release fraction (the fraction of tritium that escapes the TPBARs to form the releasable source term) of 2.4×10^{-5}. For additional conservatism, however, the TPBAR release fraction of 1×10^{-4} is applied, which results in 27.9 Ci of tritium as the releasable source term under normal conditions of transport.

The surface activity density of the crud on the TPBARs is 138.5×10^{-6} Ci/cm^2 (time-cooled corrected from a discharge value of 140×10^{-6} Ci/cm^2). For a TPBAR surface area of 1190 cm^2 per rod and 24 TPBARs, the total bar surface area is 28,560 cm^2. The total crud activity on the TPBARs is 3.96 Ci. Under normal conditions of transport, the SARP estimates that 15% of the crud spalls off the TPBARs and is aerosolized in the containment vessel fill gas to contribute to the releasable source term. Therefore, the crud contributes about 0.593 Ci to the releasable source term, as discussed in Section 4.5.4.2 of the SARP.

Because the releasable source term within the NAC-LWT containment vessel with the LTA contents includes a mixture of radionuclides, an effective A_2 is determined according to the provisions of §71.51(b). The total activity of the releasable source term under normal conditions of transport is 27.9 Ci + 0.593 Ci = 28.49 Ci. The effective A_2 of the releasable source term under normal conditions of transport is 353 Ci, as discussed in Section 4.5.4.3 of the SARP.

4.1.3.2 Maximum Allowable Release Rate under Normal Conditions of Transport
The NAC-LWT cask with the LTA contents must satisfy the quantified release rates of §71.51(a)(1). There can be no release of radioactivity from the containment vessel greater

than 10^{-6} A_2 per hour under normal transport conditions. Using the effective A_2 for the releasable source term of 353 Ci, the maximum allowable release rate is 9.8×10^{-8} Ci/s, as discussed in Section 4.5.4.4 of the SARP.

4.1.3.3 Maximum Allowable Leakage Rate under Normal Conditions of Transport
For a maximum releasable inventory of 28.49 Ci under normal conditions of transport and a containment vessel void volume of approximately 147,000 cm^3, the activity density of the releasable source term is 1.94×10^{-4} Ci/cm^3. Dividing the maximum allowable release rate of 9.8×10^{-8} Ci/s by the releasable source term activity density of 1.94×10^{-4} Ci/cm^3, the maximum allowable volumetric leakage under normal conditions of transport is 5.05×10^{-4} cm^3/s, as discussed in Section 4.5.4.4 of the SARP.

4.1.3.4 Reference Air Leakage Rate Corresponding to Normal Conditions of Transport
The maximum allowable reference air leakage rate corresponding to normal conditions of transport is addressed in Section 4.5.4.5 of the SARP. Using the maximum allowable volumetric leakage rate of 5.05×10^{-4} cm^3/s under normal conditions of transport (347.4 K, 1.99 atm) along with the constitutive equations in ANSI N14.5, results in the reference air leakage rate of 3.85×10^{-4} ref cm^3/s. The corresponding helium leakage rate under reference leakage test conditions is 4.37×10^{-4} cm^3/s (He). Since the NAC-LWT cask is designed to transport spent nuclear fuel with a maximum allowable helium leakage rate criterion of 5.5×10^{-7} cm^3/s (He) under reference leakage test conditions, the current allowable helium leakage rate for the NAC-LWT cask is bounding for use of the NAC-LWT cask to transport the LTA contents.

4.1.3.5 Permeation of Tritium Through PTFE O-Rings
Determining the permeation rate of tritium through the PTFE O-rings is important for two reasons: (1) the permeation of tritium through the O-rings contributes to the tritium release rate, and (2) tritium that decays during permeation contributes to the radiation dose to the O-rings.

The permeation rate of tritium through the PTFE seals under normal conditions of transport, calculated using the bounding releasable source term inventory of tritium of 27.9 Ci, is 1.0×10^{-9} Ci/s. The tritium permeation rate of 1.0×10^{-9} Ci/s is nearly two orders-of-magnitude less than the allowable activity release rate.

Based on the solubility of tritium in the PTFE O-rings, the bounding dose rate due to tritium decay is 5.2×10^{-4} rad/s. With this dose rate, a dose of 10^4 rads for the PTFE O-rings is reached in approximately 303 days. Because each TPBAR shipment is expected to be completed in a time period much less than 303 days and given the many levels of conservatism used in obtaining the 303 day result, the review staff concluded that, based on the releasable source terms presented in the SARP for intact TPBARs, the function of the PTFE O-rings is not compromised during a shipment.

4.1.3.6 Demonstration of Compliance with Containment Design Criterion
The review confirmed that prior to the test, the leakage rate of the untested package (when converted to reference conditions) is demonstrated to be less than or equal to the reference air leakage rate criterion. The review verified that the leakage rate of a package

subjected to the tests of §71.71 does not exceed the maximum allowable leakage rate under normal conditions of transport.

4.1.4 Containment under Hypothetical Accident Conditions

4.1.4.1 Releasable Source Term under Hypothetical Accident Conditions
The radionuclides and physical form of the contents under hypothetical accident conditions evaluated in the Containment chapter are consistent with those presented in the General Information chapter of the SARP. The maximum pressure and maximum temperature under hypothetical accident conditions are consistent with those determined in the Thermal Evaluation chapter.

Under hypothetical accident conditions, the TPBARs are assumed to be damaged. Therefore, the releasable source term is the combination of the tritium that can escape the damaged TPBARs and the crud that can spall-off of the surface of the TPBARs. An analysis is performed to ensure that there would be no release of radiation from the transportation package in excess of regulatory limits even if the TPBARs are damaged. The bounding releasable source term under hypothetical accident conditions is presented in Section 4.5.4.1 of the SARP.

According to the SARP, under the hypothetical accident condition fire, the TPBARs would be subject to temperatures above 93°C (200°F) for a time not exceeding 100 hours. Because the release rate from each damaged TPBAR at a temperature between 93°C (200°F) and 343°C (650°F) is bounded by 100 Ci for the 100 hour time period, 24 TPBARs could release 2400 Ci of tritium. If this accident sequence occurs after 1 year of normal conditions of transport, the total quantity of tritium available for release under hypothetical accident conditions is 2427.9 Ci. However, for conservatism, a value of 2794 Ci is used for the total releasable tritium inventory under hypothetical accident conditions except for the analysis of radiation dose to the O-rings, as discussed in Section 4.1.4.5 of the SER below.

The activity of crud that contributes to the releasable source term under hypothetical accident conditions is discussed in Section 4.5.4.2 of the SARP. With a spallation fraction of 1.0, the activity in the releasable source term due to crud under hypothetical accident conditions is bounded by 3.96 Ci.

Since the releasable source term within the NAC-LWT containment vessel with the LTA contents includes a mixture of radionuclides, an effective A_2 is determined according to the provisions of §71.51(b). The effective A_2 of the releasable source term under hypothetical accident conditions is 947 Ci, as presented in Section 4.5.4.3 of the SARP.

4.1.4.2 Maximum Allowable Release Rate under Hypothetical Accident Conditions
The NAC-LWT cask with the LTA contents must satisfy the quantified release rates of §71.51(a)(2). There can be no release of radioactivity from the containment vessel greater than A_2 in a week under hypothetical accident conditions.

The maximum allowable release rate under hypothetical accident conditions is addressed in Section 4.5.4.4 of the SARP. Using the effective A_2 for the releasable source term of 947 Ci for the damaged TPBARs, the maximum allowable release rate is 1.56×10^{-3} Ci/s.

4.1.4.3 Maximum Allowable Leakage Rate under Hypothetical Accident Conditions
For damaged TPBARs, the releasable activity under hypothetical accident conditions is 2798 Ci. The activity density of the releasable source term is 1.9×10^{-2} Ci/cm^3. Dividing the maximum allowable release rate of 1.56×10^{-3} Ci/s by the releasable source term activity density, the maximum allowable volumetric leakage under hypothetical accident conditions is 8.22×10^{-2} cm^3/s.

4.1.4.4 Reference Air Leakage Rate Corresponding to Hypothetical Accident Conditions
The maximum allowable reference air leakage rate corresponding to hypothetical accident conditions is addressed in Section 4.5.4.5 of the SARP. If the TPBARs are damaged, the maximum allowable volumetric leakage rate is 8.22×10^{-2} cm^3/s. Using the hypothetical accident conditions (401.9K, 11.4 atm) along with the constitutive equations in ANSI N14.5, the reference air leakage rate is 9.71×10^{-3} ref cm^3/s. The corresponding helium leakage rate under reference leakage test conditions is 1.05×10^{-2} cm^3/s (He). Since the NAC-LWT cask is certified to transport spent nuclear fuel with a maximum allowable helium leakage rate criterion under reference leakage test conditions of 5.5×10^{-7} cm^3/s (He), the current allowable reference air leakage rate for the NAC-LWT cask is bounding for use of the NAC-LWT cask to transport the LTA contents.

4.1.4.5 Permeation of Tritium Through the PTFE O-Rings
The permeation rate of tritium under hypothetical accident conditions for eight damaged TPBARs, which is calculated using the bounding releasable source term inventory of tritium of 931.3 Ci, is approximately 1.4×10^{-7} Ci/s. This release rate of tritium (T_2) due to permeation through the PTFE O-rings is about four orders-of-magnitude less than the maximum allowable release rate of 1.56×10^{-3} Ci/s.

Calculations presented in the SARP regarding radiation damage to the PTFE O-rings include various conservative assumptions and estimates, as well as uncertainties. During a conference call on June 29, 1999, the applicant indicated that the SARP would be revised to demonstrate that the total quantity of releasable radioactive material (including the crud) was, at most, only slightly greater than an A_2 quantity and that radiation damage to the seals could not result in the release of more than an A_2 quantity of radioactive material in one week after an accident. The review staff concurs with that evaluation.

4.1.4.6 Demonstration of Compliance with Containment Design Criterion
As demonstrated in the Structural and Thermal Evaluation chapters of the SARP, the package closure system is not degraded by hypothetical accident conditions. Therefore, because the allowable release rate under hypothetical accident conditions is greater than under normal conditions of transport, the NAC-LWT cask satisfies the containment criteria of 10 CFR 71.51(a)(2).

4.1.5 Leakage Rate Tests

The fabrication, periodic and maintenance leakage rate criteria are each 5.5×10^{-7} cm^3/s (He) under the reference air leakage test conditions, as presented in the Acceptance Test and Maintenance Program chapter of the SARP. This criterion is based on requirements for spent fuel shipments and is more than three orders-of-magnitude less than that needed for the LTA contents.

The pre-shipment leakage rate test criterion is 10^{-3} ref cm^3/s, consistent with ANSI N14.5.

4.1.6 Appendix

The SARP appendix provides: (1) the military specification for the PTFE O-rings; (2) the manufacturer's technical bulletins for the metallic O-rings, and (3) the containment analysis for the LTA assembly.

4.2 Evaluation Findings

4.2.1 Findings

Based on review of the statements and representations in the NAC-LWT SARP, the staff concludes that the containment design has been adequately described and evaluated and that the package design meets the containment requirements of 10 CFR Part 71.

4.2.2 Conditions of Approval

Conditions of approval specified in the certificate of compliance include:

If the pre-shipment containment vessel monitoring indicates that any of the eight TPBARs are damaged or that the release rate of all radionuclides from the eight TPBARs exceeds 20 μCi/min (10 Ci/yr), the DOE Headquarters Certifying Official must be notified and the package cannot be shipped without further authorization. This number is based on 27.9 Ci/yr for 24 TPBARs.

Additionally, if the post-shipment monitoring indicates that more than 200 mCi has been released into the containment vessel, the DOE Headquarters Certifying Official must be notified and further shipments of the LTA contents cannot occur without additional authorization. Note that since the only gases identified to be present in the LTA contents are tritium and argon, the latter of which has an activity of millicuries, only the tritium is expected to contribute to the measurement. The value of 200 mCi is based on a release rate of 20 μCi/min over a one-week period.

4.3 References

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

5. SHIELDING REVIEW

5.1 Review Procedures

The following areas applicable to the Shielding Evaluation chapter of the SARP for the NAC-LWT were reviewed.

5.1.1 Description of Shielding Design

5.1.1.1 Design Features

The design features for gamma shielding in the NAC-LWT include a 14.6-cm (5.75-in.)-thick lead annulus (between the inner and outer shell of the package) and approximately 5.56 cm (2.19 in.) of steel in the cask body and neutron shield tank. In addition, the bottom end of the cask provides 19 cm (7.5 in.) of steel and 7.6 cm (3.0 in.) of lead shielding, and the top lid provides 28.7 cm (11.3 in.) of steel shielding. Although a shield plug is placed on top of the LTA container to minimize operational dose rates when the lid is removed, no credit for this plug assembly is considered in the shielding analysis.

The NAC-LWT packaging includes a neutron shield tank. Because the LTA has no neutron source, however, the shield tank is not filled with a neutron-absorbing liquid for LTA transport.

The text and sketches describing shielding design features in the Shielding Evaluation chapter of the SARP are consistent with the General Information chapter, the engineering drawings, and the models used in the shielding evaluation. The structural components that maintain the integrity of the shielding and maintain the contents in a fixed position within the package are consistent with those in the Structural Evaluation chapter; the heat transfer features to ensure the lead does not melt are consistent with those in the Thermal Evaluation chapter.

5.1.1.2 Codes and Standards

The flux-to-dose-rate conversion factors are consistent with ANSI/ANS6.1.1-1977 over the energy range of interest.

5.1.1.3 Summary Table of Maximum Radiation Levels

Table 5.3-22 of the SARP provides a summary of radiation levels for the LTA contents. These levels are consistent with the shielding evaluation.

5.1.2 Radiation Source

The LTA source terms are described in Section 5.3.7.1 and in Attachment 3 of the SARP. The contents used in the shielding evaluation are consistent with those specified in the General Information chapter.

The source terms presented in Tables 5.3-18 and 5.3-20 of the SARP are based on an irradiation time of 550 effective full power days and a cooling time of 30 days. These conditions are conservative compared with the limitations specified in the General Information chapter.

5.1.2.1 Gamma Source

The LTA consists of three activated components: TPBARs, thimble plugs, and the hold-down assembly. The primary gamma source is due to activated stainless steel. Source terms are calculated using ORIGEN2 with appropriate input parameters, including neutron activation cross sections. The staff confirmed the gamma source term as a function of energy using ORIGEN-S and found satisfactory agreement with the values in Table 5.3-18 in the SARP.

5.1.2.2 Neutron Source

There are essentially no neutrons emitted from the LTA.

5.1.3 Shielding Model

The shielding model is discussed in Sections 5.3.7 and 5.3.7.2 of the SARP. The impact limiter, LTA container assembly, PWR basket, and shield plug are ignored in the shielding analysis, and other dimensions are conservatively selected. Because the normal conditions of transport (NCT) and hypothetical accident condition (HAC) tests have no significant effect on the NAC-LWT shielding, the models used in the shielding calculation are consistent with the engineering drawings. (The HAC analysis of lead slump, as summarized in Table 5.1-6 of the SARP for PWR fuel contents, indicates that its effect is not significant.)

5.1.3.1 Configuration of Source and Shielding

The dimensions of the source and packaging used in the shielding models are appropriate. Dose-rate points on the surface and at 1 m and 2 m from the package are sufficient to address all locations at which maximum radiation levels are prescribed in §§71.47(a) or 71.47(b), and §71.51(a)(2). Voids, streaming paths, and irregular geometries are not applicable for this packaging.

The radial dose rates are determined at the central plane of each of the three components of an LTA, and all three contributions are summed to determine the radial dose rate conservatively. The axial dose rates are determined in a more realistic, but acceptable, manner.

5.1.3.2 Material Properties

The material properties used in the shielding models of the packaging and contents are presented in Table 5.3-21 of the SARP. These properties are appropriate and are consistent with materials specified on the engineering drawings. Shielding properties will not degrade significantly during the service life of the packaging.

The homogenous source region used in the shielding model is justified, and the homogenized mass densities used are appropriate.

5.1.4 Shielding Evaluation

The shielding evaluation is presented in Section 5.3.7.3 of the SARP.

5.1.4.1 Methods

The SARP uses the SAS1 module from the SCALE system to determine the gamma dose rate from the package. SAS1 is a one-dimensional code, satisfactory for the NAC-LWT evaluation, and the cross-section library (27n - 18γ) used by the code is appropriate for these shielding calculations. The results were confirmed using both the one-dimensional code MicroShield and the three-dimensional code MCNP4B.

5.1.4.2 Input and Output Data

Key input data for the shielding calculations are identified in the SARP. Input files are provided in Attachment 4 of the SARP, and information from representative input files was checked for accuracy. These input files were also run in SAS1, and the output was consistent with the results reported in the SARP.

5.1.4.3 Flux-to-Dose-Rate Conversion

The evaluation properly converts the gamma fluxes to dose rates using an abbreviated table of conversion factors from ANSI/ANS 6.1.1-1977. These factors are tabulated as a function of the energy group structure in Table 5.4-2 of the SARP.

5.1.4.4 External Radiation Levels

The external radiation levels under NCT and HAC agree with the summary tables.

The dose rates calculated in the SARP are conservative assessments. These external radiation levels are reasonable and agree with independent calculations conducted by the staff.

Because the normal conditions of transport have no significant effect on the package, the external radiation levels will not significantly increase under the tests specified in §71.71.

5.1.5 Appendix

Attachment 3 of the SARP provides additional detail on the source term, including input files for ORIGEN2. Attachment 4 provides input files for the SAS1 shielding evaluation.

5.2 Evaluation Findings

5.2.1 Findings

Based on review of the statements and representations in the SARP, 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 Part 71.

5.2.2 Conditions of Approval

No conditions of approval related to shielding are required.

5.3 References

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

Los Alamos National Laboratory. MCNP4B, Monte Carlo N-Particle Transport Code System, Los Alamos, NM, 1998.

Grove Engineering, Inc. *MicroShield, Version 4.2 User's Manual*, Rockville, MD, 1995.

Oak Ridge National Laboratory. ORIGEN-S, Standardized Computer Analyses for Licensing Evaluation (SCALE), Version 4.3, Oak Ridge, TN, 1998.

6. CRITICALITY REVIEW

The LTA contents for the NAC-LWT has no fissile material.

7. OPERATING PROCEDURES REVIEW

7.1 Review Procedures

The following areas applicable to the Operating Procedures chapter of the SARP for the NAC-LWT were reviewed.

7.1.1 Package Loading

The loading procedures for the LTA container are described in Section 7.1.8 of the SARP. Steps 1 through 16 identify the generic procedures associated with "Preparation for Loading," steps 17 though 35 identify the generic procedures associated with "Loading of Contents," and steps 36 through 41 identify the generic procedures associated with "Preparation for Transport." Except as discussed in Section 7.1.3 of this SER below, these procedures are very similar to those for loading spent nuclear fuel.

7.1.2 Package Unloading

The unloading procedures for the LTA container are described in Section 7.2.5 of the SARP. Steps 1 through 6 identify the generic procedures associated with the "Receipt of Package from Carrier," and Steps 7 though 17 identify the generic procedures associated with the "Removal of Contents." Again, these procedures are very similar to those for unloading spent nuclear fuel.

7.1.3 Other Procedures

Because the primary radionuclide in the LTA contents is tritium, additional tritium monitoring requirements are identified in many of the steps for both the loading and unloading procedures. Although the specific details on these monitoring requirements are not included in the SARP, detailed monitoring requirements will be incorporated into the site-specific operating procedures for the loading and unloading of the LTA contents. These additional tritium monitoring procedures are necessary to (1) protect the health and safety of the workers and (2) verify that the safety envelope described in Chapters 1, 2, and 4 of the SARP remain valid (see also Section 7.1.4 of this SER below).

7.1.4 Preparation of Empty Package for Transport

The procedures for the "Preparation of an Empty Package" for shipment are broadly described in Section 7.3 of the SARP, with additional procedural requirements described in Section 7.2.5 of the SARP (Steps 18 through 33). These additional requirements are incorporated to ensure that external and internal contamination levels for tritium meet the requirements of 49 CFR 173.443 and 49 CFR 173.428.

7.1.5 Appendix

The SARP does not include an appendix to Chapter 7.

7.2 Evaluation Findings

7.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the operating controls and procedures for the package meet the requirements of 10 CFR Part 71 and the appropriate sections of 49 CFR Parts 173-178. The staff also concludes that the operating controls and procedures are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

7.2.2 Conditions of Approval

Chapter 7 of the SARP is included in the certificate of compliance as a condition of package approval. In addition, tritium monitoring procedures must be implemented during loading and unloading of the package. The DOE Headquarters Certifying Official must be notified if (1) the radionuclide release rate measured prior to shipment exceeds the limit specified in Section 4.2.2 of this SER or (2) if the post-shipment quantity of tritium released from the TPBARs into the containment vessel exceeds the limit specified in Section 4.2.2 of this SER. A more detailed discussion of these limits is presented in Chapter 4 of this SER.

8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

8.1 Review Procedures

The following areas applicable to the Acceptance Tests and Maintenance Program chapter of the SARP for the NAC-LWT were reviewed.

8.1.1 Acceptance Tests

The acceptance tests for the NAC-LWT for the LTA contents are identical to those for the other contents and are deemed acceptable. Section 8.1.3 of the SARP, "Leakage Test Requirements," is also applicable to the requirements addressed in Chapter 7 of the SARP.

8.1.2 Maintenance Program

The maintenance program specifications for the NAC-LWT with the LTA contents are identical to those for the other contents, except that an additional requirement is added to provide an annual inspection by a metallurgist of all accessible containment surfaces, welds, heat-affected zones, and O-ring seating surfaces for evidence of corrosive attack or residue.

8.1.3 Appendix

The appendix in Chapter 8 of the SARP describes the lead pour procedure used to create the lead wall between the inner and outer shells of the NAC-LWT cask.

8.2 Evaluation Findings

8.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

8.2.2 Conditions of Approval

Chapter 8 of the SARP is incorporated in its entirety as a condition of approval in the certificate of compliance.

9. QUALITY ASSURANCE REVIEW

9.1 Review Procedures

The following areas applicable to quality assurance (QA) for the NAC-LWT were reviewed. Quality assurance is discussed primarily in Section 1.3 of the SARP, with copies of specific QA documentation provided in Attachment 1.

9.1.1 Description of QA Programs for Specific Organizations

9.1.1.1 Scope

The Quality Assurance Program Description (QAPD) for the Commercial Light Water Reactor (CLWR) Production of Tritium Project, included in Attachment 1 of the SARP, assigns responsibility for the scope and implementation of quality assurance to the CLWR Project Office Director, DP-62. This QAPD further specifies that the Tritium Production Project will comply with the requirements of 10 CFR Part 50, Appendix B; 10 CFR Part 71, Subpart H; and other applicable QA requirements.

Specific functional responsibilities assigned to the CLWR Project Office Director include:

Ensuring through reviews, inspections, surveillances, and audits that the CLWR Project Office and Field Project Participant activities are being performed in accordance with written and approved documents which comply with applicable requirements

Assessing the overall effectiveness of QA Program implementation, including identification of abnormal performance, precursors of problems, and opportunities for improvement

Recommending the suspension of unsatisfactory work and the control of further processing, delivery, or installation of non-conforming items

Establishing and maintaining an adequate, trained, and qualified staff, based on workload analysis.

Attachment 1 of the SARP describes responsibilities and provides QA documentation for organizations involved with the NAC-LWT package. These responsibilities include:

NAC International (NAC):
Obtain DOE certification, provide a transport-ready package with a PWR basket installed, perform all off-site transport activities

Watts Bar Nuclear Plant (TVA):
Load Lead Test Assembly (LTA) containers into the package and prepare it for shipment in accordance with site-specific procedures

Argonne National Laboratory West (ANL-W)
Load and unload LTA container into the package and prepare it for shipment in accordance with site-specific procedures

Test Area North (TAN)–Idaho National Environmental Engineering Laboratory serves as a backup facility for ANL-W if the latter's facilities are not available in time to support LTA shipment. Load and unload containers into the package and prepare it for shipment in accordance with site-specific procedures. (As of the date of this SER, informal notification has indicated ANL-W will unload the package.)

In addition to these organizations, the CLWR Project Office has recently assigned responsibility for tritium monitoring during package loading and unloading to the Savannah River Site (SRS). Although the SARP and its attachments do not include specific QA documentation for SRS, the CLWR QAPD identifies SRS as a Field Project Participant, subject to the same general QA requirements as those organizations listed above. Section 9.2.2 of this SER below recommends a condition of package approval to address changing programmatic requirements and the impracticality of a detailed QA review of all potential participating organizations by the DOE Headquarters Certifying Official.

9.1.1.2 QA Program Documentation and Approval

As indicated in Section 1.3 of the SARP, NAC has a quality assurance program for packages approved by the NRC under the provisions of 10 CFR Part 71, Subpart H. This approval (No. 0018) was issued October 23, 1980, and expires on December 31, 1999, unless renewed. A copy of the NRC approval letter is included in Attachment 1 of the SARP. The SARP indicates that NAC is responsible for design, fabrication, assembly, testing, maintenance, repair, and modification of the NAC-LWT. A review of NUREG-0383 verified that NRC has approved NAC's QA program for these activities.

TVA has a quality assurance program for packages approved by the NRC under the provisions of 10 CFR Part 71, Subpart H. This approval (No. 0227) was issued March 29, 1979, and expires on August 31, 1999, unless renewed. A copy of the NRC approval letter is included in Attachment 1 of the SARP. As noted in Section 9.1.1.1 of this SER above, the SARP indicates that TVA will participate in loading the package. A review of NUREG-0383 verified that NRC has approved TVA's QA program for package use.

Both ANL-W and TAN-INEEL have a quality assurance program for the Tritium Production Project. Because these organizations (as well as the CLWR Project Office) have responsibilities for other project activities in addition to those related to the transportation of LTAs, their QA plans are not limited to packages. The QA plans for each of these organizations have been approved by the CLWR Project Director, and a copy of these plans is included in Attachment 1 of the SARP.

9.1.1.3 Summary of 18 Quality Criteria and Cross-Referencing Matrices

Because the QA plans/programs of participating organizations have been approved by either the CLWR Project Office or NRC for compliance with 10 CFR Part 71, Subpart H, a detailed review of the quality criteria was not performed during SARP review. Specific QA program documentation submitted for NRC approval by NAC and TVA are not included in the docket. The review did, however, note that the CLWR QAPD was organized according to the 10-criteria format of DOE O 414.4, DOE 5700.6C, and 10 CFR 830.120; the TAN-INEEL QA Plan included 15 quality criteria. As noted above, these QA plans/programs are not restricted to package activities.

The CLWR QAPD includes a matrix that cross-references the 18 quality criteria of NQA-1 (and Subpart H) with the work breakdown structure numbering system depicted in the project master schedule for DP-62, DOE-CH, ANL-W, SRS, and other organizations participating in the overall Tritium Production Project. TAN-INEEL is not listed in the matrix of the QAPD revision provided in the SARP although its QA plan/program has subsequently been approved by the CLWR Project Director.

The ANL-W QA Plan for the Tritium Production Project includes a matrix that cross-references specific ANL-W QA procedures to the 18 criteria of NQA-1. The TAN-INEEL QA Plan lists specific QA-related procedures by functional area, rather than by the 18 quality criteria.

9.1.2 Package-Specific QA Requirements

As noted in Section 9.1 of this SER above, NAC's responsibilities include design, fabrication, assembly, testing, maintenance, repair, and modification of the NAC-LWT. The QA for these activities is addressed by NAC's QA program, which has been approved by the NRC. The package activities performed by ANL-W and TAN-INEEL pertain to package use, i.e., to loading and unloading the package (and related activities), as discussed in Chapter 7, "Operating Procedures," of the SARP.

Other QA-related documentation submitted in the SARP includes the engineering drawings (Section 1.4), Acceptance Tests (Section 8.1), and the Maintenance Program (Section 8.2).

The review deems the package-specific information presented in the SARP to be acceptable considering the requirement for organizations to have a Subpart H, QA program approved by either NRC or the CLWR Project Office; the general similarity of operating procedures for the NAC-LWT with those of other packages and package-related activities; and a graded approach to QA.

9.2 Evaluation Findings

9.2.1 Findings

Based on review of the statements and representations in the SARP, the staff concludes that the quality assurance program has been adequately described and meets the quality assurance requirements of 10 CFR Part 71. Subject to the conditions listed in Section 9.2.2 below, QA requirements appear adequate to assure that the package is designed, fabricated, assembled, tested, used, maintained, modified, and repaired in a manner consistent with its evaluation.

9.2.2 Conditions of Approval

Based on the QA review, the following conditions of approval are specified in the certificate of compliance:

Engineering drawings (SARP Section 1.4), Operating Procedures (Chapter 7), and Acceptance Tests and Maintenance Program (Chapter 8) are incorporated by reference into the certificate.

The certificate requires that all package activities be performed in accordance with a QA program that meets the requirements of 10 CFR Part 71, Subpart H, as approved by either NRC or the CLWR Project Office Director.

9.3 References

U.S. Nuclear Regulatory Commission, "Directory of Certificates of Compliance for Radioactive Material Packages," NUREG-0383, Volume 3, Revision 18, October 1998.