November 2001

Revision STC-01A

NAC-STC

NAC Storage Transport Cask

SAFETY ANALYSIS REPORT

Volume 2 of 2

Docket No. 71-9235

NAC INTERNATIONAL

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3.3 Technical Specifications for Components

The heat rejection capability of the NAC-STC is the result of passive heat transfer within the cask and from the cask surface. Heat is transferred from the fuel assemblies to the fuel basket tubes, and through the tubes and steel support disks and aluminum heat transfer disks to the fuel basket surface, by conduction, convection and radiation. The steel support disks are considered to be the structural member, and the aluminum heat transfer disks are supported by the steel structure and were added to enhance the heat rejection capacity of the basket. Heat is transferred from the basket surface to the cavity wall, or from the basket surface to the canister wall, and then to the cavity wall, primarily by conduction and radiation. Heat is then transferred by conduction through the cask wall to the inside of the neutron shield. For the gap considered to be present between the lead and the outer shell or inner shell, radiation is also considered to be active. There are 24 explosively bonded copper/stainless steel heat transfer fins. The stainless steel portion of the fin is primarily a structural member supporting the neutron shield and the copper is explosively bonded to it to aid in heat transfer through the neutron shield. The solid neutron shield region that covers the majority of the length of the cask transfers the heat by conduction to the shield tank surface. Because of the presence of the impact limiters, no heat is transferred to the environment through the ends of the cask. From the radial surfaces, heat is rejected to the environment by radiation and convection. The NAC-STC heat rejection components are analyzed for normal transport conditions in Section 3.4 and for hypothetical accident conditions in Section 3.5.

3.3.1 Radiation Protection Components

Radiation protection is provided by the NAC-STC gamma and neutron shielding. The primary gamma radiation shielding components are the materials used in fabricating the multiwall body, the end forgings of the cask body, the inner lid and the outer lid. The multiwall body consists of the cast lead enclosed between the inner and outer stainless steel shells. The lead is cast in place between the cylindrical cask body shells. Neutron shielding is provided by a radial solid neutron shield and 2-inch thick disks in the bottom of the cask and the inner lid. The neutron shields are borated to suppress secondary gamma generation. The capture of neutrons by many materials produces a secondary gamma ray that must also be shielded; however, when boron-10 absorbs a neutron, an alpha particle is emitted that is stopped locally. Thus, the secondary gamma dose rate is minimized. The radiation protection components are analyzed for normal transport conditions in Section 3.4 and for hypothetical accident event conditions in Section 3.5.

3.3.2 Safe Operating Ranges

There are four major components that must be maintained within their safe operating temperature ranges; the o-rings in the inner lid and inner lid port coverplate, the lead gamma shield, the NS-4-FR solid neutron shield, and the aluminum heat transfer disks.

The safe operating ranges for the o-rings, lead gamma shield, solid neutron shield and aluminum heat transfer disks are:

Component	Safe Operating Range
Lead gamma shield	- 40°F to + 600°F
Radial NS-4-FR neutron shield	- 40°F to + 300°F
Aluminum Heat Transfer Disks	- 40°F to + 600°F
Metallic o-rings	- 40°F to + 500°F
Viton o-rings	- 40°F to + 400°F
EPDM o-rings	- 65°F to + 300°F
(Short-term allowable + 375°F)	

The safe operating range of the o-rings is obtained from the technical information presented in Section 4.5, and ensures that the contents are contained within the cask and are not released to the atmosphere due to thermal failure of the o-rings. As shown in the EPDM o-ring technical information, a temperature of +375°F for up to 10 hours is acceptable in the accident condition. The analyses of Sections 3.4 and 3.5 show that the temperatures of the o-rings are maintained within the safe operating range during normal transport and hypothetical accident conditions.

The safe operating range of the lead gamma shield is based on preventing the lead from reaching its melting point of 620°F (Baumeister). To preclude localized lead temperatures from exceeding their safe operating range, FPC (fireblock silicone foam) is used to insulate the lead from the high temperatures that occur during the 10 CFR 71 hypothetical fire accident. The fire accident analysis in Section 3.5 shows that the lead temperature is maintained in its safe operating range even without the presence of the FPC. This foam is included for extra assurance of safety. A 0.125-inch layer of the material is located around the top and bottom corners of the lead gamma shield above and below the coverage provided by the radial neutron shield.

The maximum operating temperature limit of the NS-4-FR solid neutron shield material to ensure sufficient neutron shielding capacity was determined by the product developer to be 338°F, as shown in Figure 3.3-1. Test 2 of Figure 3.3-1 was conducted to evaluate the long-term stability of the NS-4-FR material at high temperatures. The test results were based on placing a

Figure 3.4-24 Two Dimensional Reconfigured Fuel Assembly Model

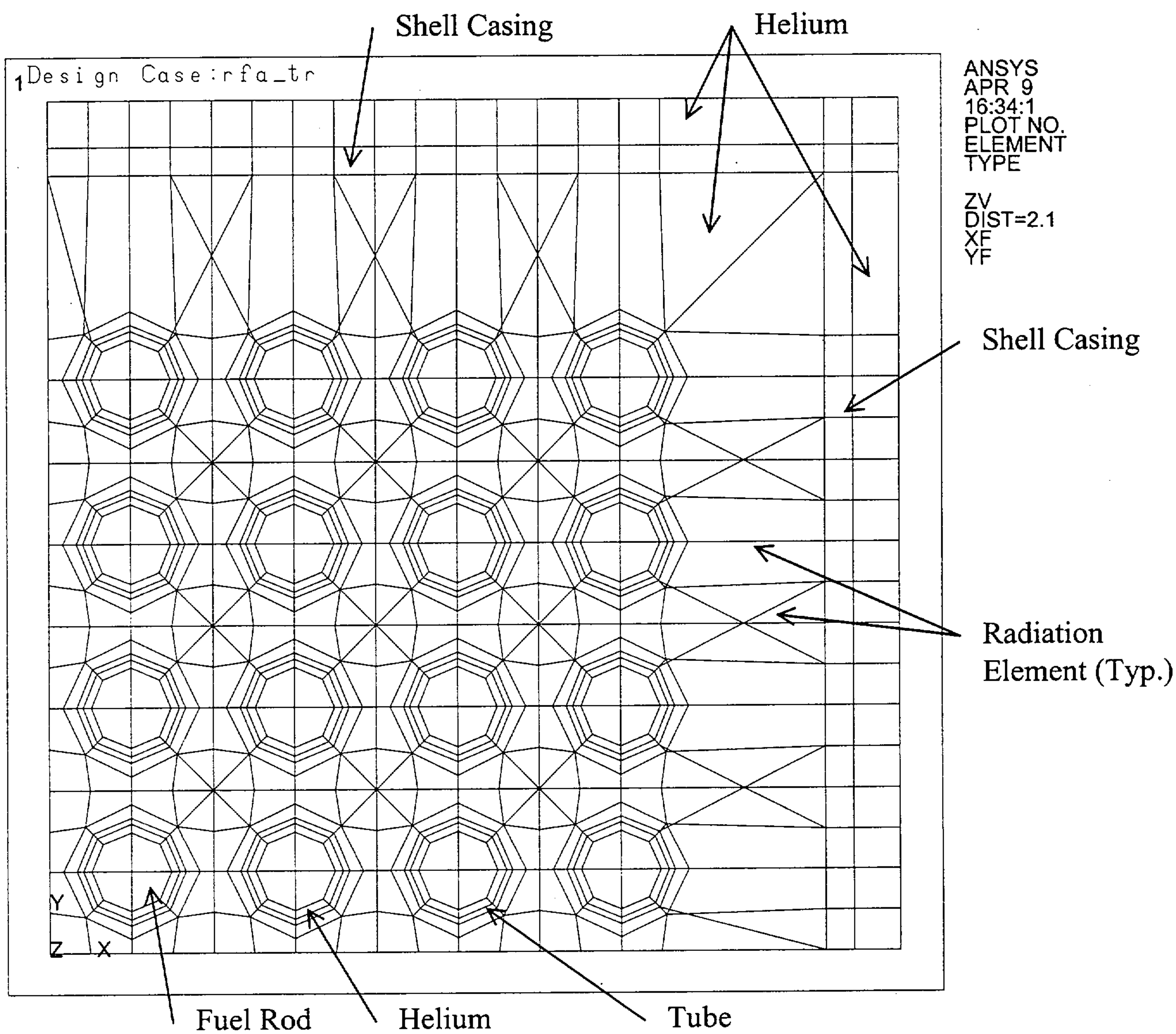

Table 3.4-1 Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Maximum Ambient Temperature - Directly Loaded and Canistered Configurations

Conditions: 100°F Ambient Temperature, Full Solar Insolation Decay Heat Load: 22.1 kW for Uncanistered Fuel; 12.5 kW for Canistered Fuel

	Directly Loaded (Uncanistered) Fuel			Canistered Fuel	
Component	**Air (°F)**	**Cavity Gas Helium (°F)**	**Note**	**Cavity Gas Helium (°F)**	**Note**
Outer Lid O-Ring	178	176	(1)	176	(4)
BTFE O-Rings	211	210	(1)	210	(4)
Metallic or Non-Metallic O-Rings	190	189	(1)	189	(4) (6)
Cask Radial Outer Surface	241	243	(2)	243	(5)
Top Neutron Shield	181	175	(1)	175	(4)
Radial Neutron Shield	284	285	(2)	270	(5)
Lead Gamma Shield	314	315	(2)	281	(5)
Aluminum Disk Exterior	338	337	(2)	---	---
Aluminum Disk Interior	491	487	(2)	536	(5)
Steel Support Disk Exterior	356	344	(2)	---	---
Steel Support Disk Interior	498	495	(2)	539	(5)
Canister Shell	---	---	---	338	(5)
Canister Lid	---	---	---	209	(5)
Canister Bottom Plate	---	---	---	255	(5)
Maximum Fuel Rod Cladding	588 (309°C)	544 (284°C)		575	(5)

Notes:
(1) Temperatures are determined from the analysis of the three dimensional quarter symmetry model of the entire cask.
(2) Temperatures are determined from the analysis of the three dimensional 180-degree section model of the entire cask.
(3) Temperatures are determined from the analysis of the two dimensional detailed model of the fuel assembly.
(4) Component not explicitly modeled in the 3-D model for canistered fuel. Temperature results from the helium case of the directly loaded fuel used (conservative).
(5) Temperatures are determined from the 3-D model for canistered fuel.
(6) Non-metallic o-rings may be either EPDM or Viton.

Table 3.4-2 Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Minimum Ambient Temperature - Directly Loaded and Canistered Configurations

Conditions: -40°F Ambient Temperature, 22.1 kW Decay Heat Load, 12.5 kW for Canistered Fuel, No Insolation

Component	Directly Loaded Fuel Temperature (°F)	Canistered Fuel Temperature (°F)	Notes
Outer Lid O-Ring	125	125	(1) (2)
PTFE O-Rings	125	125	(2)
Metallic or Non-Metallic O-Rings	129	129	(2) (4)
Cask Radial Outer Surface	144	116	(2) (3)
Top Neutron Shield	131	131	(2)
Radial Neutron Shield	181	142	(3)
Lead Gamma Shield	215	154	(3)
Fuel Basket Exterior	256	---	(3)
Maximum Basket Web	399	431	(3)
Canister Shell	---	215	(3)
Canister Lid	--	71	(3)
Canister Bottom Plate	---	121	(3)
Maximum Fuel Rod Cladding	488	473	(3)

1. Component not explicitly modeled in the 3-D model for canistered fuel. Temperatures from the directly loaded fuel are conservatively used.
2. Temperatures are determined from the 3-D model for canistered fuel.
3. Basket and fuel rod cladding temperatures are defined by adding the gradient result between the lead gamma shield and point of interest obtained from the 3-D finite element analysis with air in the cavity (Table 3.4-1).
4. Non-metallic o-rings may be either EPDM or Viton.

Table 3.4-3 Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Low Ambient, for Directly Loaded Fuel

Conditions:-20°F Ambient Temperature 22.1 kW Decay Heat Load No Insolance

Component	**Temperature (°F)**
Outer Lid O-Ring	161
PTFE O-Rings	165
Metallic, EPDM or Viton O-Rings	165
Cask Radial Outer Surface	173
Top Neutron Shield	168
Radial Neutron Shield	211
Lead Gamma Shield	245
Fuel Basket Exterior[1]	286
Maximum Basket Web[1]	429
Maximum Fuel Rod Cladding[1]	518

[1] Basket and fuel rod cladding temperatures are defined by adding the gradient result between the lead gamma shield and point of interest obtained from the 3-D finite element analysis with air in the cavity (Table 3.4-1).

Table 3.4-4 NAC-STC Thermal Performance Summary for Normal Conditions of Transport

	Uncanistered Fuel	**Canistered Fuel**	**Criterion**
Maximum Fuel Rod Cladding Temperature	309°C	302°C	<380°C—Uncanistered <340°C—Canistered
Component Safe Operating Temperature Ranges			
Metallic O-Rings	-40 to 190°F	-40 to 190°F	-40 to 500°F
EPDM O-Rings	-40 to 190°F	-40 to 190°F	-65 to 300°F[1]
Viton O-Rings	-40 to 190°F	-40 to 190°F	-40 to 400°F
Radial NS-4-FR Neutron Shield	-40 to 285°F	-40 to 270°F	-40 to 300°F
Lead Gamma Shield	-40 to 315°F	-40 to 281°F	-40 to 600°F
Aluminum Heat Transfer Disk	-40 to 491°F	-40 to 536°F	-40 to 600°F

1 +375°F for up to 10 hours in the accident condition.

Table 3.4-5 Maximum Cask Component Temperatures

Directly Loaded (Uncanistered) Fuel					Canistered Fuel	
Component ID	**Component Description**	**Air (°F)**	**Cavity Gas Helium (°F)**	**Note**	**Cavity Gas Helium (°F)**	**Note**
1	Bottom Plate	350	333	(1)	333	(3)
2	Bottom Forging	417	393	(1)	393	(3)
3	Transition Shell	300	300	(1)	300	(3)
4	Inner Shell	331	331	(2)	311	(4)
5	Outer Shell	292	293	(2)	276	(4)
6	Top Forging	211	210	(1)	210	(3)
7	Inner Lid	223	210	(1)	210	(3)
8	Outer Lid	178	176	(1)	176	(3)
9	Inner Lid Bolt	190	189	(1)	189	(3)
10	Outer Lid Bolt	178	176	(1)	176	(3)

Notes:
(1) Temperatures are determined from the analysis of the three-dimensional quarter symmetry model of the entire cask.
(2) Temperatures are determined from the analysis of the three-dimensional 180-degree section model of the entire cask.
(3) Component not explicitly modeled in the 3-D model for canistered fuel. Temperature results from the helium case of the directly loaded fuel used (conservatively).
(4) Temperatures are determined from the 3-D model for canistered fuel.

Table 3.5-1 Maximum Component Temperatures - Hypothetical Accident Conditions Fire Transient

Conditions: 30 minute, 1475°F Fire 22.1 kW decay heat for Directly Loaded Fuel 12.5 kW decay heat for Canistered Fuel

Component	**Temperature (°F)**	**Time (hours)**	**Allowable Temperature (°F)**
Inner Lid Bolt[1]	335[2]	0.9	--
Metallic O-Rings[1]	314	1.1	500
EPDM O-Rings[1, 3]	314	1.1	375
Viton O-Rings[1]	314	1.1	400
Cask Radial Outer Surface[1]	1347	0.5	--
Radial Neutron Shield[1, 4]	--	--	--
Lead Gamma Shield[1]	455	3.6	600
Support Disk Interior			
Directly Loaded Fuel	639[2]		
Canistered Fuel	713[2]		
Heat Transfer Disk Interior			
Directly Loaded Fuel	632		800
Canistered Fuel	710		800
Maximum Fuel Rod Cladding			
Directly Loaded Fuel	729		1058
Canistered Fuel	749		806

1 The maximum temperature is based on the decay heat value of 22.1 kW for Directly Loaded Fuel.

2 The maximum temperature is used to determine the allowable stress in the structural analyses.

3 Maximum temperature limit for a duration of 10 hours or less is 375°F.

4 The radial neutron shield is assumed to be lost at the end of the fire for conservatism. The axial neutron shields are not components important to safety as discussed in Section 3.3.2.

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4.0 CONTAINMENT

4.1 Containment Boundary

The NAC-STC transport containment boundary is designed and analyzed to ensure the containment of the cask contents in accordance with 10 CFR 71.51. The containment boundary is designed, fabricated and inspected in accordance with ASME Code Section III, Subsection NB, with the exception of code stamping.

The components of the containment boundary are described in Table 4.1-1 as a function of the containment condition and the contents. The containment conditions are:

- Containment Condition A: The containment boundary for the transport of directly loaded (i.e., no canister) intact PWR spent fuel assemblies following extended storage of the cask at an ISFSI licensed in accordance with 10 CFR 72.

- Containment Condition B: The containment boundary for the transport of: (1) directly loaded intact PWR spent fuel assemblies loaded immediately prior to transport using either metallic or non-metallic o-rings; or (2) canistered Yankee Class spent fuel assemblies, Reconfigured Fuel Assemblies or GTCC waste loaded into the NAC-STC immediately prior to transport using metallic o-rings.

The transportable storage canister is designed and analyzed to demonstrate that it maintains its structural integrity in accordance with the 10 CFR 71.63(b) requirement for a separate inner container for the Reconfigured Fuel Assembly (i.e., damaged fuel or fuel debris), which may contain more than 20 curies of plutonium.

The canister is leak tested at the time of loading to demonstrate that it satisfies the leaktight criteria of ANSI N14.5-1997 and the release limits of 10 CFR 71.63(b).

The NAC-STC containment boundary is designed to permit leak testing of the cask containment boundary penetrations prior to transport to confirm the leaktight integrity of the cask. The leak test criteria, minimum test sensitivity and leak test methods and locations for each containment condition are described in Table 4.1-1.

4.1.1 Containment Vessel

The primary containment vessel for the NAC-STC consists of a 71.0-inch inside diameter, 1.5-inch thick inner shell, two 1.5-inch to 2.0-inch thick transition sections, a 6.2-inch thick bottom inner forging, and a 7.85-inch thick top forging. The containment vessel components, except for the transition sections, are fabricated from ASME Boiler and Pressure Vessel Code, Type 304 stainless steel nuclear pressure vessel material. The two transition sections are ASME Boiler and Pressure Vessel Code, Type XM-19 stainless steel nuclear pressure vessel material.

The canister, which satisfies the requirements for a separate inner container, is a right circular cylinder constructed of 5/8-inch thick, Type 304L stainless steel plate. It is closed on the bottom end by a 1-inch thick Type 304L stainless steel plate, and at the top by a 5-inch thick, Type 304 stainless steel plate (shield lid) and by a 3-inch thick Type 304L stainless steel structural lid.

The canister shell welds are full penetration welds, which are radiographed. The bottom plate is joined to the canister shell by a full penetration groove weld and adjacent fillet weld, which are ultrasonically and liquid penetrant examined. The stainless steel material is selected to be compatible with the DOE MPC program guidelines for future disposal and to minimize the potential for any adverse chemical reactions in the spent fuel pool. The design of the shield lid and structural lid provides a redundant confinement boundary at the top of the canister.

The weld examination requirements for both the cask body and the transportable storage canister are defined in Table 4.1-2 and are shown on the License Drawings.

4.1.2 Containment Penetrations

The physical penetrations in the NAC-STC primary containment vessel are the inner lid and the vent and drain ports in the inner lid. The penetrations are designed to ensure sealing of the containment boundary and to ensure that the leakage from the boundary does not exceed 1×10^{-7} ref cm^3/sec using metallic o-rings and 3.1×10^{-5} ref cm^3/sec using non-metallic (EPDM or Viton) o-rings. The quick-disconnect fittings installed in the vent and drain openings and in the interseal test port in the inner lid are not considered to be part of the containment boundary.

The separate inner container (i.e., the transportable storage canister) is a completely welded vessel that has no operable penetrations.

4.1.3 Seals and Welds

4.1.3.1 Seals

The o-rings of the inner lid, the vent port coverplate, and the drain port coverplate are the seals that provide primary containment, as described in Section 4.1 and as shown in Table 4.1-1. Section 4.5 contains the specifications that describe the PTFE o-rings of the interlid and pressure port covers and the metallic or non-metallic (EPDM or Viton) o-rings used in the containment boundary and outer lid. Also included in Section 4.5 are the manufacturer's technical data bulletins for the expansion foam and the Fireblock Protective Coating (FPC) used in the NAC-STC. Leak testing of the cask is performed prior to acceptance from the manufacturer. Leak testing is also performed following fuel loading for either immediate transport or for transport following a storage period. Technical information for EPDM and Viton o-rings is provided in Sections 4.5.5 and 4.5.6, respectively.

4.1.3.1.1 Containment System Fabrication Verification

Upon completion of fabrication, a Containment System Fabrication Verification shall be performed on the cask containment boundary as described in Section 8.1.3. These leak tests verify that the leakage rate of the assembled containment does not exceed the maximum allowable leakage rate of 1×10^{-7} ref cm^3/sec using metallic o-rings or 3.1×10^{-5} ref cm^3/sec using EPDM or Viton o-rings. The allowable leak test shall conform to the o-ring design, since the inner and outer lids, and the vent and drain port coverplates must be fabricated using the o-ring groove appropriate to the o-ring design. Metallic o-rings and non-metallic o-rings cannot be used interchangeably.

4.1.3.1.2 Containment System Verification

The Containment System Verification shall be performed on the NAC-STC package containment boundary seals and components prior to each shipment, in accordance with the leak test acceptance criteria established for the Containment System Fabrication Verification. For cask transport immediately after loading, the leak test shall be performed in accordance with the procedures and acceptance criteria described in Section 7.4.1. For cask shipments following storage, the verification leak test shall be performed in accordance with the procedures and acceptance criteria described in Section 7.4.2.

Whenever a containment seal or component is replaced, the o-ring or containment component shall be leak tested following replacement using the Containment System Verification (Section

8.2.2.2). This test will verify that the replacement seal or component has been properly installed and that the leakage rate meets acceptance criteria.

4.1.3.2 Welds

The NAC-STC containment vessel and the transportable storage canister (separate inner container) are assembled by welding. A list of containment vessel and canister welds, the examinations and tests performed on the welds, and the applicable ASME Code acceptance criteria is provided in Table 4.1-2. The acceptance tests for the NAC-STC and for the transportable storage canister are provided in Section 8.1.

4.1.4 Closure

The primary closure assembly for the NAC-STC for transport consists of the inner lid, bolts, and o-rings. The inner lid is recessed and bolted into the top forging of the cask body. The 9.0-inch thick, 79.00-inch diameter inner lid is made of SA-336, Type 304 stainless steel. The inner lid is retained by 42 inner lid bolts that are 1 1/2 - 8 UN socket head cap screws fabricated from SB-637, Grade N07718 nickel alloy steel bolting material. The initial torque for installation of the inner lid bolts is specified in Table 7-1.

The vent port and the drain port are recessed into the inner lid. The vent and drain port coverplates are secured by four 1/2 - 13 UNC bolts fabricated from SA-193, Grade B6, Type 410 stainless steel. Each coverplate is sealed to the inner lid by a metallic or non-metallic o-ring, with a second, concentric o-ring of the same material, providing an annulus to test the seal.

A secondary closure is provided by the outer lid, which provides puncture protection to the primary closure assembly. The 5.25-inch thick, 86.7-inch diameter outer lid is made of SA-705, Type 630, H1150, 17-4 PH stainless steel. The outer lid is retained by 36 outer lid bolts that are 1 - 8 UNC socket head cap screws fabricated from SA-564, Type 630, H1150, 17-4 PH stainless steel. The initial torque for installation of the outer lid bolts is specified in Table 7-1. The bottom surface of the outer lid is sealed to the top forging of the cask body by a metallic or non-metallic o-ring.

Port covers protect the interlid and pressure ports, which are located in the top forging and access the region between the inner and outer lids. For transport operations, solid port covers with no penetrations are installed in the interlid and pressure ports. These port covers are secured by three 3/8 - 16 UNC bolts, fabricated from SA-193, Grade B6, Type 410 stainless steel material. Each cover is sealed to the cask body by two "piston-type" (bore seal) PTFE o-rings, with a test port located between the o-rings.

Table 4.1-1 NAC-STC Containment Boundaries

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
A (Primary) Using metallic o-rings.	Up to 26 directly loaded intact PWR spent fuel assemblies following storage operations per 10 CFR 72.	• Inner shell • Upper and lower shell rings (transition sections) • Bottom inner forging • Top forging • Inner lid • Inner lid outer metallic o-ring • Inner lid interseal test port threaded plug with metallic o-ring • Vent port coverplate • Vent port outer metallic o-ring • Vent port interseal port threaded plug with metallic o-ring • Drain port coverplate • Drain port coverplate outer metallic o-ring • Drain port coverplate interseal test port plug with metallic o-ring	• Allowable leakage rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) (i.e., leaktight) • Minimum test sensitivity is $\leq 1 \times 10^{-7}$ cm^3/s (helium)	• Vent port outer o-ring using helium sniffer probe method with helium in interseal region • Drain port outer o-ring using helium sniffer probe method with helium in interseal region • Inner lid outer o-ring using evacuated envelope method (envelope provided by outer lid with test performed through the interlid port) with helium in interseal region	These series of leak tests are performed on the NAC-STC containment boundary following directly loaded fuel storage operations. The outer o-rings are the designated boundary as access to the cask cavity to verify helium backfill conditions is not planned. Testing is performed in accordance with ANSI N14.5 requirements immediately prior to transport.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
B (Primary) Using EPDM or Viton o-rings.	Up to 26 directly loaded intact PWR spent fuel assemblies for immediate transport.	• Inner shell • Upper and lower shell rings transitional sections • Bottom inner forging • Top forging • Inner lid • Inner lid inner EPDM or Viton o-ring • Vent port coverplate • Vent port coverplate inner EPDM or Viton o-ring • Drain port coverplate • Drain port coverplate inner EPDM or Viton o-ring	• Allowable leakage rate is $\leq 4.1 \times 10^{-5}$ cm^3/s (helium) • Minimum test sensitivity is $\leq 2.0 \times 10^{-5}$ cm^3/s (helium)	• Inner lid inner o-ring using helium mass spectrometer leak detector (MSLD) connected to the interseal test port with pressurized helium in cask. • Vent port inner o-ring using the MSLD at the interseal test port with helium in the cask. • Drain port inner o-ring using the MSLD at the interseal test port with helium in the cask.	Testing is performed in accordance with ANSI N14.5 requirements immediately prior to transport.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
B (Primary) Using metallic o-rings.	Up to 26 directly loaded intact PWR spent fuel assemblies for immediate transport, or canistered Yankee Class spent fuel assemblies, Reconfigured Fuel Assemblies or GTCC waste.	• Inner shell • Upper and lower shell rings transitional sections • Bottom inner forging • Top forging • Inner lid • Inner lid inner metallic o-ring • Vent port coverplate • Vent port coverplate outer metallic o-ring • Vent port coverplate interseal port plug with metallic o-ring • Drain port coverplate • Drain port coverplate outer metallic o-ring • Drain port coverplate interseal port plug with metallic o-ring	• Allowable leakage rate is $\leq 2 \times 10^{-7}$ cm^3/s (helium) (i.e., leaktight) • Minimum test sensitivity is $\leq 1 \times 10^{-7}$ cm^3/s (helium)	• Inner lid inner o-ring using helium mass spectrometer leak detector (MSLD) connected to the interseal test port with pressurized helium in cask. • Vent port inner o-ring using the MSLD at the interseal test port with helium in the cask. • Drain port inner o-ring using the MSLD at the interseal test port with helium in the cask.	Testing is performed in accordance with ANSI N14.5 requirements immediately prior to transport.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
Separate Inner Container (Transportable Storage Canister)	Up to 36 Yankee-class intact or damaged (including fuel debris) spent fuel assemblies loaded in a transportable storage canister. The canister provides the separate inner container required for transport operations per 10 CFR 71.63 (b)	• Canister shell • Canister bottom • Canister shield lid • Canister vent port cover • Canister drain port cover • Canister structural lid	• Allowable leakage rate is $\leq 8 \times 10^{-8}$ cm^3/s (helium) (i.e., leaktight) • Minimum test sensitivity is $\leq 4 \times 10^{-8}$ cm^3/s (helium)	The separate inner containment provided by the canister assembly is leak tested using a helium leak detector at the shield lid to canister shell weld.	This test is performed during the canister closure operations described in Section 7.6.1. Subsequently, the drain port is welded to the shield lid and the structural lid is welded to the canister shell, thereby providing the redundant sealing required by 10 CFR 72.236(e).

Table 4.1-2 NAC-STC Containment Boundary Welds, Examinations and Tests

Primary Containment Boundary				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Inner shell longitudinal and inner shell rings longitudinal	Full Penetration Double Groove	A	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Inner shell circumferential	Full Penetration Double Groove	B	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leakage Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Inner shell to top and bottom inner shell rings circumferential	Full Penetration Double Groove	B	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leakage Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10; and ANSI N14.5

Table 4.1-2 NAC-STC Containment Boundary Welds, Examinations and Tests (Continued)

Primary Containment Boundary				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Top inner shell ring to upper forging and bottom inner shell ring to bottom inner forging	Full Penetration Double Groove	C	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Separate Inner Container				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Canister shell longitudinal	Full Penetration Double Groove	A	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Post-Loading Pneumatic (air-over-water) Pressure Test Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 Section V, Art. 10 and ANSI N14.5
Canister shell circumferential	Full Penetration Double Groove	B	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Post-Loading Pneumatic Pressure Test Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 Section V. Art. 10; and ANSI N14.5

Table 4.1-2 NAC-STC Containment Boundary Welds, Examinations and Tests (Continued)

Separate Inner Container				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Bottom plate to shell	Full Penetration Double Groove with Fillet Back Weld	C	VT on Tack Welds VT Final Pass PT Final Pass UT Final Weld Post-Loading Pneumatic Pressure Test Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5330 NB-6000 Section V, Art 10 and ANSI N14.5
Shield lid to shell	Partial Penetration Groove (Field Weld)	C	VT on Root Pass PT Root Pass VT Final Pass PT Final Pass Post-Loading Pneumatic Test Post-Pneumatic Test PT Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-5350 NB-4424 and NB-4427 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Vent and drain port covers to shield lid	Partial Penetration Groove (Field Weld)	D	VT on Root Pass PT Root Pass VT Final Pass PT Final Pass	NB-4424 and NB-4427 NB-5350 NB-5350 NB-4424 and NB-4427
Structural lid to shell	Partial Penetration Groove (Field Weld)	C	VT on Root Pass Volumetric Examination (UT) or Multi-Pass PT VT Final Pass	NB-4424 and NB-4427 NB-5330 (UT) or NB-5350 (PT) NB-4424 and NB-4427

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4.2 Containment Requirements for Normal Conditions of Transport

The NAC-STC has been designed to safely transport spent fuel assemblies in either of two configurations. The spent fuel assemblies may be sealed in a transportable storage canister (canistered), or loaded directly into a fuel basket installed in the cask cavity. In the canistered configuration, the NAC-STC can transport up to 36 fuel assemblies, depending on the mix of fuel assembly types within the canister, or up to 24 containers of GTCC waste. The NAC-STC is designed and tested to meet the requirements of 10 CFR 71.51 for containment of radioactive materials.

For directly loaded fuel, a reference 17 x 17 fuel assembly having a burnup of 60,000 MWD/MTU, an enrichment of 3.5 wt % ^{235}U and a cool time of 5 years is used to establish the source term for the containment analysis. The reference fuel assembly is also used in the shielding analysis and is described in Section 5.1.2. The source term of the reference fuel assembly bounds the licensed inventory as described in Chapter 5. For direct loading for immediate transport using metallic o-rings in the containment boundary, the containment boundary is tested to a leaktight condition as defined in ANSI N14.5-1997. For direct loading for immediate transport using EPDM or Viton o-rings, the containment boundary is tested to 3.1 x 10^{-5} ref cm^3/sec, or 4.1 x 10^{-5} cm^3/sec (helium).

For canistered fuel, the Combustion Engineering Type A fuel assembly has the highest burnup and, therefore, represents the limiting assembly for the design basis Yankee Class fuels. The source term for Reconfigured Fuel Assemblies is also bounded by the source term for the Combustion Engineering Type A fuel assemblies. While the releasable curie content of the 32,000 MWD/MTU CE (7.0-year cooled) fuel assemblies is higher than that of the design basis 8.1-year cooled and 36,000 MWD/MTU burned CE Type A fuel assembly, the A_2 value of the nuclide mixture is higher for the 32,000 MWD/MTU fuel assemblies. The higher A_2 value offsets the higher volumetric activity, so the CE 36,000 MWD/MTU burned (8.1-year cooled) fuel assembly has the bounding allowable release rate. While the 32,000 MWD/MTU Westinghouse fuel assembly at 19.0-years cooled has an A_2 value lower than that of the 36,000 MWD/MTU burned (8.1-year cooled) CE Type A fuel assembly, it also has a significantly lower releasable activity inventory, and is thus, bounded by the higher burned assembly.

The canistered GTCC waste consists primarily of irradiated and surface contaminated, stainless steel hardware and core components. The waste is separately enclosed in containers having the same external dimensions as the Yankee Class fuel. There are no radioactive gases associated with the GTCC waste.

The structural and thermal evaluations provided in Chapters 2 and 3 demonstrate that cask containment is maintained during normal conditions of transport. Therefore, the package satisfies the containment requirement of 10 CFR 71.51.

4.2.1 Containment of Radioactive Material

The NAC-STC uses one of two o-ring configurations based on the loading condition. For directly loading of fuel for transport without interim storage, the o-rings may be either metallic or non-metallic. For direct fuel loading for storage, the o-rings must be metallic. For loading of canistered fuel or GTCC waste for transport without interim storage, the o-rings must also be metallic. For configurations using metallic o-rings, the containment boundary is designed and tested to leaktight conditions as defined by ANSI N14.5-1997. For direct fuel loading for transport without interim storage using non-metallic (EPDM or Viton) o-rings, the allowable leak rate is calculated using the methodology of NUREG/CR-6487. Consequently, the cask meets the requirements of 10 CFR 71.51 and IAEA Safety Standard Series TS-R-1 (Paragraph 656) for directly loaded and for canistered fuel or GTCC waste. During final assembly, the canister closure is tested to leaktight conditions. Consequently, the canister provides the separate inner container (double containment) for the transport of Reconfigured Fuel Assemblies containing damaged fuel or fuel debris as required by 10 CFR 71.63.

4.2.2 Pressurization of Containment Vessel

The maximum normal operating pressure (MNOP) in the cask during normal transport conditions is calculated, based on 100 percent failure of the fuel rods, using the methodology presented in Section 3.4.4. The cask cavity under normal transport conditions is backfilled to one atmosphere with 99.9 percent pure helium gas. To determine the limiting temperature conditions, it has been assumed that the helium gas could possibly be replaced by air. Therefore, the normal operating pressure is determined for both gas conditions. From Section 3.4.4, the free gas volume, fuel fill gas volume, and fuel fission gas volumes for the two spent fuel configurations are presented below. The GTCC waste does not release any gas. The Reconfigured Fuel Assemblies contain failed fuel. The initial charge gas and any significant fission product gases have already been released from the Reconfigured Fuel Assemblies.

For normal transport conditions, 100 percent of the fuel rods are assumed to fail over a period of one year in transport in accordance with Regulatory Guide 7.8. Regulatory Guide 1.25 suggests that 10 percent of the tritium and 30 percent of the krypton-85 should be assumed to be available to escape each failed fuel rod. It is conservatively assumed that 30 percent of both tritium and krypton-85 escape each failed fuel rod. Other radiologically important gaseous nuclides are present only in negligible amounts after the 8.1-year minimum cooling period for the design basis canistered Yankee Class fuel and the 5-year minimum cooling period for directly loaded fuel. The postulated release of other radionuclides, including volatiles, fines, particulates and crud, does not contribute to an increase in internal pressure.

The calculated curie content of the fission gases in the design basis directly loaded PWR and Yankee Class canistered fuel for both normal and hypothetical accident conditions are:

Fission Gas	**Inventory - Directly Loaded Fuel**		**Inventory - Canistered Fuel**	
	Curies/Assembly	**Curies/ Cask**	**Curies/Assembly**	**Curies/ Cask**
Tritium	302	7,852	80.3	2,890
Krypton-85	4,540	118,040	1,390	50,000
Iodine-129	0.03	0.71	0.0	0.0

The directly loaded fuel inventory is based on fuel having a 3.5 wt % ^{235}U initial enrichment, 60,000 MWD/MTU burnup, and 5-year cool time, and the canistered fuel inventory is based on fuel having a 3.7 wt % ^{235}U initial enrichment, 36,000 MWD/MTU burnup and 8-year cool time.

4.2.2.1 Containment Pressurization Due to Directly Loaded Fuel

An increase in pressure within the containment boundary results from an increase in the cask cavity temperature and the postulated failure of 100 percent of the fuel rods in normal conditions of transport (MNOP).

Based on a bulk average gas temperature of 450°F when air is in the cavity (Section 3.4.4), the pressure within the cask cavity is 4.3 atm (63.2 psia = 48.5 psig). Based on a bulk average gas temperature of 401°F when helium is the cover gas, the pressure in the cask cavity is:

$$P_2 = (4.3)\left(\frac{478}{505}\right) = 4.07\,\text{atm} = 59.8\,\text{psia} = 45.1\,\text{psig}$$

This is less than the containment boundary design pressure of 75 psig.

4.2.2.2 Pressurization of Containment Due to Canistered Fuel

The MNOP during transport conditions in the transportable storage canister is calculated in Section 3.4.4, and found to be 3.23 atmospheres, or 32.8 psig. This pressure is conservatively calculated at 450°F, compared to the calculated maximum normal conditions of transport bulk gas temperature of 442°F. This pressure assumes the rupture of 100 percent of the fuel rods. The MNOP for the transportable storage canister is below the design pressure of 55 psig. As described above, the GTCC waste and Reconfigured Fuel Assemblies do not release gases to the canister cavity due to failures. Consequently, there is no increase in canister internal pressure due to these contents.

Since the canister does not fail in any of the evaluated normal transport or accident conditions, this pressure increase occurs within the canister. There is no increase in pressure in the cask cavity except that due to the increase in cavity temperature.

4.2.3 Containment Criterion for Normal Conditions of Transport

The NAC-STC is designed and tested to meet the containment criteria of 10 CFR 71.51 and IAEA Safety Standard Series Ts-R-1. The 10 CFR 71 limit for the release of radioactive material under normal conditions of transport is 10^{-6} A_2 per hour. This condition is met for the metallic o-ring configuration by testing the NAC-STC to leak tight conditions, 1 x 10^{-7} ref cm^3/ sec (air), which is equivalent to 2 x 10^{-7} cm^3/sec (helium) as defined by ANSI N14.5-1997. The leak test sensitivity is 1 x 10^{-7} cm^3/sec (helium), or less. This condition is met for the EPDM and Viton o-ring configurations by testing the NAC-STC to 3.1 x 10^{-5} ref cm^3/ sec (air), which is equivalent to 4.1 x 10^{-5} cm^3/sec (helium). The calculation of the allowable leak rate for the EPDM and Viton o-ring configurations is provided in Section 4.2.3.3. The leak test sensitivity is one-half of the allowable leak rate, or less, as recommended by ANSI N14.5-1997.

Correlation to Air Standard Conditions

The air standard leak rate is 1 x 10^{-7} ref cm^3/ sec, the leaktight condition as defined by Section 2.1 of ANSI N14.5-1997. Leak testing of the NAC-STC cask and the transportable storage canister is performed using helium gas. The NAC-STC cask leak test is performed using an allowable leak rate of 2 x 10^{-7} cm^3/sec (helium) with a detection sensitivity of 1 x 10^{-7} cm^3/sec (helium). The canister leak test is performed using an allowable leak rate of 8 x 10^{-8} cm^3/sec (helium) with a detection sensitivity of 4 x 10^{-8} cm^3/sec (helium).

4.2.3.1 Permissible Release Rate for the NAC-STC with Metallic O-Rings

Metallic o-rings must be used in the containment boundary when the cask is directly loaded for long-term storage and when the cask is loaded with a transportable storage canister for transport without interim storage. Metallic o-rings may also be used in the containment boundary when the cask is directly loaded for transport without interim storage. For the metallic o-ring configuration, the containment boundary is tested to leaktight conditions as defined by ANSI N14.5-1997 and, therefore, meets the requirements of 10 CFR 71.51 for containment of the radioactive contents.

Since the cask containment boundary is tested to demonstrate a leaktight condition, an allowable release rate, based on gases, fines, volatiles and particulates that are available for release from the directly loaded spent fuel or GTCC waste in the transportable storage canister, is not calculated.

4.2.3.2 Permissible Release Rate for the Transportable Storage Canister

The transportable storage canister welded closure is leak tested at final assembly to leak tight conditions, 1×10^{-7} ref cm^3/ sec, as defined by ANSI N14.5-1997. To meet this requirement, the allowable leak rate is 8×10^{-8} cm^3/sec (helium). The leak test sensitivity applied in testing the canister at the time it is closed is 4×10^{-8} cm^3/sec (helium), or less, to account for a test pressure difference of less than 1 atmosphere. Consequently, the canister provides adequate containment for the fuel or GTCC waste.

As shown in Section 2.6.13, the canister does not fail in any of the evaluated normal conditions of transport. Consequently, the canister and cask provide leak tight containment for Reconfigured Fuel Assemblies containing damaged fuel or fuel debris. This configuration meets the requirement of 10 CFR 71.63(b) for a separate inner container (double containment) for radioactive material containing more than 20 curies of plutonium.

4.2.3.3 Permissible Release Rate for the NAC-STC with EPDM or Viton O-Rings

EPDM or Viton non-metallic o-rings may be used in the containment boundary when the cask is directly loaded for transport without interim storage. For the EPDM and Viton o-ring configurations, the containment boundary is tested to 3.1×10^{-5} ref cm^3/sec. As described in this section, this leak rate meets the requirements of 10 CFR 71.51 for containment of the radioactive contents.

The 10 CFR 71.51 limit for the release of radioactive material under normal conditions of transport is 10^{-6} A_2/hr. In this analysis, A_2 for a mixed gas is determined by using the method described in 10 CFR 71, Appendix A. The release fractions for the various radionuclides transported in the NAC-STC are obtained from NUREG/CR-6487 and summarized in Table 4.2-1. The curie content for gases, volatiles, fines and particulates for the directly loaded 5-year cooled PWR reference fuel assembly is provided in Tables 4.2-6 through 4.2-8.

In addition to the radionuclides produced by the fuel material, fuel assemblies develop a coating of impurities deposited by cooling water during power generation. This coating is known as crud. Crud contains mostly non-radioactive elements but also contains a significant amount of ^{60}Co. NUREG/CR-6487 lists the maximum ^{60}Co concentrations on spent fuel assemblies to be 140 μCi/cm^2 for PWR assemblies at initial discharge. The surface area of the reference 17 x 17 PWR assembly is calculated to be 3.54 x 10^5 cm^2, based on the assembly characteristics provided in Table 5.2-2.

The maximum permissible leak rate from the cask under normal conditions of transport is determined from the 10 CFR 71 limit of 10^{-6} A_2/hr.

$$R_N = L_N C_N \le A_2 \times 1\times10^{-6}\,\text{hr}^{-1} \text{ or}$$
$$R_N = L_N C_N \le A_2 \times 2.78\times10^{-10}\,\text{sec}^{-1}$$

where:

L_N = Volumetric gas leakage rate [cm^3/s]
C_N = Curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]
R_N = Release rate for normal transport conditions [Ci/sec]

Activity Density of Radioactive Material (C_N)

The total inventories of fission product gases, volatiles, fines and crud are shown in Table 4.2-6 through Table 4.2-8. These inventories are calculated by using the source terms produced by the SAS2H sequence, the release fractions and the postulated crud (^{60}Co). The ^{60}Co content is decayed 5 years from discharge.

$$C_n = C_{Crud} + C_{Volaties} + C_{FissionGas} + C_{Fines}$$

$$C_{Crud} = \frac{f_C M_T}{V} = \frac{f_C S_C N_A (N_R S_{AR})}{V}$$

where:

C_{crud}	=	Activity density inside containment vessel resulting from crud spallation [Ci/cm^3]
M_T	=	Total crud activity inventory [Ci]
f_c	=	Crud spallation factor
V	=	Free volume inside containment vessel [cm$_3$]
S_C	=	Crud surface activity [Ci/cm^2]
N_R	=	Number of fuel rods per assembly
N_A	=	Number of assemblies
S_{AR}	=	Surface area per rod [cm^2]

and,

$$C_{fines} = \frac{f_F W_R A_R N_R N_A f_B}{V}$$

where:

C_{fine}	=	Activity concentration inside containment vessel resulting from fines released from cladding breaches [Ci/cm^3]
f_F	=	Fraction of fuel rod mass released as fines resulting from cladding breach
f_B	=	Fraction of fuel rods that develop cladding breach
W_R	=	Mass of the fuel in fuel rod [g]
N_R	=	Number of fuel rods per assembly
N_A	=	Number of assemblies
A_R	=	Specific activity of fines emitted from cladding breach in fuel rod [Ci/g]
V	=	Containment vessel void volume [cm^3]

and,

$$C_{vg} = C_{vol} + C_{gas} = \frac{N_R N_A f_B W_R (A_V f_V + A_G f_G)}{V}$$

where:

C_{vg}	=	Releasable activity concentration inside the containment vessel resulting from gases and volatiles released from cladding breaches [Ci/cm^3]
C_{vol}	=	Releasable activity concentration inside the containment vessel resulting from volatiles released from cladding breaches [Ci/cm^3]
C_{gas}	=	Releasable activity concentration inside the containment vessel resulting from gases released from cladding breaches [Ci/cm^3]
W_R	=	Mass of the fuel in a fuel rod [g]

N_R	=	Number fuel rods per assembly
N_A	=	Number of assemblies
f_B	=	Fraction of rods that develop cladding breaches
A_V	=	Specific activity of volatiles in fuel rod [Ci/g]
f_V	=	Fraction of volatiles in fuel rod released if rod develops cladding breach
A_G	=	Specific activity of gas in fuel rod [Ci/g]
f_G	=	Fraction of gas that would escape from fuel rod that develops cladding breach
V	=	Void volume inside containment vessel [cm^3]

Activity Values for Radionuclides

A_2 values used in this analysis (based on 10 CFR 71 Appendix A) are listed in Tables 4.2-6 through 4.2-8 for all radionuclides produced by the SAS2H analysis (plus ^{60}Co). The mixture A_2 value is shown in Table 4.2-2. For those isotopes for which no specific A_2 values are given in 10 CFR 71 Appendix A, the generic values listed in Table A.2 of Appendix A are applied. A_2 values for mixed isotopes are calculated from the following:

$$A_2 = \frac{1}{\sum \frac{F_i}{A_2^i}}$$

where:

$$F_i = \frac{S_i}{S_n}$$

and

F_i = Fraction of isotope i with respect to the entire mixture

S_i = Activity of isotope i [Ci]

S_n = Total group activity [Ci]

Mixture A_2 values are determined for gas, volatile, fine and crud mixtures and are then combined for a total cask mixture A_2 value. Table 4.2-2 provides the source term and A_2 values per group for directly loaded PWR fuel release rate calculations.

Maximum Allowable Leak Rate for EPDM and Viton O-Rings

On the basis of the methodology described, the maximum allowable leak rate for PWR fuel directly loaded for immediate transport in normal conditions of transport is calculated to be 3.1×10^{-5} cm^3/sec (Table 4.2-3).

Correlation of Allowable Leak Rates to Air Standard

The volumetric gas leak rate, L, is independent of cask pressure and temperature. The maximum allowable release must be correlated with air standard leak rates, which depend on gas temperatures, pressures, and leakage path length and diameter. This correlation requires calculation of the capillary opening diameter through which the flow occurs. Depending on pressure and condition of the flow, a combination of continuum and molecular flow occurs.

Continuum flow and molecular flow equations are obtained from NUREG/CR-6487, Section 2, which are adjusted to upstream flow rate in accordance with NUREG/CR-6487 and ANSI N14.5-1997. The continuum volumetric flow rate of the gas (cm^3/sec), L_c, is given by:

$$L_c = \frac{2.48x10^6 D^4}{a\mu}(P_u - P_d) * \frac{P_a}{P_u} = F_c * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

F_c	=	Coefficient for continuum flow [cm^3/atm-s]
D	=	Capillary diameter [cm]
a	=	Capillary length [cm]
μ	=	Fluid viscosity [cP]
P_u	=	Upstream pressure [atm] – pressure inside containment
P_d	=	Downstream pressure [atm] – pressure outside containment
P_a	=	Average pressure $(P_u+P_d)/2$ [atm]

and, the molecular volumetric flow rate of the gas (cm^3/sec), L_m, is given by:

$$L_m = \frac{3.81x10^3 D^3 \sqrt{\frac{T}{M}}}{aP_a}(P_u - P_d) * \frac{P_a}{P_u} = F_m * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

L_m	=	Volumetric flow rate of gas at P_a [cm^3/sec]
F_m	=	Coefficient for molecular flow [cm^3/atm-s]
D	=	Capillary diameter [cm]
T	=	Gas temperature [K]
M	=	Gas molecular weight [g/mole]
P_a	=	Average pressure $(P_u+P_d)/2$ [atm]
P_u	=	Upstream pressure [atm]
P_d	=	Downstream pressure [atm].
a	=	Capillary diameter [cm]

For this analysis, the gas temperature used for molecular flow analysis is identical to the upstream temperature. Pressure and temperature at normal operating conditions are summarized in Table 4.2-4. Based on the pressure, temperature and allowable leakage rate (L_N), the capillary diameter of the leak is determined. The calculated capillary diameter is then used to determine the air standard leak rate and helium test leak rate. Air standard condition leak rates are determined for air leaking from 1 atmosphere to 0.01 atmosphere at a temperature of 298°K. The test gas is helium leaking from 1 atmosphere (0 psig) to a vacuum. Table 4.2-3 provides the reference and test leak rates. The sensitivity for these tests is one-half the air reference leak rate as recommended by ANSI-N14.5-1997. Key containment analysis parameters are summarized in Table 4.2-5.

This analysis is conservative, since a higher upstream pressure, which could result from a higher average gas temperature based on decay heat, results in a higher allowable leak rate assuming that the leak path length and the leak path diameter (calculated based on the reference air condition) are held constant. Since the test condition pressure cannot be less than 1 atmosphere and the average gas temperature does not have a first order effect on calculated leak rate, the helium test condition is conservative with respect to the allowable reference leak rate.

Table 4.2-1 Release Fractions: Normal and Accident Conditions

Radionuclide Origin	Fraction: Normal Conditions	Fraction: Accident Conditions
Volatiles releasable	2.00E-04	2.00E-04
Fission gas releasable	0.3	0.3
Rod mass released	3.00E-05	3.00E-05
Crud spallation factor	0.15	1.0
Fraction of fuel that fails	0.03	1.0

Table 4.2-2 Allowable Release Rate Source and A_2 Values for Directly Loaded PWR Fuel: Normal Conditions

Reference 17 x 17 PWR Fuel[1]	Crud	Gas	Volatiles	Fines	Total
Total Activity per Assembly (Ci)	N/C[2]	4.84E+03	1.76E+05	2.70E+05	4.51E+05
Releasable Activity per Cask (Ci)	1.00E+02	1.13E+03	2.75E+01	6.32E+00	1.27E+03
Cask Volumetric Activity (Ci/cm^3)	1.35E-05	1.52E-04	3.69E-06	8.49E-07	1.70E-04
A_2 Value (Ci)	10.80	283.25	6.08	0.13	300.26
Fraction of Activity	0.079	0.894	0.022	0.005	1.000
Fraction of Activity / A_2 (1/Ci)	0.0073	0.0032	0.0036	0.0391	0.0532
			Mixture A_2 Value (Ci)		18.80

1. Based on 3% rod failure.
2. Not explicitly calculated.

Table 4.2-3 Leak Rate and Leak Test Sensitivity: Normal Conditions

Contents	O-Rings	Vol. Activity (Ci/cm^3)	Leak Rate (cm^3/sec)		
			Volumetric (L)	Air Reference (L_R)	Test Sensitivity
Directly Loaded Reference 17 x 17 PWR Fuel	EPDM or Viton	1.7E-04	3.1E-05	3.1E-05[1]	1.6E-05[1]

1. The corresponding helium test leak rates and leak test sensitivities for the directly loaded PWR fuel configuration are 4.1 x 10^{-5} cm^3/sec and 2.0 x 10^{-5} cm^3/sec, respectively, at standard conditions.

Table 4.2-4 Cask Free Volumes and Pressures: Normal and Accident Conditions

Contents	Directly Loaded Reference PWR Fuel	
Cask Operating Condition	Normal	Accident[1]
Free Gas Volume (liters)[2]	7440	7540
Pressure (atm)[2]	1.80	5.72
Average Gas Temperature (K)	505.0	675.0

1. The accident condition for this analysis is 100% rod failure in combination with a fire accident that raises the cask temperature. This hypothetical dual-failure accident conservatively maximizes both available releasable material and cask pressure.

2. Bounding values were chosen for free volume (minimum) and pressure (maximum). This conservatively minimizes free volume and capillary diameter.

Table 4.2-5 Containment Parameters for Non-Metallic O-Rings in Normal Conditions of Transport

Contents	Crud Surface Activity (Ci/cm^2)	Containment Free Volume (cm^3)	Capillary Length (cm)	Capillary Diameter (cm)	Upstream Pressure (atm)	Gas Temperature (K)
Reference 17 x 17 PWR Fuel	7.3E-5	7.44E+6	0.597	6.9E-4	1.80	505

Note: Based on 3% of the fuel rods failing in normal transport conditions.

Table 4.2-6 17x17 Reference Fuel SAS2H Output and Group A_2 Values – Gases

Isotope	Curies	Fraction of Source	A_2 Value	Frac/A_2	Composite A_2
H 3	3.02E+02	6.24E-02	1080	5.775E-05	
I129	2.73E-02	5.64E-06	1E+60	5.638E-66	
KR 85	4.54E+03	9.38E-01	270	3.473E-03	
Total	4.84E+03			3.53E-03	283.252

Table 4.2-7 17x17 Reference Fuel SAS2H Output and Group A_2 Values – Volatiles

Isotope	Curies	Fraction of Source	A_2 Value	Frac/A_2	Composite A_2
CS134	3.32E+04	1.89E-01	13.5	1.397E-02	
CS135	3.35E-01	1.90E-06	24.3	7.833E-08	
CS137	7.96E+04	4.52E-01	13.5	3.350E-02	
RU106	1.51E+04	8.58E-02	5.41	1.586E-02	
SR 90	4.81E+04	2.73E-01	2.7	1.012E-01	
Total	1.76E+05			1.65E-01	6.077

Table 4.2-8 17x17 Reference Fuel SAS2H Output and Group A_2 Values – Fines

Isotope	Curies	Fraction of Source	A_2 Value	Frac/A_2	Composite A_2
AG108	1.19E-03	4.41E-09	0.5	8.82E-09	
AG108M	1.36E-02	5.04E-08	16.2	3.11E-09	
AG109M	1.22E-04	4.52E-10	0.5	9.04E-10	
AG110	4.80E-01	1.78E-06	0.5	3.56E-06	
AG110M	3.53E+01	1.31E-04	10.8	1.21E-05	
AM241	6.97E+02	2.58E-03	0.00541	4.77E-01	
AM242	6.67E+00	2.47E-05	0.5	4.94E-05	
AM242M	6.70E+00	2.48E-05	0.00541	4.59E-03	
AM243	4.18E+01	1.55E-04	0.00541	2.86E-02	
BA137M	7.51E+04	2.78E-01	0.5	5.56E-01	
BI212	5.01E-02	1.86E-07	8.11	2.29E-08	
BK249	5.88E-03	2.18E-08	2.16	1.01E-08	
C 14	9.81E-05	3.63E-10	54.1	6.72E-12	
CD109	1.22E-04	4.52E-10	27	1.67E-11	
CD113M	2.75E+01	1.02E-04	2.43	4.19E-05	
CE144	7.05E+03	2.61E-02	5.41	4.83E-03	
CF249	8.42E-04	3.12E-09	0.00541	5.77E-07	
CF250	3.38E-03	1.25E-08	0.0135	9.27E-07	
CF251	4.18E-05	1.55E-10	0.00541	2.86E-08	
CF252	5.19E-03	1.92E-08	0.027	7.12E-07	
CM242	2.73E+01	1.01E-04	0.27	3.75E-04	
CM243	3.20E+01	1.19E-04	0.00811	1.46E-02	
CM244	7.77E+03	2.88E-02	0.0108	2.67E+00	
CM245	9.51E-01	3.52E-06	0.00541	6.51E-04	
CM246	5.93E-01	2.20E-06	0.00541	4.06E-04	
CM248	3.94E-05	1.46E-10	0.00135	1.08E-07	
EU152	2.36E+00	8.74E-06	24.3	3.60E-07	
EU154	8.15E+03	3.02E-02	13.5	2.24E-03	
EU155	3.88E+03	1.44E-02	54.1	2.66E-04	
GD153	1.11E-01	4.11E-07	135	3.05E-09	
HO166M	1.68E-03	6.22E-09	8.11	7.67E-10	
NB 93M	2.45E-01	9.08E-07	162	5.60E-09	
NB 94	8.06E-05	2.99E-10	16.2	1.84E-11	
NB 95	4.30E-03	1.59E-08	27	5.90E-10	
NP235	1.90E-03	7.04E-09	1080	6.52E-12	
NP236	5.62E-05	2.08E-10	0.027	7.71E-09	
NP237	2.82E-01	1.04E-06	0.00541	1.93E-04	
NP238	3.01E-02	1.12E-07	0.5	2.23E-07	
NP239	4.18E+01	1.55E-04	13.5	1.15E-05	

Table 4.2-8 17x17 Reference Fuel SAS2H Output and Group A_2 Values – Fines (Continued)

Isotope	Curies	Fraction of Source	A_2 Value	Frac/A_2	Composite A_2
PA233	2.82E-01	1.04E-06	24.3	4.30E-08	
PA234	1.86E-04	6.89E-10	0.5	1.38E-09	
PA234M	1.43E-01	5.30E-07	0.5	1.06E-06	
PB212	5.01E-02	1.86E-07	8.11	2.29E-08	
PD107	1.27E-01	4.70E-07	1E+60	4.70E-67	
PM145	1.32E-01	4.89E-07	189	2.59E-09	
PM146	2.37E+00	8.78E-06	0.5	1.76E-05	
PM147	2.17E+04	8.04E-02	24.3	3.31E-03	
PO212	3.21E-02	1.19E-07	0.000541	2.20E-04	
PO216	5.01E-02	1.86E-07	0.000541	3.43E-04	
PR144	7.05E+03	2.61E-02	0.5	5.22E-02	
PR144M	9.86E+01	3.65E-04	0.5	7.31E-04	
PU236	6.02E-01	2.23E-06	0.0189	1.18E-04	
PU238	3.84E+03	1.42E-02	0.00541	2.63E+00	
PU239	1.54E+02	5.70E-04	0.00541	1.05E-01	
PU240	3.22E+02	1.19E-03	0.00541	2.20E-01	
PU241	6.82E+04	2.53E-01	0.27	9.36E-01	
PU242	2.52E+00	9.34E-06	0.00541	1.73E-03	
RA224	5.01E-02	1.86E-07	1.62	1.15E-07	
RH102	5.61E-01	2.08E-06	13.5	1.54E-07	
RH106	1.51E+04	5.59E-02	0.5	1.12E-01	
RN220	5.01E-02	1.86E-07	0.000541	3.43E-04	
SB125	1.81E+03	6.71E-03	24.3	2.76E-04	
SB126	7.40E-02	2.74E-07	10.8	2.54E-08	
SB126M	5.29E-01	1.96E-06	0.5	3.92E-06	
SE 79	5.39E-02	2.00E-07	54.1	3.69E-09	
SM145	3.34E-02	1.24E-07	541	2.29E-10	
SM151	2.33E+02	8.63E-04	108	7.99E-06	
SN119M	6.58E-01	2.44E-06	1080	2.26E-09	
SN121	1.64E+00	6.08E-06	0.5	1.22E-05	
SN121M	2.11E+00	7.82E-06	24.3	3.22E-07	
SN123	1.84E-02	6.82E-08	13.5	5.05E-09	
SN126	5.29E-01	1.96E-06	8.11	2.42E-07	
TC 99	1.00E+01	3.70E-05	24.3	1.52E-06	
TE123M	3.99E-04	1.48E-09	189	7.82E-12	
TE125M	4.42E+02	1.64E-03	243	6.74E-06	
TE127	8.06E-02	2.99E-07	13.5	2.21E-08	
TE127M	8.23E-02	3.05E-07	13.5	2.26E-08	
TH228	4.99E-02	1.85E-07	0.0108	1.71E-05	

Table 4.2-8 17x17 Reference Fuel SAS2H Output and Group A_2 Values – Fines (Continued)

Isotope	Curies	Fraction of Source	A_2 Value	Frac/A_2	Composite A_2
TH231	2.32E-03	8.59E-09	24.3	3.54E-10	
TH234	1.43E-01	5.30E-07	5.41	9.79E-08	
TL208	1.80E-02	6.67E-08	0.5	1.33E-07	
TM171	5.13E-04	1.90E-09	270	7.04E-12	
U232	7.30E-02	2.70E-07	0.00811	3.33E-05	
U233	9.82E-06	3.64E-11	0.027	1.35E-09	
U234	6.91E-02	2.56E-07	0.027	9.48E-06	
U235	2.32E-03	8.59E-09	1E+60	8.59E-69	
U236	1.45E-01	5.37E-07	0.027	1.99E-05	
U237	1.63E+00	6.04E-06	0.5	1.21E-05	
U238	1.43E-01	5.30E-07	1E+60	5.30E-67	
Y 90	4.81E+04	1.78E-01	5.41	3.29E-02	
Y 91	1.79E-04	6.63E-10	8.11	8.18E-11	
ZR 93	8.99E-01	3.33E-06	5.41	6.16E-07	
ZR 95	1.89E-03	7.00E-09	24.3	2.88E-10	
Total	2.70E+05			7.85E+00	0.127

4.3 Containment Requirements For Hypothetical Accident Conditions

The NAC-STC has been designed to safely transport 26 directly loaded PWR fuel assemblies, or in the canister configuration, up to 36 Yankee Class fuel assemblies or 24 containers of GTCC waste. The structural integrity of the cask containment during hypothetical accident conditions is demonstrated in Section 2.7. Therefore, the cask containment is maintained under hypothetical accident conditions. As described in Section 2.7.11, the transportable storage canister does not fail in any of the evaluated transport accident conditions defined in 10 CFR 71.73. Consequently, for the configurations using metallic o-rings, the leaktight condition is maintained in the hypothetical accident conditions. As described in Section 4.1, metallic o-rings must be used for the direct loading of fuel for long-term storage and for the transportable storage canister loaded into the cask for transport without interim storage. Either metallic or non-metallic (EPDM or Viton) o-rings may be used for spent fuel that is directly loaded for transport without interim storage.

For direct loading for transport without interim storage using EPDM or Viton o-rings, the containment boundary requirement under hypothetical accident conditions is met by ensuring that a leak rate limit of 1.5×10^{-3} ref$\cdot$cm^3/sec is not exceeded. Calculation of this limit is provided in Section 4.3.2.

Assuming a simultaneous occurrence of a fire accident and a 100% rod failure, and on the basis of a bulk average gas temperature of 675K resulting from air in the cavity, the pressure within the cask cavity is calculated to be 5.72 atm. The hypothetical presence of air in the cask provides an upper bound on the gas temperature. This pressure represents the maximum possible cask internal pressure.

4.3.1 Fission Gas Products

The calculated amounts of fission gases contained by the design basis directly loaded and canistered PWR fuel assemblies for both normal and hypothetical accident conditions are reported in Section 4.2.2. The accident conditions assume a 100% fuel rod failure with 30% of the available tritium and 30% of the available krypton-85 being released to the cask cavity or to the canister. These gases contribute to an increase in the cask cavity pressure due to the postulated failure of the directly loaded, intact, fuel and to an increase in the canister pressure due to the postulated failure of the canistered fuel.

Other released radionuclides, including crud, volatiles, fines and particulates, are not assumed to contribute to an increase in internal pressure of either transport configuration. The GTCC waste does not contain any gaseous products and does not have a failure mode in the hypothetical accident conditions. Consequently, there is no increase in pressure due to the GTCC contents. The Reconfigured Fuel Assemblies contain fuel already classified as failed. Consequently, the initial charge gas and the fission product gases have already been released. There is no additional release of gases in the hypothetical accident events due to the Reconfigured Fuel Assembly contents.

4.3.2 Containment of Radioactive Material

For directly loaded fuel intended for transport without interim storage using metallic o-rings, the containment boundary is tested to a leaktight condition as defined in ANSI N14.5-1997. As shown in Section 2.7 for the NAC-STC cask and in Section 2.7.11 for the transportable storage canister, the containment boundary of the cask and canister do not fail during the hypothetical accident events. Consequently, leaktight containment is maintained by both the cask and the canister in the hypothetical accident events. For the EPDM or Viton o-ring configurations, the allowable leak rate in the hypothetical accident condition is 1.5 x 10^{-3} cm^3/sec, as shown in Section 4.3.3. The curie content of the reference PWR fuel assembly used for the analysis is provided in Section 4.2.3.

4.3.3 Calculation of Allowable Leak Rate

The allowable leak rates under hypothetical accident conditions are calculated by using the method described in Section 4.2.1.1 for normal conditions of transport. The total inventories of fission product gases, volatiles, fines and crud are calculated by using the source terms generated by SAS2H, using the release fractions. Using the A_2 values from 10 CFR 71, Appendix A (Table 4.3-1), the mixture A_2 values are determined for gas, volatile, fine and crud mixtures. Finally, the maximum allowable release rates are calculated by using the hypothetical accident conditions allowable release limit:

$$R_A = L_A C_A \leq A_2 \cdot \text{per week}$$

or

$$R_A = L_A C_A \leq 1.65\text{x}10^{-6}\, A_2 \text{ per sec}$$

where:

L_A	=	Volumetric gas leakage rate [cm^3/sec]
C_A	=	Curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]
R_A	=	Release rate for accident transport conditions

The assumptions applied to the calculations for the hypothetical accident conditions are that 100% of the fuel rods fail and that 100% of the assumed crud is released. The gas, volatile, fine and crud mixture A_2 is not affected by the change in the magnitude of releasable material. However, the combined A_2 changes based on the change in activity fraction in each group.

The calculated maximum permissible release rate for the reference directly loaded PWR fuel under hypothetical accident conditions using either EPDM or Viton o-rings is tabulated in Table 4.3-2.

Correlation of Allowable Leak Rates to Air Standard

The maximum allowable leak rate for the hypothetical accident conditions is correlated with the standard leak rate by using the methodology described in Section 4.2.1.2. The results for the reference PWR fuel loaded for transport without interim storage, using either EPDM or Viton o-rings, are shown in Table 4.3-2.

4.3.4 Containment Criterion for Accident Conditions

The containment criteria of 10 CFR 71 limits the release rate in accident conditions to A_2 per week. The NAC-STC cask using metallic o-rings and the transportable storage canister are designed and tested to leaktight conditions as defined in Section 2.1 of ANSI-N14.5-1997. The allowable leak rate calculated for the EPDM or Viton o-ring configurations in the hypothetical accident conditions is much greater than that required for the normal conditions of transport. Consequently, the cask meets the regulatory containment criterion for the hypothetical accident conditions in either the metallic o-ring or non-metallic o-ring configuration.

Table 4.3-1 Allowable Release Rate Source and A_2 Values for Directly Loaded PWR Fuel: Accident Conditions Using Non-Metallic O-Rings

17 x 17 Hybrid	**Crud**	**Gas**	**Volatiles**	**Fines**	**Total**
Total Activity per Assembly (Ci)	N/C[1]	4.84E+03	1.76E+05	2.70E+05	4.51E+05
Releasable Activity per Cask (Ci)	6.68E+02	3.78E+04	9.15E+02	2.11E+02	3.96E+04
Cask Volumetric Activity (Ci/cm^3)	8.86E-05	5.01E-03	1.21E-04	2.79E-05	5.25E-03
A_2 Value (Ci)	10.80	283.25	6.08	0.13	300.26
Fraction of Activity	0.017	0.955	0.023	0.005	1.000
Fraction of Activity / A_2 (1/Ci)	0.0016	0.0034	0.0038	0.0418	0.0505
			Mixture A_2 Value (Ci)		19.79

1 Not explicitly calculated.

Table 4.3-2 Standard Leak Rate for the Accident Condition

Contents	**O-Rings**	**Vol. Activity (Ci/cm^3)**	**Leak Rate (cm^3/sec)**	
			Volumetric (L)	**Air Reference (L_R)**
Directly Loaded Reference 17 x 17 PWR Fuel	EPDM	5.2E-03	6.2E-03	1.5E-03

Table 4.3-3 Containment Parameters for Non-Metallic O-Rings in the Accident Condition

Contents	**Crud Surface Activity (Ci/cm^2)**	**Containment Free Volume (cm^3)**	**Capillary Length (cm)**	**Capillary Diameter (cm)**	**Upstream Pressure (atm)**	**Gas Temperature (K)**
Reference 17 x 17 PWR Fuel	7.3E-5	7.54E+6	0.597	3.0E-4	5.72	675

Note: 100 % of the fuel rods are postulated to fail in the accident condition.

How to Specify O-Rings

Denotes Metallic O-Ring | Tubing OD (Thirty-Seconds) | Wall Thickness (Thousandths)

U 2 3 12—03 625 SE A

Materials

1—Alloy 718	7—Stainless Steel 304
2—Stainless Steel 321	8—Stainless Steel 316
3—Aluminum	9—Stainless Steel 347
4—Copper	X—As Specified
5—Alloy 600	
6—Alloy X-750	

Metallic O-Ring OD (inches) (Thousandths)

Type

- SE—Self-energized on ID
- PF—Pressure filled
- NP—Not self-energized, not pressure filled
- SO—Self-energized on OD
- SX—Self-energized as spec.

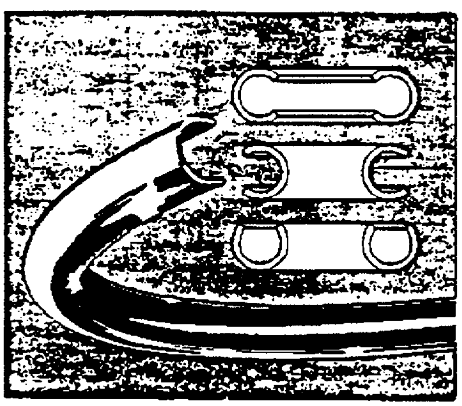
Example:

U2312-03625SEA

The above example, U2312-03625SEA, indicates a type 321 stainless steel O-Ring, 3/32" (2.38 mm) tube size, .012 (0.30 mm) wall thickness, 3.625" (92.08 mm) OD, self-energized (ID) and .001-.002" (0.03/0.05 mm) silver coating.

Coatings

A—Silver	.001/.002 (0.03/0.05)	N—None	
B—Silver	.002/.003 (0.05/0.08)	P—Lead	.001/.002 (0.03/0.05)
D—Teflon®	.001/.003 (0.03/0.08)	R—Indium	.001/.002 (0.03/0.05)
E—Teflon	.003/.004 (0.08/0.10)	T—Nickel	.001/.002 (0.03/0.05)
L—Copper	.001/.002 (0.03/0.05)	V—Gold	.0005/.001 (0.02/0.03)
X—As Specified			



Fluorocarbon Metallic C-Rings

Fluorocarbon Metallic C-Rings (designated MCR) are designed for static sealing on machinery or equipment and are available for internal pressure, external pressure, or axial pressure ID/OD applications. Because C-Rings are designed with an open side on the pressure side of the installation, the seal is self-energizing. Fluorocarbon C-Rings are offered in round or irregular shapes in a broad range of sizes from .126" (3.2 mm) OD x .032" (0.81 mm) free height to over 300" (7620 mm) OD x 2" (50.80 mm) free height. They are available in a wide variety of metal alloys and metallic or Teflon coatings. Sealing application temperature range is from cryogenic to 3,000° F. (1650°C.); pressure tolerances are from 10^{-10} torr to 100,000 psi (6,804 atm). Where customer requirements are large, the C-Ring provides the lowest unit price of any high performance seal on the market.

®Teflon is DuPont's Registered Trademark.

Helicoflex
Components Division
P.O. Box 9889
Columbia, South Carolina 29290
Telephone (803) 783-1880
FAX (803) 783-4279

4.5.2 Blended Polytetrafluoroethylene (PTFE) O-Rings

This section contains applicable technical data from a typical manufacturer of blended polytetrafluoroethylene (PTFE) o-rings. The PTFE o-rings used in the NAC-STC port covers are manufactured from virgin (unreprocessed) polytetrafluoroethylene base material filled with plastic. One product that satisfies the design requirements is the Fluoroloy K o-ring manufactured by the Furon Company, which has an operating temperature range of -450°F to +650°F.

HOW SUPPLIED

FPC (fireblock silicone foam) is provided in continuous rolls or converted into 2" or 3" wide tape with and without pressure sensitive adhesive. Custom sheets or forms are prepared to customer specifications

Cell structure	Modified closed-cell silicone foam on fiberglass
Color	White
Density	32 lbs /cu. ft.
Thickness	From .062" to .250"
Width	Up to 36"
Length	Continuous Rolls

All materials manufactured in the U.S.A.

MATERIAL ADVANTAGES

- Non combustible
- Zero flame spread
- Low smoke generation
- Non-toxic
- Lightweight, resilient, flexible, conformable
- Effective thermal, acoustical and dielectric properties
- Superior UV resistance
- Combines readily with structural or decorative materials

APPLICATIONS

- **AIRCRAFT/AEROSPACE INDUSTRY.** Firewall protection, cable and conduit insulation, engine compartment protection, interiors (sealing, walls, floors, galley trim), structural composite protection, cargo lining
- **MILITARY INDUSTRY.** High-performance fire/thermal insulation, sound transmission insulation, cable, conduit and pipe insulation, hull insulation, weaponry, packaging materials
- **MASS TRANSIT INDUSTRY.** Cable and conduit insulation, car underbodies, truck and brake areas, battery box transformer; control box, third rail protector, arc protection, firewall, noise dampening, power shoes, sealing
- **OTHER INDUSTRY APPLICATIONS.** Gasketing, welding blankets and curtains, cable, conduit and pipe insulation, electric arc isolation, appliance insulation

ANOTHER BREAKTHROUGH SILICONE PRODUCT FROM BISCO

Lightweight, flexible, elastomeric FPC silicone foam has extraordinary fire blocking properties. When valuable, crucially important equipment must withstand fire and intense heat to function, FPC assures performance. FPC, used in combination with structural composite and other engineering materials, adds fire safe capability to the most stringent industry requirements.

FPC simply will not burn. Tests show that only a 1/16" layer is needed to block a 1900°F flame for several hours.

Bisco's FPC also possesses other physical characteristics typical of silicone elastomeric foams. It is inherently non toxic and non corrosive and can be used in sensitive medical and electronic applications.

ABOUT BISCO PRODUCTS, INC.

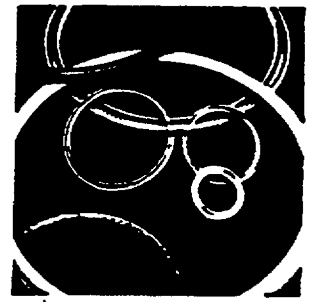
Bisco Products, Inc., one of the Brand Companies, has developed specialty high-performance products to meet many unique needs of industry.

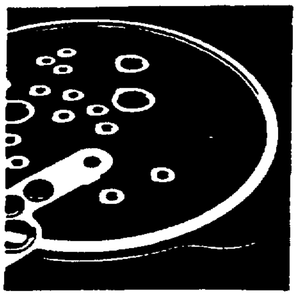
For applications requiring fire safety, or a high degree of thermal or chemical resistance, we manufacture a variety of solid or cellular elastomeric sheets, including fabric reinforced materials. These products are used by aerospace, electronics, automotive and medical industries. Other manufactured products include radiation shielding and radiation resistant materials and systems for the nuclear, power and defense industries.

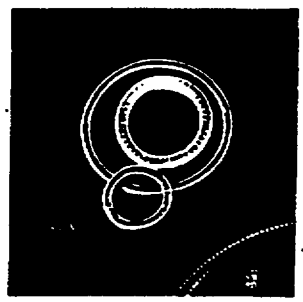
4.5.5 EPDM O-Rings

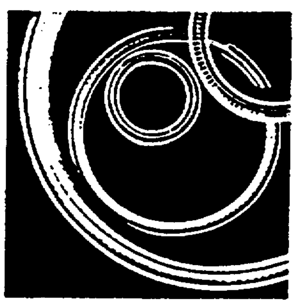
This appendix contains the manufacturer's technical bulletins for EPDM O-rings and seals.

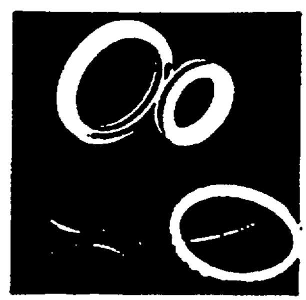
FLUOROCARBON MECHANICAL SEAL DIVISION

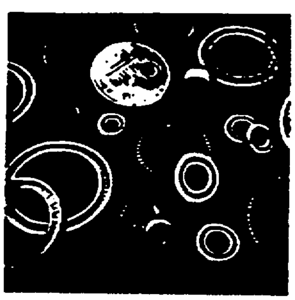








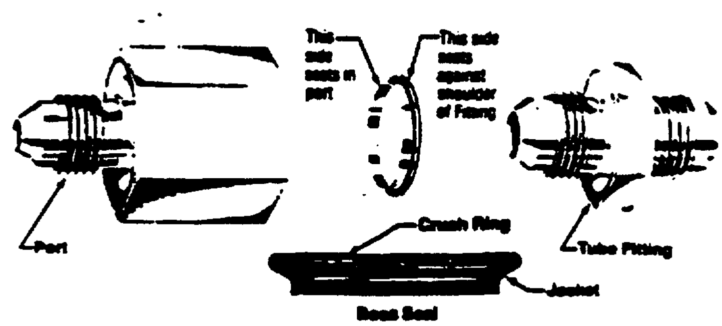


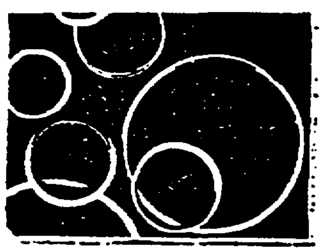


Omniseal® Design Handbook

Boss Seal

Description:
The Metal Boss Seal is a pressure loaded, positive sealing device designed for service in the AN (MS, MC) port-to-tube fitting in applications beyond the capabilities of available elastomer O-rings. The initial seal is accomplished by means of the "crush-ring" which mates the coating of the seal to the surfaces of the port and tube fitting. Once the seal is installed, it is spring loaded and pressure sensitive to insure sealing over a wide range of temperature and pressure. The advantages of both "crush" and "pressure sensitive" type seals are inherent in the design of the Boss Seal.





Ordering Information
Seal Part Number should be written as follows:

10061-XX-X-X

Basic Part No.

Material
- 0 321 Cres Jacket
  302 Cres Crush Ring
- 1 Inconel X-718 Jacket (Special Order)
  Inconel X-718 Crush Ring

	Coating	Temp. Range[1]
0	Teflon	-423° F. to 500°
1	Silver	-320° F. to 1300°
2	Gold	-423° F. to 1400°
3	Nickel	-453° F. to 1500°
4	Copper	-423° F. to 1500°
5	No Coating[2]	-453° F. to 1500°

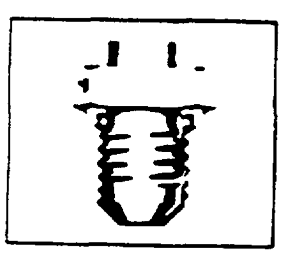
[1] Temperatures given are achieved with Cres 321/302. Inconel seals are available for higher temperatures and corrosion resistance.

[2] Uncoated seals may require polishing of mating surfaces.

Size Dash Number
- 02 AND 10050-2/MS33649-2
- 03 AND 10050-3/MS33649-3
- 04 AND 10050-4/MS33649-4
- 05 AND 10050-5/MS33649-5
- 06 AND 10050-6/MS33649-6
- 08 AND 10050-8/MS33649-8
- 10 AND 10050-10/MS33649-10
- 12 AND 10050-12/MS33649-12
- 16 AND 10050-16/MS33649-16

Larger sizes available on special order.

Advantages
1. Unlimited shelf life.
2. Temperature capability from -423° F. 1500° F
3. Designed to operate within wide pressure range.
4. Compatible with corrosive and radioactive fluids.
5. Sealing force increases with increased pressure.
6. Allows metal-to-metal seat of fittings.
7. Permits full thread engagement.
8. Non-permeable
9. Fits standard AN, MS and MC hardware
10. Seal pressure limit exceeds proof pressure of hardware

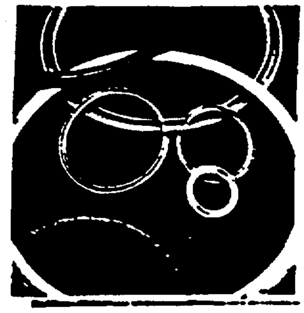


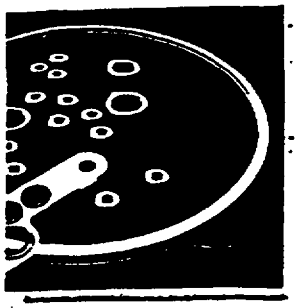
Installation
1. Lightly lubricate fitting threads, boss cavity, and top of seal with suitable lubricant except where prohibited by cleaning requirements.
2. Place seal over threads of the fitting and assemble into mating port.
3. Apply recommended torque. If no lubricant is used a higher torque may be required to bottom the fitting against the boss.

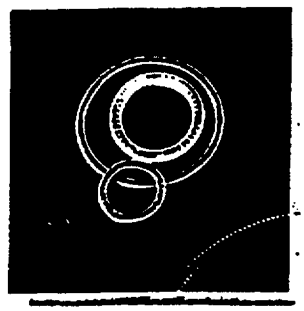
Recommended Torque

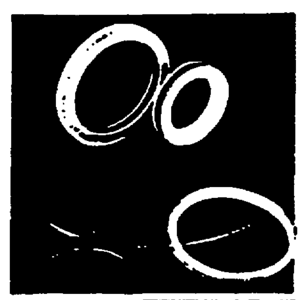
Boss Seal	Stainless Steel (in Lbs.)	Aluminum (in Lbs.)
10061 - 02	120 - 165	[illegible]
10061 - 03	200 - 250	[illegible]
10061 - 04	300 - 360	[illegible]
10061 - 05	380 - 420	[illegible]
10061 - 06	420 - 480	[illegible]
10061 - 08	600 - 720	[illegible]
10061 - 10	960 - 1080	[illegible]
10061 - 12	1380 - 1500	[illegible]
10061 - 16	1440 - 1560	[illegible]

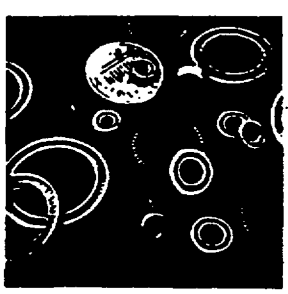
Torque values based on lubricated threads and seal surfaces.

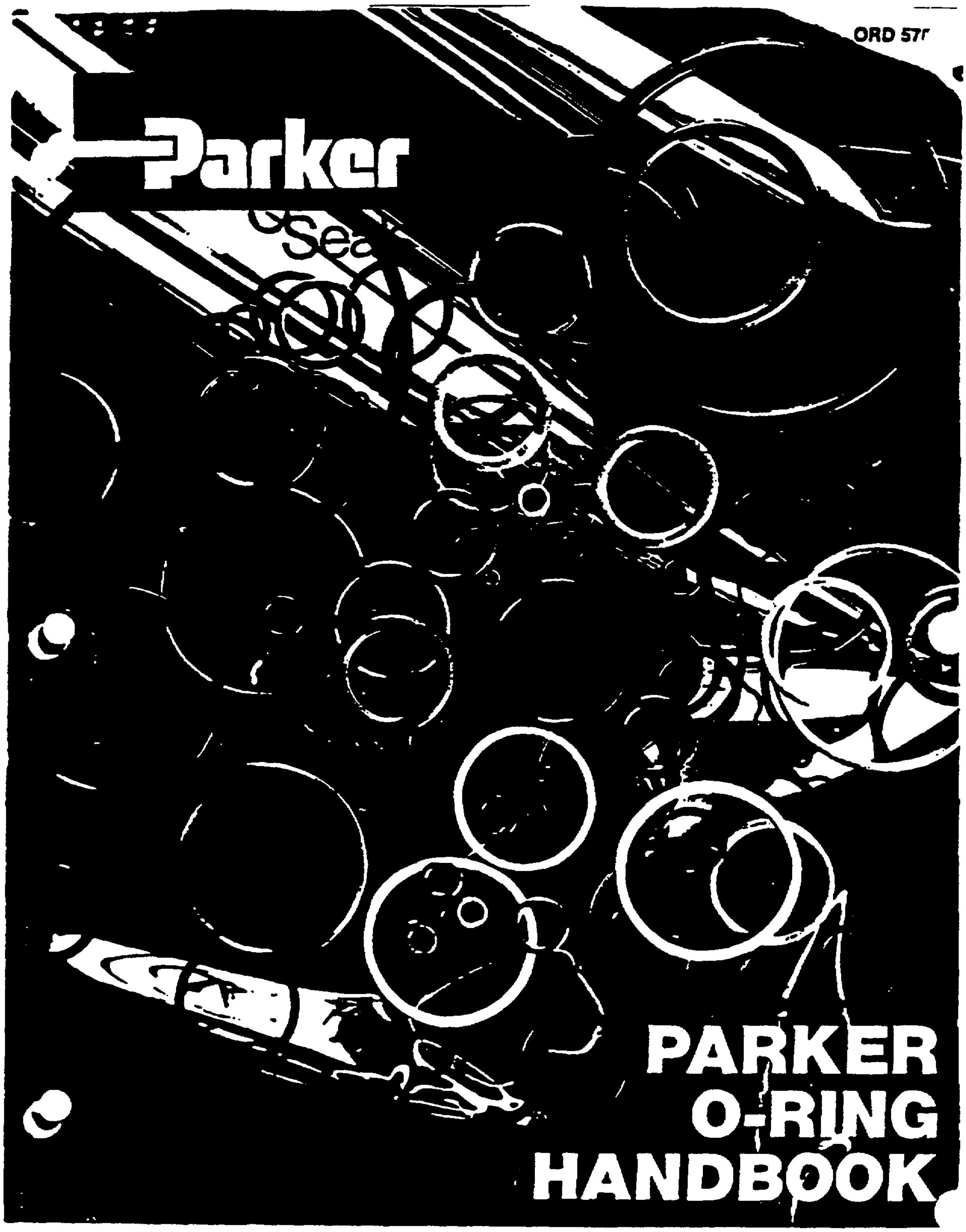








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PARKER
O-RING
HANDBOOK

ETHYLENE PROPYLENE RUBBER (EPM, EPDM)

Typical Trade Names:

Nordel..................... E. I. duPont de Nemours Co.
Royalene .... Uniroyal
Vistalon..................... Exxon Chemical Co. USA
Epsyn .... Copolymer Rubber & Chemical Corp.
Epcar .... B. F. Goodrich Co.

Ethylene propylene rubber is an elastomer prepared from ethylene and propylene monomers (ethylene propylene copolymer) and at times with a small amount of a third monomer (ethylene propylene terpolymers). It was introduced to the rubber industry in 1961 and quickly won broad acceptance in the sealing world because of its excellent resistance to Skydrol and other phosphate ester type hydraulic fluids.

Ethylene propylene has a temperature range of −65 to +300°F (−54 to +149°C) for most applications.

EP is recommended for:

Phosphate ester base hydraulic fluids (Skydrol, Fyrquel, Pydraul).
Steam (to 400°F) (204°C).
Water.
Silicone oils and greases.
Dilute acids.
Dilute alkalies.
Ketones (MEK, acetone).
Alcohols.
Automotive brake fluids.

EP is not recommended for:

Petroleum oils.
Di-ester base lubricants.

FLUOROCARBON RUBBER (FKM)

Typical Trade Names:

Fluorel 3M
Kalrez (high temp) .... E.I. duPont de Nemours Company
Kel-F .... 3M (formerly Kellogg)
Viton .... E.I duPont de Nemours Company

Fluorocarbon elastomers were first introduced in the mid 1950's. Since then they have grown to major importance in the seal industry. Due to its wide spectrum chemical compatibility and temperature range and its low compression set, fluorocarbon rubber is the most significant single elastomer development in recent history.

Its working temperature range is considered to be −15 to +400°F (−29 to +204°C), but it will take temperatures up to 600°F (316°C) for short periods of time, and Du Pont's Kalrez is normally recommended up to 500°F (260°C). On the low temperature end, Parker's compound V835-75 will seal down to −40°F (−40°C) in a static seal. Though the standard compounds have been known to seal at −65°F (−54°C) in some special static applications, the normal low temperature limit is −15°F (−26°C).

Special formulations having extra chemical resistance are also available, and new types are being developed constantly.

Fluorocarbon O-rings should be considered for seal use in aircraft, automobile and other mechanical devices requiring maximum resistance to elevated temperature and to many functional fluids.

FKM is recommended for:

Petroleum oils.
Di-ester base lubricants (MIL-L-7808, MIL-L-6085).
Silicate ester base lubricants (MLO 8200, MLO 8515, OS-45.)
Silicone fluids and greases.
Halogenated hydrocarbons (carbon tetrachloride, trichloroethylene).
Selected phosphate ester fluids.
Acids.

FKM is not recommended for:

Ketones (MEK, acetone).
Skydrol fluids.
Amines (UDMH), anhydrous ammonia.
Low molecular weight esters and ethers.
Hot hydrofluoric or chlorosulfonic acids.

FLUOROSILICONE (FSI)

Typical Trade Name:

Silastic LS. .... Dow Corning Corp.

Fluorosilicone combines the good high- and low-temperature properties of silicone with basic fuel and oil resistance. The primary uses of fluorosilicones are in fuel systems at temperatures up to 350°F (177°C), and in applications where the dry-heat resistance of silicone is required, but the seal may be exposed to petroleum oils and/or hydro-carbon fuels. In some fuels and oils, however, the high temperature limit is more conservative because temperatures approaching 350°F may degrade the fluid, producing acids which attack fluorosilicone elastomers.

On the other end of the temperature scale, fluorosilicones typically seal at temperatures as low as −100°F (−73°C). High strength type fluorosilicones are available. Certain of these exhibit much improved resistance to compression set.

ISOPRENE RUBBER-SYNTHETIC (IR)

Typical Trade Names:

Natsyn .... Goodyear Tire & Rubber Co.

Polyisoprene has the distinction of being a synthetic elastomer which has the same chemical composition as natural rubber. For a guide to its chemical and physical properties, refer to Natural Rubber below.

NATURAL RUBBER—NATURAL POLYISOPRENE (NR)

Crude natural rubber is found in the juices of many plants, including the shrub guayule, Russian dandelion, goldenrod and dozens of other shrubs, vines and trees. The principal source is the tree Hevea Brasiliensis which is native to Brazil. Petroleum oils are the greatest enemy of natural rubber compounds. The synthetics have all but completely replaced natural rubber for seal use.

NR is recommended for:

Automotive brake fluid.

NR is not recommended for:

Petroleum products.

A3-4

NEOPRENE RUBBER (CHLOROPRENE, CR)

Trade Name:

Neoprene (formerly E. I. duPont de Nemours Company)

Butaclor .... Distugil

Denka .... Denka Chem .Co.

Neoprenes are homopolymers of chloroprene (chlorobutadiene) and were among the earliest of the synthetic rubbers available to the seal manufacturers.

Neoprene can be compounded for service at temperatures of −65 to +300°F (−54 to +149°C). Most elastomers are either resistant to deterioration from exposure to petroleum lubricants or oxygen. Neoprene is unusual in having limited resistance to both. This, combined with broad temperature range and moderate cost accounts for its desirability in many seal applications.

Chloroprene is recommended for:

Refrigerants (Freons, ammonia)

High aniline point petroleum oils.

Mild acid resistance.

Silicate ester lubricants.

Chloroprene is not recommended for:

Phosphate ester fluids.

Ketones (MEK, acetone).

NITRILE OR BUNA N (NBR)

Typical Trade Names:

Chemigum .... Goodyear Tire & Rubber Co.

Paracril .... Uniroyal

Hycar .... Goodrich Chemical Co.

Krynac .... Polysar, Ltd.

Ny Syn .... Copolymer Rubber & Chem. Corp.

Nitrile, chemically, is a copolymer of butadiene and acrylonitrile. Acrylonitrile content is varied in commercial products from 18% to 48%. As the nitrile content increases, resistance to petroleum base oils and hydrocarbon fuels increases, but low temperature flexibility decreases.

Due to its excellent resistance to petroleum products, and its ability to be compounded for service over a temperature range of −65 to +275°F (−54 to +135°C), nitrile is the most widely used elastomer in the seal industry today. Most military rubber specifications for fuel and oil resistant MS and AN O-rings require nitrile base compounds. It should be mentioned, however, that to obtain good resistance to low temperature with nitrile compounding, it is almost always necessary to sacrifice some high temperature fuel and oil resistance.

Nitrile compounds are superior to most elastomers with regard to compression set or cold flow, tear and abrasion resistance. Inherently, they do not possess good resistance to ozone, sunlight or weather but this can be substantially improved through compounding. However, since ozone and weather resistance are not always built in, seals from nitrile bases should not be stored near electric motors or other equipment which may generate ozone, or in direct sunlight.

Nitrile is recommended for:

General purpose sealing.

Petroleum oils and fluids.

Cold Water.

Silicone greases and oils.

Di-ester base lubricants (MIL-L-7808).

Ethylene glycol base fluids (Hydrolubes).

Nitrile is not recommended for:

Halogenated hydrocarbons (carbon tetrachloride, trichlorethylene).

Nitro hydrocarbons (nitrobenzene, aniline).

Phosphate ester hydraulic fluids (Skydrol, Fyrquel, Pydraul).

Ketones (MEK, acetone).

Strong acids.

Ozone.

Automotive brake fluid.

POLYPHOSPHAZENE FLUOROELASTOMER (FZ)

Trade Name:

EYPEL-F .... Ethyl Corp.

EYPEL-F elastomer should effectively solve many difficult sealing problems due to its combination of physical properties, fluid resistance and temperature range.

The base polymer was developed for the U.S. Army by Firestone Tire and Rubber Company, and it has much the same temperature range (-65 to +325/350°F) and fluid resistance (especially petroleum products) as fluorosilicone elastomers but physical properties are definitely better — enough so that polyphosphazene compounds have performed adequately in dynamic and extrusion tests. Major disadvantage is its resistance to water which is only fair to poor.

POLYACRYLATE RUBBER (ACM)

Typical Trade Names:

Cyanacryl .... American Cyanamid Co.

Hycar .... B. F. Goodrich Chemical Co.

This material has outstanding resistance to petroleum fuel and oil. In addition, it possesses complete resistance to oxidation, ozone, and sunlight, combined with an ability to resist flex cracking. Compounds from this base polymer have been developed which are adaptable for continuous service in hot oil over the temperature range 0° to +350°F (−18° to +177°C). Resistance to hot air is slightly superior to nitrile polymers, but strength, compression set and water resistance are inferior to many of the other polymers.

There are several polyacrylate types available commercially, but all are polymerization products of acrylic acid esters.

Greatest usage of polyacrylate is by the automotive industry in automatic transmissions and power steering gears using Type A fluid.

A8-4

Elastomers

Elastomers

TABLE A3-8

PARKER SIZE NUMBER	CLASS III				CLASS IV				CLASS V				CLASS VI			
	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±	ID	TOL ±	W	TOL ±
2-457	13.751	.120	.271	.008	13.682	.137	.289	.009	13.612	.154	.268	.009	13.542	.170	.266	.009
2-458	14.243	.122			14.171	.139			14.099	.157			14.026	.174		
2-459	14.736	.124	.271	.008	14.661	.142	.289	.009	14.586	.160	.268	.009	14.511	.181	.266	.009
2-460	15.227	.128	.271	.008	15.150	.144	.289	.009	15.073	.163	.268	.009	14.995	.181	.266	.009
2-461	15.700	.132			15.620	.152			15.540	.171			15.460	.190		
2-462	16.192	.134			16.109	.154			16.027	.174			15.945	.193		
2-463	16.684	.141			16.599	.161			16.514	.182			16.429	.202		
2-464	17.176	.148			17.088	.169			17.001	.180			16.914	.211		
2-465	17.668	.150			17.578	.171			17.488	.193			17.398	.214		
2-466	18.160	.151			18.067	.174			17.975	.196			17.883	.218		
2-467	18.652	.158			18.557	.181			18.462	.204			18.367	.226		
2-468	19.144	.160			19.046	.183			18.949	.207			18.852	.230		
2-369	19.636	.167			19.536	.191			19.436	.215			19.336	.239		
2-470	20.620	.170			20.515	.196			20.410	.221			20.305	.248		
2-471	21.604	.179			21.494	.206			21.384	.232			21.274	.258		
2-472	22.573	.188			22.458	.215			22.344	.243			22.229	.270		
2-473	23.557	.196			23.437	.225			23.318	.254			23.198	.282		
2-474	23.541	.205			24.416	.235			24.292	.265			24.167	.295		
2-475	25.525	.213	.271	.008	25.395	.245	.289	.009	25.266	.276	.268	.009	25.136	.307	.266	.009
3-901	.182	.006	.056	.004	.181	.006	.056	.004	.180	.006	.065	.004	.179	.006	.064	.004
3-902	.235	.006	.063		.234	.006	.063		.233	.006	.062		.232	.007	.063	
3-903	.296	.006	.063		.295	.006	.063		.293	.007	.062		.292	.007	.062	
3-904	.345	.006	.071		.344	.007	.070		.342	.007	.070		.340	.008	.070	
3-905	407	.006	.071		405	007	.070		.403	.007	.070		.401	.008	.070	
3-906	461	.007	.077		.458	.007	.076		.456	.008	.076		.453	.008	.076	
3-907	.522	.009	.081		.519	.010	.080		.516	.010	.080		.514	.011	.079	
3-908	.634	.011	.086		.630	.012	.085		.627	.013	.085		.624	.014	.084	
3-909	.695	.012	.095		.691	.012	.095		.688	.013	.094		.684	.014	.094	
3-910	743	012	.095	004	739	.013	095	.004	.736	.014	.094	.004	.732	.014	.094	.004
3-911	.849	.012	.114	.005	.845	.013	114	.005	.841	.014	.113	.006	.836	.015	.112	.006
3-912	.909	.012	114		.905	.013	114		.900	.015	.113		.896	.016	.113	
3-913	.970	.014	114		.965	.015	114		.960	.016	.113		.955	.017	.113	
3-914	1.030	.014	.114		1 025	.015	114		1.020	.016	.113		1.015	.018	.113	
3-916	1 152	014	114		1 146	016	114		1.141	.017	.113		1.135	.018	.113	
3-918	1 333	017	114		1.327	019	114		1.320	.020	.113		1.313	.022	.113	
3-920	1 451	.019	116		1 444	021	116		1.437	.023	.115		1.429	.025	.114	
3 924	1 692	.020	116		1 684	022	116		1.675	.024	.115		1.667	.026	.114	
3 928	2 057	026	116		2 046	028	116		2.036	.031	.115		2.025	.033	.114	
3 932	2.300	.026	116	005	2 288	029	116	.005	2.276	.032	.115	.006	2.265	.035	114	.006

A3-31

The high temperature limits assigned to compounds in this handbook are conservative estimates of the maximum temperature for 1000 hours of continuous service in the media the compounds are most often called on to seal. Since the top limit for any compound varies with the medium, the high temperature limit for many compounds is shown as a range rather than a single figure. This range may be reduced or extended in unusual fluids.

Since some fluids decompose at a temperature lower than the maximum temperature limit of the elastomer, the temperature limits of both the seal and the fluid must be considered in determining limits for a system.

Low temperature service ratings in the past have been based on values obtained by ASTM test methods D736 and D746. The present ASTM D2000/SAE 200 specification still calls for the ASTM D746 low temperature test (ASTM D736 is obsolete). For O-rings and other compression seals, however, the TR-10 value per ASTM D1414 provides a better means of approximating the low temperature capability of an elastomer compression seal, the low temperature sealing limit being generally about 15°F below the TR-10 value. This is the formula that has been used, with a few exceptions, to establish the recommended low temperature limits for Parker Seal Group compounds in tables A3-13, B5, and B10.

This is the lowest temperature normally recommended for static seals. In dynamic use, or static applications with pulsing pressure, sealing may not be accomplished below the TR10 temperature, or 15°F higher than the low limit recommendation in the Handbook.

These recommendations are based on Parker tests. Some manufacturers use a less conservative method to arrive at low temperature recommendations, but similar compounds with the same TR10 temperature would be expected to have the same actual low temperature limit regardless of catalog recommendations.

A few degrees may sometimes be gained by increasing the squeeze on the O-ring section, while insufficient squeeze may cause O-ring leakage before the recommended low temperature limit is reached.

The low temperature limit on an O-ring seal may be compromised if the seal is previously exposed to extra high temperature or a fluid that causes it to take a set, or to a fluid that causes the seal compound to shrink. Conversely, the limit may be lowered significantly if the fluid swells the compound.

With decreasing temperature, elastomers shrink approximately ten times as much as surrounding metal parts. In a rod type assembly, whether static or dynamic, this effect causes the sealing element to hug the rod more firmly as the temperature goes down. Therefore, an O-ring may seal below the recommended low temperature limit when used as a rod type seal.

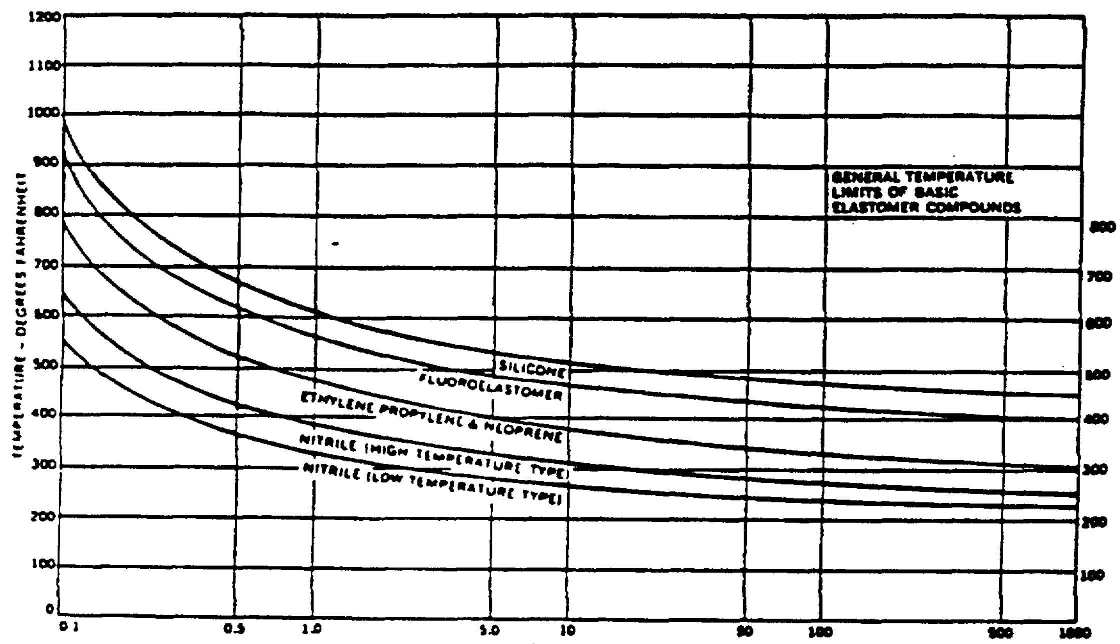

FIGURE A3-4 SEAL LIFE AT TEMPERATURE

This chart is intended only as a rough guide. It cannot be used for precise predictions of seal life. Results will vary with compound and fluid medium.

A3-36

TABLE B6

PARKER SERIES 2-XXX O-RINGS SIZE CROSS REFERENCE TABLE

1	2	3			4				5	6				7
PARKER SIZE NO. (Size Only) (a)	Size Only (Size Only) AS 568A Uniform Dash No.	NOMINAL SIZE (Inches) (REF. ONLY)			STANDARD O-RING SIZE (Units Are in Inches) Actual (b) Per AS 568A				Basic Volume Cu. In	METRIC O-RING SIZE (Units Are in Millimeters) Actual (b) Per AS 568A				PARKER SIZE NO (Size Only) (a)
		I.D.	O.D.	W.	I.D.	Tolerance + -	(REF. ONLY) W	±		I.D.	Tolerance + -	W	±	
2-175	-175	9	9 3/16	3/32	8.987	.050	.103	.003	.2379	228.27	1.27	2.62	0.08	2-175
2-176	-176	9 1/4	9 7/16	3/32	9.237	.055	.103	.003	.2445	234.62	1.40	2.62	0.08	2-176
2-177	-177	9 1/2	9 11/16	3/32	9.487	.055	.103	.003	.2510	240.97	1.40	2.62	0.08	2-177
2-178	-178	9 3/4	9 15/16	3/32	9.737	.055	.103	.003	.2576	247.32	1.40	2.62	0.08	2-178
2-201	-201	3/16	7/16	1/8	.171	.005	.139	.004	.0148	4.34	0.13	3.53	0.10	2-201
2-202	-202	1/4	1/2	1/8	.234	.005	.139	.004	.0178	5.94	0.13	3.53	0.10	2-202
2-203	-203	5/16	9/16	1/8	.296	.005	.139	.004	.0207	7.52	0.13	3.53	0.10	2-203
2-204	-204	3/8	5/8	1/8	.359	.005	.139	.004	.0237	9.12	0.13	3.53	0.10	2-204
2-205	-205	7/16	11/16	1/8	.421	.005	.139	.004	.0267	10.69	0.13	3.53	0.10	2-205
2-206	-206	1/2	3/4	1/8	.484	.005	.139	.004	.0297	12.29	0.13	3.53	0.10	2-206
2-207	-207	9/16	13/16	1/8	.546	.007	.139	.004	.0327	13.87	0.18	3.53	0.10	2-207
2-208	-208	5/8	7/8	1/8	.609	.009	.139	.004	.0357	15.47	0.23	3.53	0.10	2-208
2-209	-209	11/16	15/16	1/8	.671	.010	.139	.004	.0386	17.04	0.23	3.53	0.10	2-209
2-210	-210	3/4	1	1/8	.734	.010	.139	.004	.0416	18.64	0.25	3.53	0.10	2-210
2-211	-211	13/16	1 1/16	1/8	.796	.010	.139	.004	.0446	20.22	0.25	3.53	0.10	2-211
2-212	-212	7/8	1 1/8	1/8	.859	.010	.139	.004	.0476	21.82	0.25	3.53	0.10	2-212
2-213	-213	15/16	1 3/16	1/8	.921	.010	.139	.004	.0505	23.39	0.25	3.53	0.10	2-213
2-214	-214	1	1 1/4	1/8	.984	.010	.139	.004	.0535	24.99	0.25	3.53	0.10	2-214
2-215	-215	1 1/16	1 5/16	1/8	1.046	.010	.139	.004	.0565	26.57	0.25	3.53	0.10	2-215
2-216	-216	1 1/8	1 3/8	1/8	1.109	.012	.139	.004	.0595	28.17	0.30	3.53	0.10	2-216
2-217	-217	1 3/16	1 7/16	1/8	1.171	.012	.139	.004	.0624	29.74	0.30	3.53	0.10	2-217
2-218	-218	1 1/4	1 1/2	1/8	1.234	.012	.139	.004	.0654	31.34	0.30	3.53	0.10	2-218
2-219	-219	1 5/16	1 9/16	1/8	1.296	.012	.139	.004	.0684	32.92	0.30	3.53	0.10	2-219
2-220	-220	1 3/8	1 5/8	1/8	1.359	.012	.139	.004	.0714	34.52	0.30	3.53	0.10	2-220
2-221	-221	1 7/16	1 11/16	1/8	1.421	.012	.139	.004	.0744	36.09	0.30	3.53	0.10	2-221
2-222	-222	1 1/2	1 3/4	1/8	1.484	.015	.139	.004	.0774	37.69	0.38	3.53	0.10	2-222
2-223	-223	1 5/8	1 7/8	1/8	1.609	.015	.139	.004	.0833	40.87	0.38	3.53	0.10	2-223
2-224	-224	1 3/4	2	1/8	1.734	.015	.139	.004	.0893	44.04	0.38	3.53	0.10	2-224
2-225	-225	1 7/8	2 1/8	1/8	1.859	.018	.139	.004	.0952	47.22	0.46	3.53	0.10	2-225
2-226	-226	2	2 1/4	1/8	1.984	.018	.139	.004	.1012	50.39	0.46	3.53	0.10	2-226
2-227	-227	2 1/8	2 3/8	1/8	2.109	.018	.139	.004	.1072	53.57	0.46	3.53	0.10	2-227
2-228	-228	2 1/4	2 1/2	1/8	2.234	.020	.139	.004	.1131	56.74	0.51	3.53	0.10	2-228
2-229	-229	2 3/8	2 5/8	1/8	2.359	.020	.139	.004	.1191	59.92	0.51	3.53	0.10	2-229
2-230	-230	2 1/2	2 3/4	1/8	2.484	.020	.139	.004	.1250	63.09	0.51	3.53	0.10	2-230
2-231	-231	2 5/8	2 7/8	1/8	2.609	.020	.139	.004	.1310	66.27	0.51	3.53	0.10	2-231
2-232	-232	2 3/4	3	1/8	2.734	.024	.139	.004	.1370	69.44	0.61	3.53	0.10	2-232
2-233	-233	2 7/8	3 1/8	1/8	2.859	.024	.139	.004	.1429	72.62	0.61	3.53	0.10	2-233
2-234	-234	3	3 1/4	1/8	2.984	.024	.139	.004	.1489	75.79	0.61	3.53	0.10	2-234
2-235	-235	3 1/8	3 3/8	1/8	3.109	.024	.139	.004	.1548	78.97	0.61	3.53	0.10	2-235
2-236	-236	3 1/4	3 1/2	1/8	3.234	.024	.139	.004	.1608	82.14	0.61	3.53	0.10	2-236
2-237	-237	3 3/8	3 5/8	1/8	3.359	.024	.139	.004	.1668	85.32	0.61	3.53	0.10	2-237

(a) The rubber compound must be added when ordering by the 2- size number (i.e. 2-007 N674-70). See table 8-10 for standard Parker O-ring compounds.

(b) Table B6 provides dimensions for standard (AN shrinkage) materials ONLY. These correspond to AS568A dimensions. O-rings manufactured out of compounds with different shrinkage rates (other than AN) will produce slightly different dimensions and tolerances. See pages A3-15 through A3-31 for more information.

.103 AREA = .008392
.139 AREA = .015175

Sizes


Parker
Seals
PARKER SEAL GROUP
O-RING DIVISION
2360 PALUMBO DRIVE
P.O. BOX 11751
LEXINGTON, KY 40512
(606) 269-2351
GL-9/92

4.5.6 Viton O-Rings

This appendix provides a description of the leak testing performed using the Viton o-rings at temperatures exceeding the manufacturer's elevated temperature limit. In addition, it also contains the o-ring manufacturer's material report on the Viton material.

NAC, with the aid of an independent laboratory, performed leak testing in excess of 550°F to demonstrate the capability of Viton to perform at the elevated temperature and to determine the leak rate of the alternate port cover design at the elevated temperature. It was determined that the alternate port cover o-ring maintains its sealing capability at a temperature of 575°F after prolonged heating above 400°F. Testing was done in accordance with NAC Specifications. Two fixtures were put into a thermal test chamber. All the fittings attached to the test assemblies were checked and confirmed leaktight. The assemblies were heated in a manner that conservatively approximates the fire-transient analysis and one fixture was held at a temperature above 550°F for more than 4 hours, 37 minutes. The region inside the port cover was evacuated to below 2 psia, backfilled with helium at 0 psig, evacuated and backfilled again and then leak checked. The leak test procedure emulates the testing of the o-ring with one atmosphere of pressure acting on the o-ring during the test. The data pertinent to the test is:

	Test Assembly 16	Test Assembly 64	Fire-Transient
Time Above 400°F	~6:32 hours	~5:52 hours	4:37 hours
Time Above 550°F	~5:05 hours	~4:25 hours	0 hours
Maximum Seal Temperature	~575°F	~575°F	547°F

The test temperature of 550°F was selected because it approximates the maximum calculated o-ring temperature in the fire-transient analysis. The duration was selected because it is the calculated duration that the o-ring is above the manufacturer's maximum recommended o-ring temperature of 400°F. This results in a conservative test due to the slower heat-up rate of the oven compared to the heat-up rate of the port cover in the fire-transient analysis.

Each test assembly was leak checked after the temperature test, while at a temperature of approximately 575°F. The measured leak rate for each of the assemblies was less than 4.0×10^{-8} atm-cc/sec. In conclusion, the Viton o-rings can provide a leaktight seal, in accordance with ANSI N14.5-1997, at an elevated temperature.

Sep-17-99 03:35P P.01

Parker Seals

Software Version: 2.0 9/17/99

Customer Identification

Company:	NAC International		
Contact:	George Carver		
Project Name:			
Address:			
City:		Zip Code:	
State:			
Telephone No.:	770-447-1797 fax		
Date/Time:	9-17-1999 15:27		

Ordering Specifications

Application:	O-ring Only
Compound Number:	V0835-75
Size:	

Compound Information

Search Parameter

Material Selection Method:	Compound Search
Contained Media:	
Desired Temperature Range	
High:	
Low:	

Selected Material Information

Durometer (Shore A):	75
Polymer:	Fluorocarbon GLT - LOWTEMP COMPOUND.
Temperature	
Normal High:	400 °F
Extended High:	400 °F
Normal Low:	-40 °F
Color:	Black
Static Application Only:	No
Military Spec.:	MIL-R-83485
AMS NAS Spec.:	None
SAE/ASTM Spec.:	None

Seal Size Information

Sizing Selection Method:	Known: O-ring P/N. Search for: O-ring dimensions.

Sep-17-99 03:35P

P.02

Compound Data Sheet
O-Ring Division United States

MATERIAL REPORT

REPORT NUMBER: KJ0835
DATE: 10/10/89

TITLE: Test of Parker Compound V0835-75 to MIL-R-83485, Type I.

PURPOSE: To determine if V0835-75 meets MIL-R-83485, Type I.

CONCLUSION: V0835-75 meets the above specification.

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Kentucky 40509
(606) 269-2351

Sep-17-99 03:35P

P.03

REPORT DATA

Report Number: KJ0835

ORIGINAL	MIL-R-83485 TYPE 1, O-RINGS & COMPRESSION SEALS	V0835-75 ACTUAL VALUES
Specific Gravity	As determined	1.75
Hardness points	75 ± 5	78
Tensile Strength, psi. min.	1600	1708
Elongation, % min.	120	180
Temperature Retraction, 10% (TR-10), °F, max.	-20	-22
AFTER AIR AGING, 70 HRS.@ 75° + 5°F, Compression Set		
% of original deflection, max.	25	-- (14)
AFTER AGING, 70 HRS. @ 75°F IN TT-S-735, TYPE III		
Hardness Change, pts.	+5	77 (-1)
Tensile Strength decrease, %, max.	30	1662 (-3)
Elongation decrease, %, max.	20	165 (-8)
Volume change, %, max.	1 to 10	-- (+2)
AFTER AIR AGING, 70 HRS. @ 528° ± 5°F		
Hardness change, pts.	+5	78 (0)
Tensile Strength decrease, % max.	35	1136 (-33)
Elongation decrease, %, max.	10	235 (+31)
Weight loss, %, max.	12	-- (-7)
AFTER AIR AGING, 166 HRS @ 347° ± 5°F, COMPRESSION SET		
% of original deflection, max.	25	-- (15)
18 hrs. cooling		-- (24)
AFTER AIR AGING, 22 HRS @ 392° ± 5°F, COMPRESSION SET		
% of original deflection, max.	20	-- (11)

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P.04

AFTER AGING, 70 HRS. @ 347°MIL-R-83485 ±5°F in AMS-3021	MIL-R-83485 TYPE 1, O-RINGS % COMPRESSION SEALS	V0835-75 ACTUAL VALUES
Hardness change, pts	+0, -15	73
Tensile Strength decrease, %, max.	35	1406 (-18)
Elongation decrease, %, max.	20	171 (-5)
Volume change, %	1 to 20	-- (+15)
Compression set, % of original deflection, max.	10	-- 7
18 hr. cooling		-- 9

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5.0 SHIELDING EVALUATION

The NAC-STC uses an optimized multiwall design to provide the most efficient shielding arrangement possible, and to comply with 10 CFR 71 limits. This chapter provides a description of the NAC-STC shield design, design basis contents, and the conservative shielding analyses used to determine the transport dose rates.

The NAC-STC is designed to safely transport intact spent fuel assemblies in two configurations: directly loaded and canistered. In the directly loaded configuration, standard PWR fuel assemblies are placed directly into a fuel basket installed in the cask cavity. In the canistered configuration, a sealed transportable storage canister loaded with fuel assemblies is placed in an empty cask cavity with top and bottom spacers. In the directly loaded configuration, the NAC-STC can transport up to 26 standard PWR fuel assemblies. In the canistered configuration, the NAC-STC can transport up to 36 Yankee Class fuel assemblies.

For directly loaded fuel, the shielding evaluation considers reference fuel assemblies in 14 x 14, 15 x 15, 16 x 16 and 17 x 17 array sizes. The reference fuel assemblies have parameters selected from all of the fuel assemblies of the same array size to maximize the shielding source terms. The design basis fuel for the canistered configuration is the Yankee Class, Combustion Engineering, Type A, 16 x 16 PWR fuel assembly.

The NAC-STC can also safely transport Greater Than Class C (GTCC) waste in a canistered configuration. The GTCC waste, consisting of activated steel, is placed in a container (see Figure 5.1-4) that is the same size as a Yankee Class fuel assembly. Up to 24 GTCC containers can be loaded into the GTCC canister.

The NAC-STC is assigned a nominal Transport Index for shielding of 21 (TI = 21) based on the requirement of 10 CFR 71.4 and the analysis of Section 5.1.4. The maximum dose rate at 1 meter from the NAC-STC in normal conditions of transport is 20.3 mrem per hour, based on the directly loaded reference fuel. The actual measured dose rate is expected to be less.

The shielding evaluation for directly loaded fuel, canistered fuel and GTCC waste demonstrates compliance with 10 CFR 71 limits. The dose rates for the canistered Yankee Class fuel and fuel and GTCC waste dose rates are shown to be significantly less than those for the directly loaded fuel configuration for both normal and accident conditions.

The shielding evaluation of the directly loaded configuration is performed using the SAS2H sequence (Hermann, 1995) of the SCALE-4.3 package for the PC (ORNL, 1995). This sequence uses the computer code ORIGEN-S (Hermann, 1989) to calculate the source terms. The MCBEND (AEA Technology, 2000) computer code is used to calculate the cask dose rates for normal transport and hypothetical accident conditions. The shielding analyses show that the dose rates are below regulatory limits.

The shielding evaluation of the Yankee Class canister fuel and GTCC waste is performed using SCALE 4.3 for the PC (ORNL, 1995). This code uses SAS2H (Herman, 1995) to calculate source terms. 1D shielding evaluations were performed using SAS1 (Knight, 1995). The shielding analyses show that the dose rates are well below the regulatory limits stated in 10 CFR 71 and are well below the dose rates reported for the directly loaded fuel.

Directly Loaded Fuel

The directly loaded basket construction is based on a tube and disk design. PWR fuel is loaded into 26 fuel tubes fabricated from Type 304 stainless steel sheets. BORAL neutron absorber is encased in stainless steel on the outside face of the fuel tube. Twenty 5/8-inch thick aluminum disks are spaced between thirty-three 1/2-inch thick Type 17-4 PH stainless steel support disks to provide heat transfer. Radial shielding of PWR fuel in the directly loaded basket is provided by the multi-wall design of the NAC-STC cask body. Axial shielding is provided by the cask body closure lids and end forgings and the impact limiters.

Canistered Fuel and GTCC Waste

The canister containing Yankee Class fuel or GTCC waste is placed in the NAC-STC cavity with top and bottom spacers. The placement of the canister between the top and bottom spacers effectively precludes the source regions from streaming through areas above and below the neutron shield and tapered regions of the lead. In addition to the radial and axial shielding provided by the cask body and lids, radial and axial shielding is provided by the canister 5/8 inch shell, the 8 inches of stainless steel from the canister lids and 1 inch of steel from the canister bottom.

The Yankee-MPC fuel basket is of the same design as the steel/aluminum directly loaded basket. It has a shorter overall length to accommodate the dimensions of the design basis Yankee Class fuel, and a smaller diameter to accommodate the inside dimension of the canister. Consistent with these smaller dimensions, the Yankee-MPC basket also has fewer support plates and heat transfer disks than the directly loaded basket.

The Yankee-MPC GTCC basket is a simplified tube and disk design. The steel tubes holding the GTCC waste containers are surrounded by a 2.5-inch steel basket support wall, and are held in place by steel support disks. Aluminum heat transfer disks are not used.

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5.1 Discussion and Results

The radiation protection provided by the NAC-STC is in the form of solid multi-walled shielding materials, which totally surround the fuel. These shielding materials include steel and lead for gamma shielding and a borated polymer (NS-4-FR) for neutron shielding. The multi-walled arrangement of steel and lead in the NAC-STC provides optimal weight for gamma attenuation. The NS-4-FR neutron shielding material has a hydrogen density close to that of water and serves to moderate fast neutrons, which are then captured in the boron. Boron capture in the neutron shield minimizes the contribution of secondary capture gammas to surface dose rates.

The NAC-STC uses a multi-walled arrangement for both radial and axial shields. The arrangement of the radial gamma shielding in the cask body is a 1.5-inch thick stainless steel inner shell and a 2.65-inch thick stainless steel outer shell with a 3.70-inch thick lead filled annulus between them. The radial neutron shield is arranged around the outer steel shell with a 5.5-inch thick NS-4-FR layer, covered by a 0.25-inch (6 mm) thick neutron shield shell. The bottom of the cask contains a steel/NS-4-FR/steel shield arrangement with the two stainless steel components providing 11.65 inches of gamma shielding and 2 inches of NS-4-FR neutron shielding. The top of the cask has shields in the form of two closure lids. The inner lid also has a steel/NS-4-FR/steel arrangement with 6.0 inches of steel below 2 inches of NS-4-FR and 1.0 inch of steel above it. The outer lid is a 5.25-inch thick steel disk.

5.1.1 Design Criteria

The shielding design criteria for the NAC-STC meets the requirements of 10 CFR 71 and IAEA Safety Standard Series No. ST-1. For normal conditions, the dose rate limits specified in 10 CFR 71.47 and paragraph 572 of IAEA Safety Standard Series No. ST-1 for consignments under exclusive use are: 1,000 mrem/hour on the surface of the enclosed package, 200 mrem/hour on the outer surfaces of transport vehicle and 10 mrem/hour at 2 meters from the vertical planes represented by the outer lateral surfaces of the transport vehicle. The cask surface dose rate is less than 200 mrem/hour, except at the gap between the neutron shield and the upper impact limiter and at the rotation trunnions, where the maximum dose rate is 366 mrem/hr. The maximum dose rate at the personnel barrier, which is the accessible surface of the package, adjacent to the gap between the neutron shield and upper impact limiter, is significantly less than 200 mrem/hr. Note: The cask tie-down structure that is present at this location is conservatively not considered. The 10 mrem/hr criterion has also been met at all locations 2 meters from the railcar. Under hypothetical accident conditions, 10 CFR 71.51 and IAEA Safety Standard Series

No. ST-1, paragraph 656, specify a dose rate limit of 1,000 mrem/hour at 1 meter from the surface of the cask. This criterion has also been met at all locations.

The accessible surface of the package is defined as a personnel barrier that will be on the same plane as the outer radial surface of the top half of the impact limiters. The personnel barrier will attach to the edge of the railcar between the impact limiters. The personnel barrier location is shown in Drawing 423-901.

5.1.2 Design Basis Fuel

The NAC-STC has two configurations for transport of design basis fuel: directly loaded and canistered. The design basis fuel for the directly loaded configuration is described in Section 5.1.2.1. The second configuration is for canistered Yankee Class fuel and GTCC waste. The design basis fuel for shielding for this configuration is described in Section 5.1.2.2.

5.1.2.1 Design Basis Directly Loaded Fuel

The NAC-STC can transport up to 26 directly loaded, intact PWR fuel assemblies over a range of burnups, initial ^{235}U enrichments, and minimum allowable cool times. Reference fuel assemblies have been developed and analyzed to envelope PWR fuel for 14×14, 15×15, 16×16, and 17×17 array sizes. These assemblies are constructed by surveying assembly data for assemblies less than 165 inches in length (the length of the STC cavity) and using bounding fuel parameters to maximize fuel mass (MTU) and hardware source terms. Decay heats and dose rates have been calculated for a finite range of burnups, initial ^{235}U enrichments, and cool times to generate an allowable loading table, or minimum cool timetable. Adherence to the cool timetable ensures that heat load and dose rate limits will not be exceeded.

Three-dimensional dose rates are calculated using a response function methodology. Each of the four fuel assembly array sizes is analyzed over a range of source regions and source types with unit source in each relevant energy group. Source types considered are fuel neutron, fuel gamma, fuel secondary gamma (n-gamma), in-core fuel hardware (grid spacers, steel guide tubes, etc.), plenum, and end fitting hardware. These sources are analyzed in a finite number of energy groups with a unit source in each group. The scalar product of source term and response function allows for the creation of large arrays of dose rate results, whether they are for a single detector, or the maximum or average over a detector surface. In this analysis, detector maximum responses have been used exclusively to generate minimum cool timetables.

5.1.2.2 Design Basis Yankee Class Canistered Fuel and GTCC Waste

The design basis fuel for the Yankee Class canistered configuration for shielding purposes is the Combustion Engineering (CE), Type A, 16 x 16 PWR assembly with an initial enrichment of 3.7 wt % ^{235}U, a uranium mass of 239.4 kilograms, a burnup of 36,000 MWD/MTU and 8.0-year cooling time. To meet maximum cask decay heat limits, an 8.1-year cool time is required. The 8.0-year cooled source terms are conservatively used as the shielding design basis. The dose rates resulting from this assembly are higher than those of the other Yankee Class fuels: CE Type B, and Westinghouse, Exxon, and United Nuclear Type A and B fuel assemblies. The design basis Yankee Class fuel characteristics are given in Table 5.1-1. The design basis Yankee Class fuel physical parameters are presented in Table 5.1-2. The design basis canister fuel assembly source terms are presented in Table 5.1-3, and a sketch of the fuel assembly is shown in Figure 5.1-3.

Source terms and dose rate evaluations concluded that for the Westinghouse, United Nuclear, and CE Yankee Class fuel assemblies at 32,000 MWD/MTU require minimum cooling times of 19, 11 and 7 years, respectively. The minimum enrichments for these assemblies are 4.94, 4.0 and 3.5 wt %, respectively. Exxon fuel, with a burnup of 36,000 MWD/MTU and a minimum initial enrichment of 3.5 wt %, requires a minimum cooling time of 16 years for assemblies containing steel hardware in the active fuel region, and 9 years for assemblies with Zircaloy hardware. Combustion Engineering fuel with an initial enrichment of 3.5 wt % is limited to 15,000 MWD/MTU at 6.8 years cooling time.

The NAC-STC can also safely transport Yankee GTCC waste. The GTCC waste, consisting of activated steel, is placed in a container (see Drawing 455-888 and Figure 5.1-4) that is the same size as a Yankee Class fuel assembly. Up to 24 GTCC containers can be loaded into the GTCC canister basket. The GTCC canister is loaded in the NAC-STC for transport.

The design basis gamma source for Yankee GTCC waste is determined from dose rate measurements and chemical assay of the GTCC waste. This gamma source is primarily due to the activation of the core baffle from 30 years of neutron flux exposure and to a lesser extent from surface contamination. The design basis source term for the GTCC waste canister is 9.493 x 10^{15} photon/s, which is equivalent to 125,000 curies of ^{60}Co. The design basis thermal output is 1.93 kW.

The transportable storage canister may contain one or more Reconfigured Fuel Assemblies. The Reconfigured Fuel Assembly is designed to confine Yankee Class spent fuel rods, or portions thereof, which have been classified as failed. Each assembly can accommodate up to a total of 64 fuel rods. Due to the low number of rods, the reconfigured assembly fuel mass is significantly less than the fuel mass contained in the design basis fuel assemblies. Because source term (neutron and gamma) is directly proportional to fuel mass, for a given burnup, the reconfigured assembly source term is bounded by that of the design basis Yankee Class fuel assemblies. The lower source term of the 64 rod reconfigured assembly more than offsets any reduced self shielding associated with its lower mass. In addition, each Reconfigured Fuel Assembly fuel rod is placed within a steel enveloping rod. Consequently, a rigorous shielding analysis is not required for the Reconfigured Fuel Assembly.

5.1.3 Shielding Materials

The shielding materials are selected and arranged to minimize cask weight while maintaining overall shield effectiveness. Lead and steel are chosen as effective gamma radiation shields, and NS-4-FR is provided to efficiently moderate and absorb the neutron radiation, while minimizing the generation of secondary gamma radiation.

5.1.4 Results

For both the directly loaded and the canistered transport configurations, this section demonstrates that the NAC-STC satisfies the regulatory criteria of 10 CFR 71.47 and paragraph 572 of IAEA Safety Standard Series No. ST-1 under normal transport conditions; and 10 CFR 71.51(a) and paragraph 656 of IAEA Safety Standard Series No. ST-1 for hypothetical accident conditions. Specifically, for an exclusive use shipment in an enclosed transport vehicle, the dose rates remain less than 1,000 mrem/hour on the surface of the package, less than 200 mrem/hour at all locations on the surface of the personnel barrier and less than 10 mrem/hour at all locations 2 meters from the edge of the railcar (any point 2 meters from the vertical planes projected from the outer edges of the conveyance). Also, under hypothetical accident conditions, the dose rate is less than 1,000 mrem/hr at 1-meter from the surface of the package. Therefore, the NAC-STC satisfies the shielding criteria of 10 CFR 71 and IAEA Safety Standard Series No. ST-1.

5.1.4.1 Results of the Shielding Evaluation for Directly Loaded Fuel

The maximum dose rates calculated for the normal transport conditions are shown in Table 5.1-4, with locations of the maximum dose rates shown in Figure 5.1-2. Cask surface dose rates do not exceed the regulatory limit for a closed transport vehicle of 1,000 mrem/hour at the surface of the package. The dose rates at 2 meters from the railcar comply with the 10 mrem/hour regulatory limit.

The maximum normal conditions surface dose rate at the cask radial midplane is 41 mrem/hour. The highest dose rate, occurring on the surface of the cask at the gap between the radial neutron shield and the upper impact limiter, is 366.4 mrem/hour. All cask surface dose rates are much less than 1,000 mrem/hour. Ducting of neutrons through the copper/stainless steel fins is considered in Section 5.4.1.1. The results of the ducting evaluation show that this phenomenon has a very small effect on the total cask dose rate. Azimuthal variations in the calculated dose rate are considered in the explicit heat fin and neutron shield model. The neutron dose rate increase resulting from the ducting is offset by the reduction of the gamma dose rate resulting from the additional shielding provided by the fins.

Table 5.1-5 provides accident dose rates that could occur in the event of the loss of all gaseous elements in the neutron shield combined with radial and axial lead slumps due to cask side and end drops. Although the neutron shield material exceeds its safe operating temperature limits in the fire accident, a complete loss of neutron shielding is not credible for the NAC-STC. Some of the neutron shielding capability may be lost, however, as a result of the fire accident. Therefore, the accident shielding calculations conservatively assume a complete loss of gaseous elements in the neutron shielding. In the event of a cask end drop, it is possible for the lead gamma shielding to slump and fill the annular gap (if one exists) created by the cooling of the lead after fabrication. For worst case conditions, this accident could create a 2.35-inch gap at the top of the lead annulus. If the cask is subject to a side drop, the lead gamma shielding could slump and create a void on the upper side of the cask. An evaluation of this accident shows the lead thickness may be reduced by a maximum of 0.928-inch. The dose rates shown in Table 5.1-5 show that neither the loss of the neutron shielding nor the slumping of the lead will result in a dose rate that exceeds the hypothetical accident dose rate limit of 1,000 mrem/hour at 1 meter from the cask surface.

Therefore, the NAC-STC fulfills the design criteria of Chapter 1 in that under normal transport conditions, the maximum dose rates are less than 1,000 mrem/hour on the surface of the package, less than 200 mrem/hour at all locations at the surface of the personnel barrier, and less than 10 mrem/hour at all locations 2 meters from the personnel barrier. The cask also satisfies the hypothetical accident criteria of 1,000 mrem/hour at 1 meter from the cask surface.

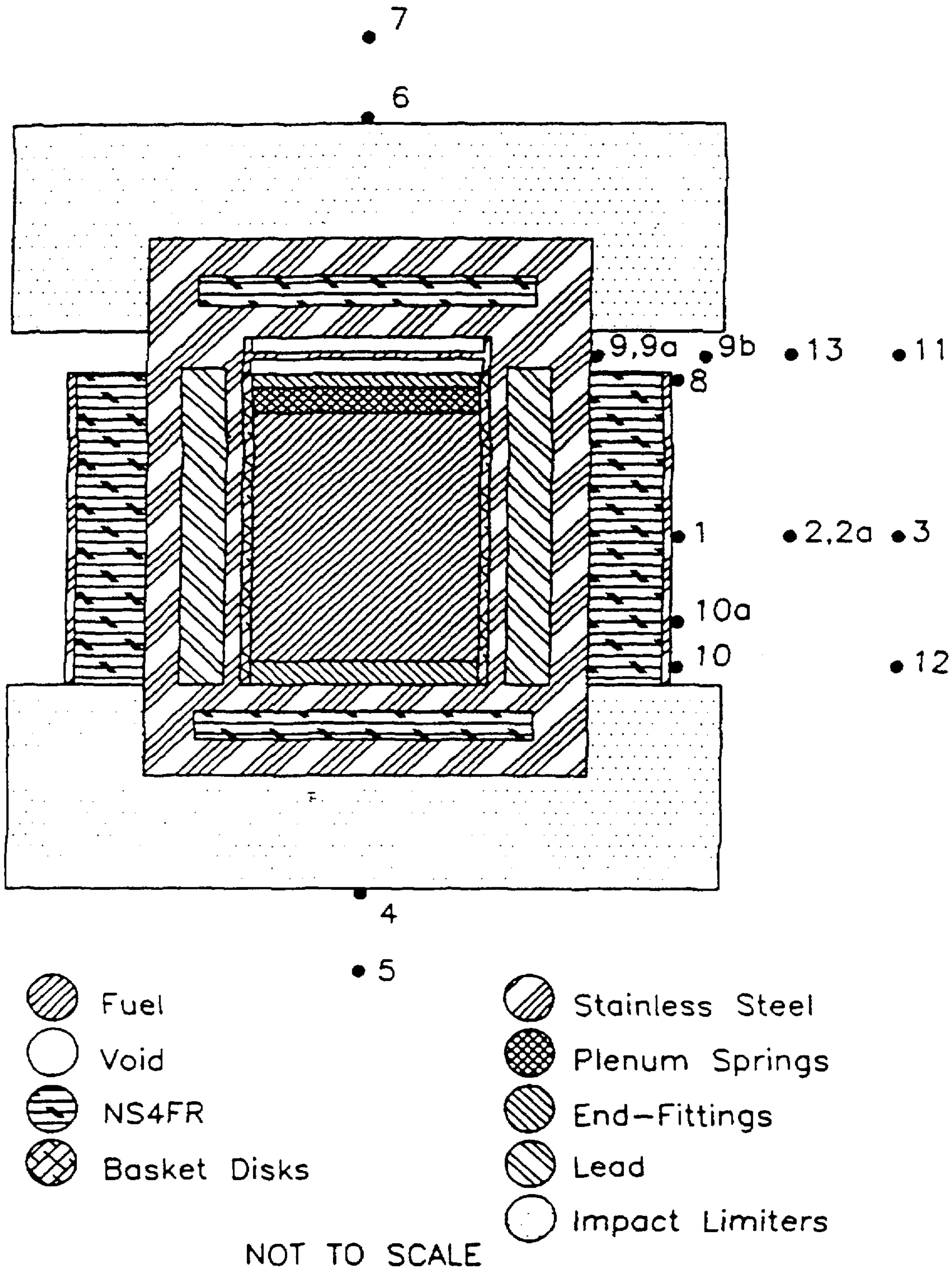
5.1.4.2 Shielding Evaluation for Yankee Class Canistered Fuel and GTCC Waste

A 1-D radial and axial shielding analysis was performed for both the canistered Yankee Class fuel and GTCC waste under normal and hypothetical accident conditions. The dose rates for canistered fuel (Combustion Engineering, 36,000 MWD/MTU, 8-year cooled) are provided in Tables 5.1-7 and 5.1-8. These dose rates are provided in Tables 5.1-9 and 5.1-10 for GTCC waste. Under normal conditions, the canister is positioned in the cavity with top and bottom spacers, and the impact limiters are in place on the cask. Under accident conditions (i.e., 30-foot drop and fire accident), the radial midplane results assume loss of neutron shielding. A complete loss of neutron shielding is not credible for the NAC-STC. However, because of the elevated fire accident temperatures, the neutron shields exceed their safe operating limits and some neutron shielding capability may be lost. Also, in the axial models, it is assumed that the cavity spacers are crushed, the impact limiters are lost, and the canister is positioned at either the top or the bottom of the cavity.

The maximum calculated dose at the surface of the cask centerline when loaded with canistered Yankee Class fuel in normal conditions of transport is 10.25 mrem/hour. This is much less than the 41 mrem/hour for the same location with the directly loaded reference fuel in the cask. In the accident condition involving a loss of neutron shielding and lead slump, a maximum dose rate of 262.76 mrem/hour is calculated at 1 meter from the radial midplane of the NAC-STC. This is also much less than the directly loaded reference fuel accident dose rates shown in Table 5.1-5 and is well below 10 CFR 71 regulatory limits.

The maximum calculated dose at the surface of the cask centerline when loaded with GTCC waste under normal conditions of transport is 7.03 mrem/hour. This is much less than the 41 mrem/hour calculated for the same location with the directly loaded design basis fuel in the cask. In the accident condition, a maximum dose rate of 55.77 mrem/hour is calculated at 1 meter from the radial surface of the NAC-STC. This is also much less than the directly loaded design basis fuel accident dose rates shown in Table 5.1-5 and is well below 10 CFR 71 regulatory limits.

Figure 5.1-1 Detector Locations



Detector locations are described in Tables 5.1-5, 5.1-6 and 5.1-7.

Figure 5.1-2 Maximum Dose Rate Locations for the Three-Dimensional Directly Loaded Fuel Analysis in Normal Conditions

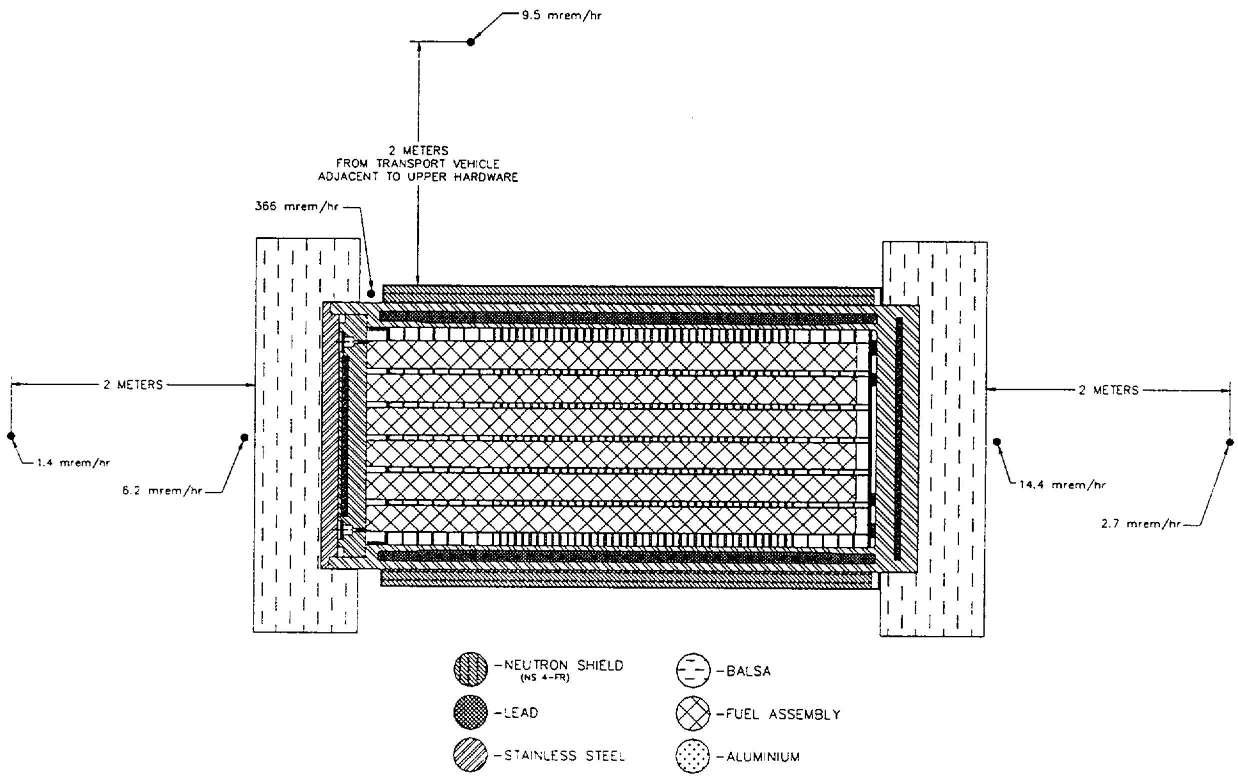


Figure 5.1-3 Design Basis "Yankee Class" Combustion Engineering Fuel Assembly

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.1-4 GTCC Waste Container

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Table 5.1-1 Type, Form, Quantity and Potential Sources of the Fuel Used for Design Basis Directly Loaded and Canistered Fuel

	Design Basis Directly Loaded Fuel	Design Basis Canistered Fuel
Fuel Type	• PWR, 14 x 14, 15 x 15, 16 x 16 and 17 x 17 • Array-dependent maximum uranium mass • Variable minimum initial ^{235}U enrichment • 45,000 MWD/MTU maximum burnup[2] • 0.85 kW per assembly maximum decay heat, 22.1 kW per cask for 26 assemblies • Variable minimum cool time	• Yankee Class PWR Combustion Engineering, 16x16 Type A • 239.4 kg maximum uranium mass • 3.7 wt % maximum initial ^{235}U enrichment[1] • 36,000 MWD/MTU maximum burnup[3] • 0.347 kW per assembly maximum decay heat, 12.5 kW per cask for 36 assemblies • 8.1 years (or more) decay time after reactor discharge[3]
Fuel form	Intact assemblies	Intact assemblies
Quantity	26 design basis fuel assemblies	36 design basis fuel assemblies
Heat Load	22.1 kilowatts, thermal per cask	12.5 kilowatts, thermal per cask
Sources of Fuel	Commercial PWR nuclear power reactors	Commercial Yankee Class nuclear power reactors

1. 3.7 wt % ^{235}U is used for the 36,000 MWD/MTU fuel assembly shielding source terms. It yields higher source terms than the 3.9 wt % used in the criticality analysis.
2. Source term and shielding evaluations were conservatively performed at burnups up to 60,000 MWD/MTU.
3. Yankee Class Westinghouse, United Nuclear and Combustion Engineering (3.5 wt % ^{235}U) fuel assemblies with burnups up to 32,000 MWD/MTU require minimum cool times of 19, 11 and 7 years, respectively. Exxon assemblies with burnups up to 36,000 MWD/MTU require a minimum cool time of 16 years for assemblies containing steel hardware in the active fuel region and 9 years for assemblies with Zircaloy hardware.

Table 5.1-2 Design Basis Yankee-MPC Canistered Fuel - Physical Parameters

PARAMETER	VALUE
Assembly Rod Array	16 x 16
Assembly Weight, lb	776
Assembly Length, in	111.79
Active Fuel Length, in	91
No. of Fuel Rods	231
Rod Pitch, in	0.472
Cladding Material	Zircaloy-4
Rod Diameter, in	0.365
Cladding Thickness, in	0.024
Pellet Diameter, in	0.3105
Pellet Material	UO_2 (sintered)
Maximum Fuel Rod Pressure, psig	315
Theoretical Density, percent	95
Maximum Initial Enrichment, wt % ^{235}U	3.9
Design Basis Burnup, MWD/MTU	36,000
Weight of U, kg (typical)	239.4
Weight of UO_2, kg (typical)	271.6
Upper End-Fitting, kg/assembly	5.5
Lower End-Fitting, kg/assembly	5.18
Upper Plenum Springs, kg/assembly	0.762
Upper Plenum Grid Grid/assembly	0.590
Lower Plenum Grid/assembly	NA

Table 5.1-3 Nuclear and Thermal Parameters of the Design Basis Yankee Class Fuels and GTCC Waste

Parameter	Fuel	GTCC[1]
No. of Fuel Assemblies or Containers	36	24
Burnup, MWD/MTU	36,000	N/A
Cooling Time, years	8	N/A
Decay Heat, kW	12.5	2.5
Gamma Source, MeV/s , photons/s	2.856×10^{16} 6.423×10^{16}	1.16×10^{16} 9.493×10^{15}
Neutron Source	2.415×10^{9}	N/A
Core Grids, neutrons/s	0.0	N/A
Upper end-fitting ^{60}Co Source, photons/s	8.330×10^{13}	N/A
Lower end-fitting ^{60}Co Source, photons/s	7.876×10^{13}	N/A
Upper Plenum Hardware ^{60}Co Source, photons/s	2.309×10^{13}	N/A
Lower Plenum ^{60}Co Source, photons/s Hardware	5.242×10^{13}	N/A

1 Includes depleted Sb-Be source vanes and core baffle steel.

Table 5.1-4 Directly Loaded Fuel Maximum Dose Rates for Normal Conditions of Transport

Detector	Source	Surface		2 meter[1]	
		mrem/hr	**RSD**	**mrem/hr[2]**	**RSD**
Top Axial	Neutron	0.5	0.2%	0.2	0.2%
	Gamma	5.7	0.4%	1.2	0.4%
	Total	6.2	0.3%	1.4	0.3%
Radial	Neutron	152.2	0.3%	2.9	0.3%
	Gamma	214.2	0.3%	6.6	0.4%
	Total	366.4	0.2%	9.5	0.3%
Bottom Axial	Neutron	4.0	0.3%	0.8	0.2%
	Gamma	10.4	0.6%	1.9	0.9%
	Total	14.4	0.4%	2.7	0.7%

1. Dose rates are rounded to the indicated precision.
2. Dose rates at 2 meter locations radially are 2 meters from the railcar. Dose rates at 2 meter locations axially are measured from the ends of the impact limiters.

Table 5.1-5 Directly Loaded Fuel Maximum Dose Rates for Hypothetical Accident Conditions

Detector	Source	Surface[1]		1 meter[1]	
		mrem/hr[2]	RSD	mrem/hr[2]	RSD
Top Axial	Neutron	31.3	0.5%	23.4	0.9%
	Gamma	20.5	0.8%	10.9	2.7%
	Total	51.8	0.5%	34.3	1.1%
Radial[3]	Neutron	1880	0.2%	685	0.2%
	Gamma	47	8.6%	23	7.3%
	Total	1927	0.3%	708	0.3%
Bottom Axial	Neutron	139.9	0.3%	60.1	6.3%
	Gamma	44.7	1.0%	11.9	1.0%
	Total	184.6	0.3%	72.0	5.3%

1. The hypothetical accident conditions include a loss of all oxygen, hydrogen, and nitrogen in the radial neutron shield material and radial and axial lead slumps.
2. Dose rates are rounded to the indicated precision.
3. The azimuthal maximum radial dose rates are 2012 (1.9%) and 771 (4.5%) mrem/hr at the surface and at 1 meter from the surface, respectively.

Table 5.1-6 Combined Top, Radial Midplane and Bottom Dose Rates for Canistered Yankee Class Fuel in Normal Conditions of Transport

Location	Detector I.D.	Radiation	Dose Rate (mrem/hr)
Radial Surface, fuel midplane	1	Fuel Gamma Fuel Neutron (n,γ) TOTAL	3.89 3.46 2.90 10.25
Radial, 1m from cask surface, fuel midplane	2	Fuel Gamma Fuel Neutron (n,γ) TOTAL	1.73 1.29 1.09 4.11
Radial, 2m from transport vehicle, fuel midplane[1]	3	Fuel Gamma Fuel Neutron (n,γ) TOTAL	0.79 0.52 0.41 1.72
Bottom impact limiter surface, axial centerline	4	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.09 0.13 0.37 0.01 0.04 0.64
Bottom, 2m from surface of impact limiter, axial centerline	5	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.05 0.07 0.00* 0.00* 0.02 0.14
Top impact limiter surface, axial centerline	6	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.00* 0.00* 0.00* 0.00* 0.00* 0.00
Top, 2m from surface of impact limiter, axial centerline	7	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.00* 0.00* 0.00* 0.00* 0.00* 0.00

*values are less than 0.005.

Table 5.1-7 Combined Top, Radial Midplane and Bottom Dose Rates for Canistered Yankee Class Fuel in Accident Conditions

Location	Detector I.D.	Radiation	Dose Rate (mrem/hr)
Radial, 1m from cask surface, fuel midplane without neutron shield[1]	2a	Fuel Gamma Fuel Neutron (n,γ) TOTAL	32.11[4] 230.14 0.49 262.78
Bottom, 1m from cask surface, axial centerline, without neutron shield (assumes loss of impact limiter) [1,2]	4	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.81 1.35 4.04 5.35 0.10 11.65
Top, 1m from cask surface, axial centerline, without neutron shield (assumes loss of impact limiter) [2]	6	Fuel Gamma Upper Plenum Gamma Top Endfitting Gamma Fuel Neutron (n,γ) TOTAL	0.00[3] 0.00[3] 0.00[3] 18.20 0.01 18.25

1. Assumes complete loss of neutron shielding material.
2. Assumes loss of impact limiters and positioning of the canister in either the top or bottom of cavity.
3. Values are less than 0.005.
4. Assumes 0.88 reduction in lead shielding due to side drop lead slump.

Table 5.1-8 Canistered Yankee GTCC Waste Dose Rates in Normal Conditions of Transport

Location	Detector I.D.	Radiation	Dose Rate(mrem/hr)
Radial Surface, fuel midplane	1	Neutron Gamma Total	0.00 7.03 7.03
Radial, 1m from cask surface, fuel midplane	2	Neutron Gamma Total	0.00 3.17 3.17
Radial, 2m from transport vehicle, midplane	3	Neutron Gamma Total	0.00 1.49 1.49
Top impact limiter surface, axial centerline	6	Neutron Gamma Total	0.00 0.00 0.0
Top 2m from impact limiter surface, axial centerline	7	Neutron Gamma Total	0.00 0.00 0.0
Bottom impact limiter surface, axial centerline	4	Neutron Gamma Total	0.0 2.54 2.54
Bottom, 2m from cask surface, axial centerline	5	Neutron Gamma Total	0.00 0.46 0.46

Table 5.1-9 Canistered Yankee GTCC Waste Dose Rates in Accident Conditions

Location	Detector I.D.	Dose Rate (mrem/hr)
Radial, 1m from cask surface, fuel midplane, without neutron shielding[1]	2a	55.77
Top surface 1m from cask surface, axial centerline[2]	6	0.01
Bottom, 1m from cask surface, axial centerline, without neutron shield[2,3]	4	22.88

1 Assumes complete loss of neutron shielding material and lead slump. Loss of neutron shielding alone results in a dose of 12.15 mrem/hr. This dose is increased by a factor of 4.59 to account for a 0.88 inch reduction in lead thickness due to lead slump.

2 Assumes loss of impact limiters and positioning of the canister in either the top or bottom of the cavity.

3 Assumes complete loss of neutron shielding material.

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5.2 Source Specification

This section presents the source specifications for the directly loaded fuel and for the Yankee-MPC fuel and GTCC waste configurations.

5.2.1 Directly Loaded Fuel Source Specification

The directly loaded NAC-STC is designed to safely transport a range of 14×14, 15×15, 16×16, and 17×17 fuel assemblies. The analyzed fuel assemblies are reference fuel assemblies, with assembly geometry and activated hardware masses chosen to maximum uranium loading (MTU) and activated hardware source term.

In order to generate a minimum cool time table for directly loaded fuel, each fuel assembly is analyzed over a range of burnups, initial ^{235}U enrichments and cool times. Fuel assembly burnup is evaluated from 30,000 MWD/MTU to 60,000 MWD/MTU in 5,000 MWD/MTU increments. Initial ^{235}U enrichments are evaluated from 1.7 to 4.9 wt % ^{235}U in 0.2 wt % increments. Cool times range from 5 to 40 years with varying increments. This matrix creates a total of 2,142 source terms for each assembly (7 burnups × 17 enrichments × 18 cool times). A maximum of 60,000 MWD/MTU was conservatively evaluated. The limiting assembly burnup of the NAC-STC directly loaded configuration is 45,000 MWD/MTU.

Neutron and gamma source terms for the directly loaded design basis fuel are calculated with the ORIGEN-S computer code (Hermann, 1989) as part of the SAS2H sequence (Hermann, 1995) in the SCALE 4.3 code package for the PC (ORNL, 1995). ORIGEN-S also calculates the gamma spectrum, the neutron spectrum, and the concentration of radiologically important isotopes such as ^{3}H, ^{131}Xe, ^{129}I, ^{85}Kr, ^{134}Cs, ^{137}Cs and ^{60}Co. Reactor operating conditions assumed for the analysis are shown in Table 5.2-3. The SAS2H-generated source spectra are rebinned onto the standard 28 group neutron and 22 group gamma scheme used in MCBEND as shown in Tables 5.2-8 and 5.2-9, respectively. Source terms are generated for the fuel and fuel assembly hardware. The hardware activation is calculated by light element transmutation using the in-core neutron flux spectrum produced by the SAS2H neutronics model.

The fuel-dependent input data for the shielding and source term evaluations of directly loaded design basis PWR assemblies are given in Table 5.2-2. Fuel assembly parameters have been selected to maximize fuel mass and, therefore, fuel source terms. Fuel assembly hardware masses have likewise been selected to maximize hardware source term.

Fuel assembly parameters used in the SAS2H source term analysis and the MCBEND shielding analysis are identical. Parameters necessary to generate SAS2H input are shown in Table 5.2-4.

Fuel neutron, fuel gamma, and hardware gamma radiation contribute at varying levels to cask dose rates due to significant changes in the material composition of the cask shields at different radial and axial locations. As such, no single source term produces a bounding set of dose rates at all locations. For example, the radial maximum surface dose rate of 366 mrem/hr is produced by a 40,000 MWD/MTU, 2.3 wt % ^{235}U, 10-year cooled source term. Top axial maximum dose rates are produced by a 30,000 MWD/MTU, 2.3 wt % ^{235}U, 6-year cooled source term. By employing the response function method to calculate maximum dose rates, the limiting source term becomes a result of the analysis, rather than an input, and the limiting source term and dose rate are captured for radial and axial detectors and normal and accident conditions.

The end-fitting, plenum spring and grid spacer activations are calculated by ORIGEN-S using the same burnup cycle as the fuel. The fuel hardware masses activated are provided in Table 5.2-2 for the directly loaded fuel. The grid spacers and other fuel hardware in the core region are conservatively assumed to be exposed to 100 percent of the flux in the core. For the plenum springs, the grid spacers in the plenum region, and the bottom end-fittings, 20 percent of the flux in the core is used for irradiation purposes. For the top end-fittings, 10 percent of the flux in the core is used for irradiation purposes. These irradiation values are taken from Luksic. The amount of ^{59}Co present in the grid spacers and end-fittings was taken as 1.2 gram per kilogram of material, irrespective of being Inconel or Type 304 stainless steel. However, the value is conservative for both stainless steel and Inconel, as most nuclear-grade material specifications require less than 1 gram of ^{59}Co per kilogram of metal. It is conservatively assumed that all of the cobalt is ^{59}Co. When ^{59}Co absorbs a neutron, it becomes ^{60}Co.

5.2.1.1 Directly Loaded Fuel Neutron Source

As described in Section 5.2.1, a total of 2,142 neutron source terms have been calculated for each directly loaded fuel assembly. Neutron source terms have been rebinned onto the MCBEND 28 group structure, shown in Table 5.2-8. The neutron source results from actinide spontaneous fission and from (α,n) reactions with oxygen in UO_2. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all but a few percent of the spontaneous fission neutrons and (α,n) source in light water reactor fuel. The next largest contribution is from (α,n) reactions of ^{238}Pu with oxygen. The neutron spectrum from spontaneous fission is based on fission spectrum measurements of ^{235}U and ^{252}Cf. Neutron spectra from (α,n) reactions are based on Po-α-O source measurements. These spectra are included in the ORIGEN-S nuclear data libraries of the SCALE 4.3 code package. The spectra are automatically collapsed from the energy group structure of the data library into that of the standard MCBEND 28 group structure.

The effect of subcritical neutron multiplication is not directly computed in the MCBEND analysis conducted for directly loaded fuel, due to difficulties in adequately biasing the calculation. Instead, neutron source rates are scaled by a subcritical multiplication factor based on the system multiplication factor, k_{eff}:

$$\text{Scale Factor} = \frac{1}{1 - k_{eff}}$$

For the dry cask conditions of transport, the system k_{eff} is taken as 0.4, with a resulting scale factor of 1.67. This scale factor is input as a scaling factor on the source strength input in MCBEND.

5.2.1.2 Directly Loaded Fuel Gamma Sources

As described in Section 5.2.1, 2,142 gamma source terms have been calculated for each directly loaded fuel assembly. Gamma source terms have been rebinned onto the MCBEND 22 group structure, shown in Table 5.2-9. The hardware gamma spectrum for directly loaded fuel contains contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with some minor contributions from ^{59}Ni and ^{58}Fe. The magnitude of these spectra is based on the irradiation of 1 kg of stainless steel in the in-core flux spectrum produced by the SAS2H neutronics calculation.

The activated fuel assembly hardware source terms are found by multiplying the source strength from 1 kilogram by the kilograms of steel or inconel material in the plenum, upper end fitting or lower end fitting regions, and by multiplying by a regional flux ratio. The regional flux ratio accounts for the effects of both magnitude and spectrum variation on hardware activation. These ratios are based on empirical data (Luksic). A flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region (i.e., upper and lower plenum) and a flux ratio of 0.1 is applied to hardware regions once removed from the active core region (i.e., upper and lower end fitting region). Activated mass in each region and the corresponding flux factor are summarized in Table 5.2-10 for each array size. In the case of CE 16×16 fuel, which has a longer plenum, the upper end fitting (upper nozzle) flux factor is reduced to 0.05 (Luksic).

5.2.1.3 Directly Loaded Fuel Source Axial Profiles

The design basis axial burnup profile used in the directly loaded fuel shielding evaluations is shown in Figure 5.2-1. Neutron and gamma source profiles are computed based on the relation between burnup, B, and source strength, S, in the form:

$$S = aB^b$$

where parameters a and b are determined based on fits to SAS2H computed source rates at various fuel burnups. The parameter a is simply a scaling factor and is not relevant to the analysis. For neutron sources, parameter b is 4.22. For gamma sources, the relation between burnup and source rate is linear and b is 1.0. Table 5.2–11 gives the resulting neutron and gamma source rate profiles for directly loaded fuel. The relative source strength in each axial interval is shown, and these values are used directly in the MCBEND source strength description by defining an axial source mesh within the fuel region at the indicated elevations for each fuel type. A plot of the axial source profiles is shown in Figure 5.2–4.

5.2.2 Yankee Class Fuel and GTCC Waste Source Specification

The canistered fuel design basis source terms are based on the CE 16 x 16 Yankee Class fuel assembly with a burnup of 36,000 MWD/MTU and 8.1 years cooling time. An enrichment of 3.7 wt % ^{235}U is selected to maximize the neutron source for this type of fuel. Dose rates associated with the Yankee Class Westinghouse, United Nuclear, and CE (3.5 wt % ^{235}U) fuel types at 32,000 MWD/MTU are bounded by the canister fuel design basis for cooling times of 19, 11 and 7 years, respectively. Exxon fuel at 36,000 MWD/MTU with steel or Zircaloy fuel hardware is bounded by the canister fuel design basis for cooling times of 16 and 9 years, respectively.

Neutron and gamma source terms for the canistered design basis fuel are calculated with the SAS2H code sequence of the SCALE 4.3 code package for the PC. SAS2H includes an XSDRNPM neutronics model of the fuel assembly and ORIGEN-S fuel depletion/source term calculations. The canister fuel assembly input data for SAS2H is summarized in Table 5.2-1. Source terms are generated for both UO_2 fuel and fuel assembly hardware. The hardware activation is calculated by light element transmutation using the in-core neutron flux spectrum produced by the SAS2H neutronics model. The hardware is assumed to be Type 304 stainless steel with 1.2 g/kg of ^{59}Co impurity. The effects of axial flux spectrum and magnitude variation on hardware activation are estimated by flux ratios based on empirical data (Luksic).

5.2.2.1 Yankee Class Fuel Neutron Source

The Yankee Class canistered fuel neutron spectrum is shown in Table 5.2-5. The neutron source results from actinide spontaneous fission and from (α,n) reactions with oxygen in UO_2. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all but a few percent of the spontaneous fission neutrons and (α,n) source in light water reactor fuel. The next largest contribution is from (α,n) reactions of ^{238}Pu with oxygen. The neutron spectrum from spontaneous fission is based on fission spectrum measurements of ^{235}U and ^{252}Cf. Neutron spectra from (α,n) reactions are based on Po-α-O source measurements. These spectra are included in the ORIGEN-S nuclear data libraries of the SCALE 4.3 code package. The spectra are automatically collapsed from the energy group structure of the data library into that of the SCALE 27 group neutron cross-section library.

5.2.2.2 Yankee Class Fuel and Yankee GTCC Waste Gamma Sources

The design basis gamma spectrum for Yankee Class canistered fuel is shown in Table 5.2-6. The fuel gamma radiation source consists primarily of decay gammas from fission products. Actinides also emit a significant amount of gamma radiation. The gamma source strength depends on the irradiation period and the cooling time after discharge from the reactor core.

An additional source of gamma radiation is from ^{59}Co activation in the fuel hardware materials. The fuel hardware gamma spectrum for canistered fuel is shown in Table 5.2-7. The gamma spectrum for the decay of ^{60}Co in the activated hardware was calculated using ORIGEN-S. The total source in each hardware region depends on the flux used to irradiate the region and the mass of material in that region. The default gamma energy group spectrum of the ORIGEN-S code differs from that of the standard 18-group gamma library of the SCALE-4.0 package. To account for the differences in the energy groups, the source spectra from the fuel and hardware gamma sources were rebinned to the SCALE-4.0 18-group structure using ORIGEN-S. This method regroups the source based on the actual energy spectrum of each specific nuclide, yielding more

accurate results than those achieved by simply multiplying the individual energy group source strength by the ratio of the old to new mean energies of each respective group.
The hardware gamma spectra contains contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with some minor contributions from ^{59}Ni and ^{58}Fe. The magnitude of these spectra is based on the irradiation of 1 kg of stainless steel in the in-core flux spectrum produced by the SAS2H neutronics calculation. This activated hardware spectra is used for the design basis GTCC waste spectra, but the magnitude is scaled up from 103 curies of ^{60}Co in the 1 kg of activated hardware to 1.25×10^5 curies ^{60}Co in the GTCC waste.

The activated fuel assembly hardware source terms are found by multiplying the source strength from 1 kilogram by the kilograms of steel or inconel material in the plenum, upper end fitting or lower end fitting regions, and by multiplying by a regional flux ratio. The regional flux ratio accounts for the effects of both magnitude and spectrum variation on hardware activation. These ratios are based on empirical data (Luksic). A flux ratio of 0.2 is applied to hardware regions directly adjacent to the active core region (i.e., upper and lower plenum) and a flux ratio of 0.1 is applied to hardware regions once removed from the active core region (i.e., upper and lower end fitting region).

5.2.2.3 Yankee Class Fuel Source Axial Profiles

The Yankee Class fuel axial burnup profile used in the shielding evaluations is shown in Figure 5.2-2. This is based on core calculations of Yankee Class fuel in the range of 30,000 to 36,000 MWD/MTU of burnup. This burnup profile has a peaking factor of 1.15. Thus, a peaking factor of 1.15 is applied to the radial midplane gamma dose rates and a peaking factor of $(1.15)^{4.2}$ = 1.80 is applied to the radial midplane neutron dose rates reported in Tables 5.1-6 and 5.1-7.

The design basis gamma source profile for Yankee GTCC waste (activated stainless steel core baffle) is shown in Figure 5.2-3. A GTCC gamma source peaking factor of 1.23 is determined from actual dose rate measurements of the GTCC waste containers. This peaking factor is due to the activation of the core baffle from 30 years of neutron flux exposure. This neutron flux exposure produces an activation profile similar to the chopped cosine axial shape of the neutron flux during reactor operation. The GTCC source term includes an estimated contribution from crud (as surface contamination).

Figure 5.2-1 Directly Loaded Fuel Design Basis Burnup Profile

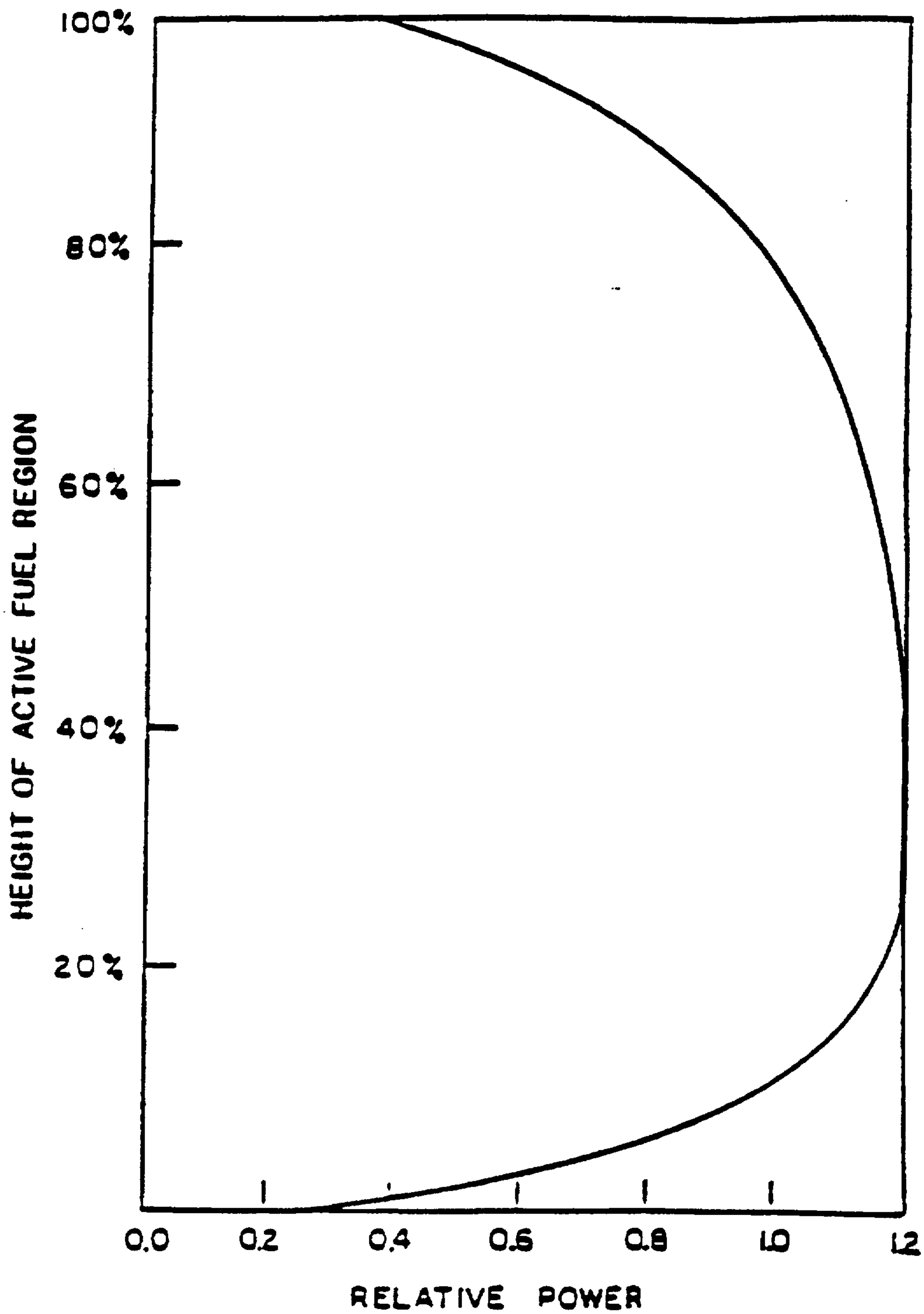

Figure 5.2-2 Yankee-MPC Fuel Burnup Profile

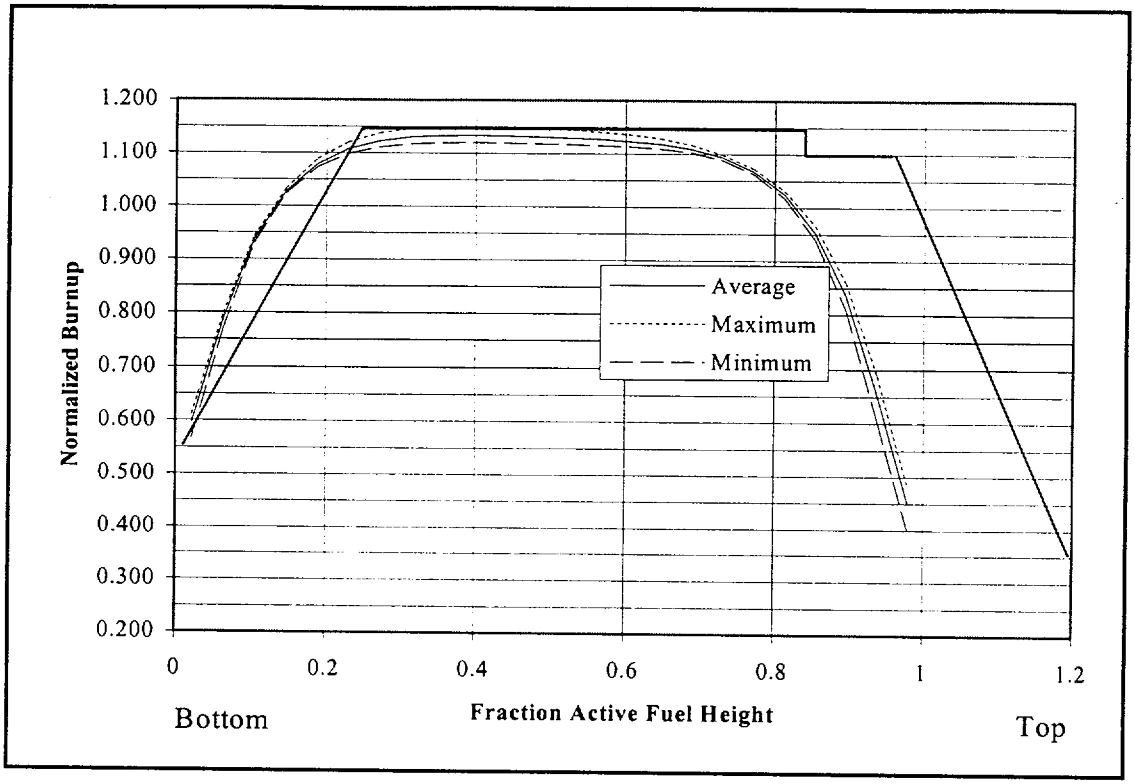

Figure 5.2-3 Yankee GTCC Container Gamma Source Profile Based on Dose Rate Measurements

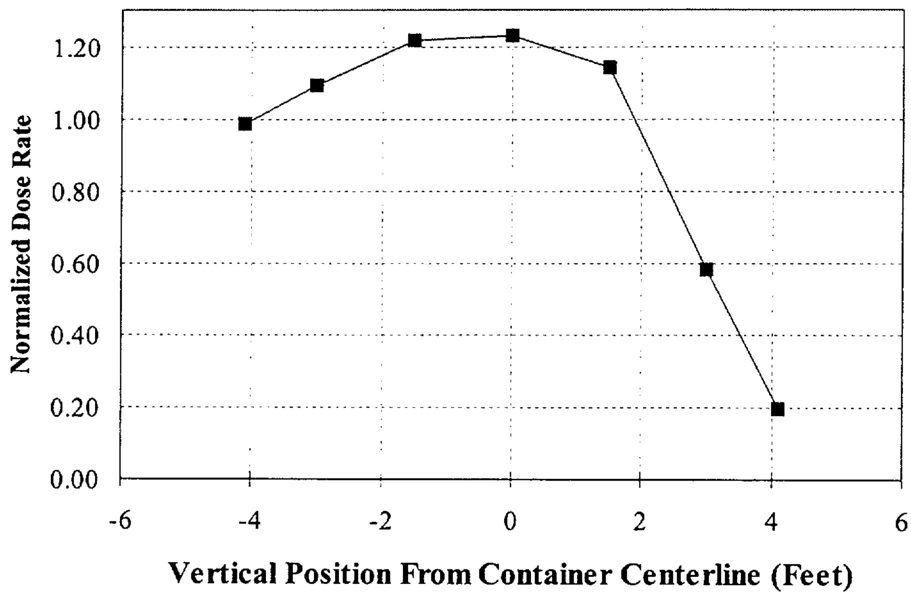

Figure 5.2-4 Directly Loaded Fuel Neutron and Gamma Source Profiles

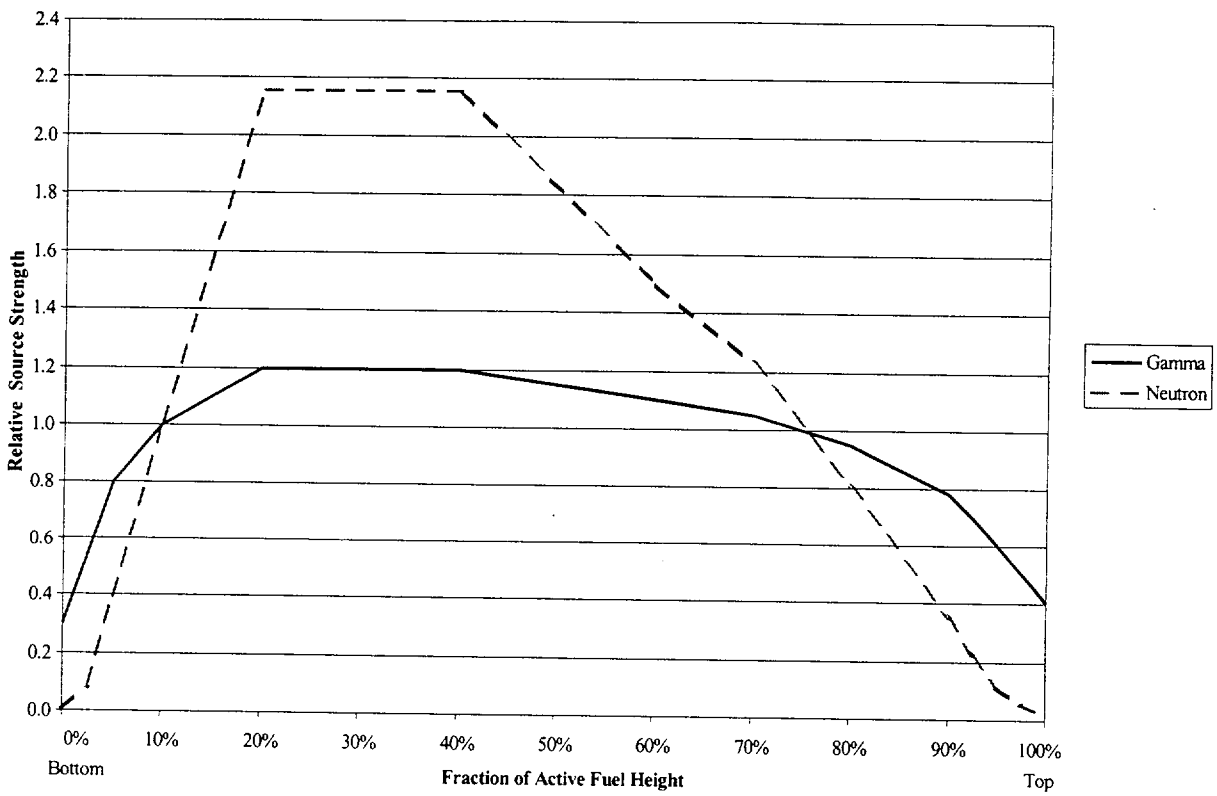

Table 5.2-1 Design Basis Yankee Class Fuel Input Parameters for SAS2H

Parameter	Value
Basket Configuration	Canistered
Fuel assembly type	CE 16x16 Yankee Class
Weight of U, kg/assembly	239.4
In core grids, kg/assembly	2.36 (4 Zirc)
Plenum spring, kg/assembly	0.762
Grids in plenum springs, kg/assembly	0.590 (Zirc)
Upper end fittings, kg/assembly	5.5
Lower end fittings, kg/assembly	5.2
Lower Plenum Hardware, kg/assembly	1.73
Fuel enrichment, wt.% ^{235}U	3.7
Fuel burnup, MWD/MTU	36,000
Cooling time	8
Burnup cycle, power cycles, down cycles	2 cycles of 496 days 1 of 60 days
Burnup, MWD/assembly	8,618
Irradiation power, MW	8.486
^{59}Co concentration in steel hardware, g/kg	1.2
Irradiation flux, grid spacers in core region grid spacers in plenum region upper plenum springs upper end-fittings lower end-fittings lower plenum hardware	100% 20% 20% 10% 10% 20%
Fuel temperature, K	787
Clad temperature, K	600
Coolant temperature, K	551
Boron content in coolant, ppm (by weight)	800

Table 5.2-2 Directly Loaded Three-Dimensional PWR Reference Fuel Assembly Descriptions

Parameter Description	Reference Fuel Assembly			
	14×14	15×15	16×16	17×17
Fuel Rod Height [inch]	152.360	152.756	146.499	152.300
Top End-Cap Height [inch]	0.685	0.685	0.750	0.685
Bottom End-Cap Height [inch]	0.685	0.685	0.891	0.895
Active Fuel Region Height [inch]	145.2	144.0	136.7	144.0
Fuel Rod Diameter [inch]	0.422	0.422	0.382	0.374
Fuel Clad Thickness [inch]	0.023	0.024	0.025	0.022
Fuel Pellet Diameter [inch]	0.367	0.367	0.325	0.323
Array	14	15	16	17
Fuel Rod Pitch [inch]	0.556	0.563	0.506	0.496
Number of Guide Tubes	16	20	4	24
Guide Tube OD [inch]	0.481	0.484	1.115	0.474
Guide Tube Thickness [inch]	0.034	0.015	0.026	0.015
Number of Instrument Tubes	1	1	1	1
Instrument Tube OD [inch]	0.481	0.484	1.115	0.474
Instrument Tube Thickness [inch]	0.034	0.015	0.026	0.015
Fuel Assembly Height [inch]	161.100	160.100	158.129	161.693
Fuel Assembly Width [inch]	7.763	8.449	8.100	8.430
Lower Nozzle Height [inch]	2.738	2.738	3.821	2.421
Upper Nozzle Height [inch]	3.500	3.480	6.821	3.480
Gap Fuel Rod to Bottom Nozzle [inch]	0.000	0.813	0.000	0.791
Gap Fuel Rod to Top Nozzle [inch]	2.502	0.313	0.988	2.701
Upper Plenum Region Height [inch]	5.790	7.386	8.158	6.720
Number of Fuel Rods	179	204	236	264
Calculated MTU [MTU]	0.4144	0.4671	0.4025	0.4636
Lower Nozzle Hardware Mass [kg]	7.893	5.680	5.400	6.307
In-core Hardware Mass [kg][1]	14.880	17.450	1.360	5.440
Upper Plenum Hardware Mass [kg]	8.050	4.120	10.700	5.410
Upper Nozzle Hardware Mass [kg]	9.890	11.840	9.500	7.850

1. Increased hardware mass in 14×14 and 15×15 assemblies due to steel guide/instrument tubes in reference fuel models.

Table 5.2-3 PWR Fuel Reactor Operating Conditions for Directly Loaded Fuel

Reference Assembly Parameter	**14×14**	**15×15**	**16×16**	**17×17**
Assembly Power, MW	13.72	16.33	16.62	18.55
Fuel Temperature, K	900	900	900	900
Clad Temperature, K	620	620	620	620
Moderator Temperature, K	580	580	580	580
Moderator Density, g/cc	0.725	0.725	0.725	0.725
Boron, ppm	550	550	550	550
Down Time, days	60	60	60	60

Table 5.2-4 PWR Cycle Length Calculation for Directly Loaded Fuel Source Terms

Reference Fuel Type	Burnup [MWD/MTU]	% TD	Pellet OD [cm]	Active Length [cm]	Number of Rods	Volume [cm^3]	MTU [MTU]	Assy Power [MW]	Number of Cycles	Cycle Length [days]
14×14	30000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	2	453.14
	35000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	352.45
	40000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	402.80
	45000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	453.14
	50000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	503.49
	55000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	553.84
	60000	0.95	0.9332	368.808	179	4.52E+04	0.4144	13.065	3	604.19
15×15	30000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	2	429.15
	35000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	333.78
	40000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	381.46
	45000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	429.15
	50000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	476.83
	55000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	524.51
	60000	0.95	0.9319	365.76	204	5.09E+04	0.4671	15.55	3	572.20
16×16	30000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	2	363.26
	35000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	282.53
	40000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	322.90
	45000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	363.26
	50000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	403.62
	55000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	443.98
	60000	0.95	0.8255	347.218	236	4.39E+04	0.4025	15.83	3	484.34
17×17	30000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	2	374.82
	35000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	291.53
	40000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	333.18
	45000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	374.82
	50000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	416.47
	55000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	458.12
	60000	0.943	0.8192	365.76	264	5.09E+04	0.4636	17.67	3	499.76

Table 5.2-5 Design Basis Yankee Class Fuel Neutron Source Spectra at 36,000 MWD/MTU and 8 Years Cooling

GROUP	E_{HI} (MeV)	E_{LOW} (MeV)	Neutrons/Sec-Assembly
1	2.00E+01	6.43E+00	1.2290E+06
2	6.43E+00	3.00E+00	1.4080E+07
3	3.00E+00	1.85E+00	1.5760E+07
4	1.85E+00	1.40E+00	8.7930E+06
5	1.40E+00	9.00E-01	1.1840E+07
6	9.00E-01	4.00E-01	1.2870E+07
7	4.00E-01	1.00E-01	2.5190E+06
8	1.00E-01	1.70E-02	0.0000E+00
9	1.70E-02	3.00E-03	0.0000E+00
10	3.00E-03	5.50E-04	0.0000E+00
11	5.50E-04	1.00E-04	0.0000E+00
12	1.00E-04	3.00E-05	0.0000E+00
13	3.00E-05	1.00E-05	0.0000E+00
14	1.00E-05	3.05E-06	0.0000E+00
15	3.05E-06	1.77E-06	0.0000E+00
16	1.77E-06	1.30E-06	0.0000E+00
17	1.30E-06	1.13E-06	0.0000E+00
18	1.13E-06	1.00E-06	0.0000E+00
19	1.00E-06	8.00E-07	0.0000E+00
20	8.00E-07	4.00E-07	0.0000E+00
21	4.00E-07	3.25E-07	0.0000E+00
22	3.25E-07	2.25E-07	0.0000E+00
23	2.25E-07	1.00E-07	0.0000E+00
24	1.00E-07	5.00E-08	0.0000E+00
25	5.00E-08	3.00E-08	0.0000E+00
26	3.00E-08	1.00E-08	0.0000E+00
27	1.00E-08	1.00E-11	0.0000E+00
TOTAL			6.7090E+07

Table 5.2-6 Design Basis Canistered Fuel Gamma Source Spectra at 36,000 MWD/MTU and 8 Years Cooling

GROUP	E_{HI} (MeV)	E_{LOW} (MeV)	Photons/Sec-Assembly
1	1.00E+01	8.00E+00	3.7701E+04
2	8.00E+00	6.50E+00	1.7759E+05
3	6.50E+00	5.00E+00	9.0547E+05
4	5.00E+00	4.00E+00	2.2566E+06
5	4.00E+00	3.00E+00	6.2676E+08
6	3.00E+00	2.50E+00	5.1211E+09
7	2.50E+00	2.00E+00	1.0789E+11
8	2.00E+00	1.66E+00	9.9933E+10
9	1.66E+00	1.33E+00	4.8070E+12
10	1.33E+00	1.00E+00	3.4718E+13
11	1.00E+00	8.00E-01	6.3503E+13
12	8.00E-01	6.00E-01	8.2333E+14
13	6.00E-01	4.00E-01	1.1897E+14
14	4.00E-01	3.00E-01	1.7831E+13
15	3.00E-01	2.00E-01	2.8386E+13
16	2.00E-01	1.00E-01	1.0201E+14
17	1.00E-01	5.00E-02	1.3136E+14
18	5.00E-02	1.00E-02	4.5899E+14
Total			1.7842E+15

Table 5.2-7 Design Basis Yankee Canistered Fuel Hardware and GTCC Waste Gamma Spectra

GROUP	E_{HI} (MeV)	E_{LOW} (MeV)	Photons/Sec-kg
1	1.00E+01	8.00E+00	0.0000E+00
2	8.00E+00	6.50E+00	0.0000E+00
3	6.50E+00	5.00E+00	0.0000E+00
4	5.00E+00	4.00E+00	0.0000E+00
5	4.00E+00	3.00E+00	1.0141E-15
6	3.00E+00	2.50E+00	3.3511E+04
7	2.50E+00	2.00E+00	2.1611E+07
8	2.00E+00	1.66E+00	9.5163E-03
9	1.66E+00	1.33E+00	9.1066E+11
10	1.33E+00	1.00E+00	3.2247E+12
11	1.00E+00	8.00E-01	4.3841E+09
12	8.00E-01	6.00E-01	3.8100E+06
13	6.00E-01	4.00E-01	1.0971E+07
14	4.00E-01	3.00E-01	1.7359E+08
15	3.00E-01	2.00E-01	1.3230E+08
16	2.00E-01	1.00E-01	2.6645E+09
17	1.00E-01	5.00E-02	1.1044E+10
18	5.00E-02	1.00E-02	5.5673E+10
TOTAL			4.2095E+12

Table 5.2-8 MCBEND Standard 28 Group Neutron Boundaries

Group	E Lower [MeV]	E Upper [MeV]	E Average [MeV]
1	1.360E+01	1.460E+01	1.410E+01
2	1.250E+01	1.360E+01	1.305E+01
3	1.125E+01	1.250E+01	1.188E+01
4	1.000E+01	1.125E+01	1.063E+01
5	8.250E+00	1.000E+01	9.125E+00
6	7.000E+00	8.250E+00	7.625E+00
7	6.070E+00	7.000E+00	6.535E+00
8	4.720E+00	6.070E+00	5.395E+00
9	3.680E+00	4.720E+00	4.200E+00
10	2.870E+00	3.680E+00	3.275E+00
11	1.740E+00	2.870E+00	2.305E+00
12	6.400E-01	1.740E+00	1.190E+00
13	3.900E-01	6.400E-01	5.150E-01
14	1.100E-01	3.900E-01	2.500E-01
15	6.740E-02	1.100E-01	8.870E-02
16	2.480E-02	6.740E-02	4.610E-02
17	9.120E-03	2.480E-02	1.696E-02
18	2.950E-03	9.120E-03	6.035E-03
19	9.610E-04	2.950E-03	1.956E-03
20	3.540E-04	9.610E-04	6.575E-04
21	1.660E-04	3.540E-04	2.600E-04
22	4.810E-05	1.660E-04	1.071E-04
23	1.600E-05	4.810E-05	3.205E-05
24	4.000E-06	1.600E-05	1.000E-05
25	1.500E-06	4.000E-06	2.750E-06
26	5.500E-07	1.500E-06	1.025E-06
27	7.090E-08	5.500E-07	3.105E-07
28	1.000E-11	7.090E-08	3.546E-08

Table 5.2-9 MCBEND Standard 22 Group Gamma Boundaries

Group	E Lower [MeV]	E Upper [MeV]	E Average [MeV]
1	1.200E+01	1.400E+01	1.300E+01
2	1.000E+01	1.200E+01	1.100E+01
3	8.000E+00	1.000E+01	9.000E+00
4	6.500E+00	8.000E+00	7.250E+00
5	5.000E+00	6.500E+00	5.750E+00
6	4.000E+00	5.000E+00	4.500E+00
7	3.000E+00	4.000E+00	3.500E+00
8	2.500E+00	3.000E+00	2.750E+00
9	2.000E+00	2.500E+00	2.250E+00
10	1.660E+00	2.000E+00	1.830E+00
11	1.440E+00	1.660E+00	1.550E+00
12	1.220E+00	1.440E+00	1.330E+00
13	1.000E+00	1.220E+00	1.110E+00
14	8.000E-01	1.000E+00	9.000E-01
15	6.000E-01	8.000E-01	7.000E-01
16	4.000E-01	6.000E-01	5.000E-01
17	3.000E-01	4.000E-01	3.500E-01
18	2.000E-01	3.000E-01	2.500E-01
19	1.000E-01	2.000E-01	1.500E-01
20	5.000E-02	1.000E-01	7.500E-02
21	2.000E-02	5.000E-02	3.500E-02
22	1.000E-02	2.000E-02	1.500E-02

Table 5.2-10 Directly Loaded PWR Fuel Assembly Hardware Mass and Activation Scale Factors by Source Region

Reference Fuel Type	Region	Activated Mass [kg/assy]	Flux Factor
14×14	Lower Nozzle	7.89	0.20
	Fuel	14.88	1.00
	Upper Plenum	8.05	0.20
	Upper Nozzle	9.89	0.10
15×15	Lower Nozzle	5.68	0.20
	Fuel	17.45	1.00
	Upper Plenum	4.12	0.20
	Upper Nozzle	11.84	0.10
16×16	Lower Nozzle	5.40	0.20
	Fuel	1.36	1.00
	Upper Plenum	10.70	0.20
	Upper Nozzle	9.50	0.05
17×17	Lower Nozzle	6.31	0.20
	Fuel	5.44	1.00
	Upper Plenum	5.41	0.20
	Upper Nozzle	7.85	0.10

Table 5.2-11 Directly Loaded Fuel Axial Gamma and Neutron Source Profiles

% Core Height	Burnup Profile	Gamma Interval	Neutron Interval
0.0	0.30	--	--
2.5	0.55	4.250E-01	4.322E-02
5.0	0.80	6.750E-01	2.351E-01
10.0	1.00	9.000E-01	6.950E-01
20.0	1.20	1.100E+00	1.579E+00
40.0	1.20	1.200E+00	2.158E+00
60.0	1.10	1.150E+00	1.827E+00
70.0	1.05	1.075E+00	1.362E+00
80.0	0.95	1.000E+00	1.017E+00
90.0	0.78	8.650E-01	5.779E-01
92.5	0.70	7.400E-01	2.862E-01
95.0	0.60	6.500E-01	1.689E-01
97.5	0.50	5.500E-01	8.474E-02
100.0	0.40	4.500E-01	3.729E-02

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5.3 Model Specification

The radiation protection provided by the NAC-STC is in the form of solid multi-walled shielding materials, which totally surround the fuel. These shielding materials include steel and lead for gamma shielding and a borated polymer (NS-4-FR) for neutron shielding. The multi-walled arrangement of steel and lead in the NAC-STC provides optimal weight for gamma attenuation. The NS-4-FR neutron shielding material has a hydrogen density close to that of water and serves to moderate fast neutrons, which are then captured in the boron. Boron capture in the neutron shield minimizes the contribution of secondary capture gammas to surface dose rates.

The NAC-STC uses a multi-walled arrangement for both radial and axial shields. The arrangement of the radial gamma shielding in the cask body is a 1.5-inch thick stainless steel inner shell and a 2.65-inch thick stainless steel outer shell with a 3.70-inch lead annulus between them. The radial neutron shield is arranged around the outer steel shell with a 5.5-inch minimum, 5.925-inch maximum thickness of NS-4-FR that is covered by a 0.236-inch (6 mm) thick neutron shield shell. The bottom of the cask contains a steel/NS-4-FR/steel shield arrangement with the two stainless steel components providing 11.65 inches of gamma shielding and 2 inches of NS-4-FR neutron shielding. The top of the cask has shields in the form of two closure lids. The inner lid also has a steel/NS-4-FR/steel arrangement with 6.0 inches of steel below 2 inches of NS-4-FR and 1.0 inch of steel above it. The outer lid is a 5.25-inch thick steel disk.

5.3.1 Directly Loaded Fuel Model

MCBEND three-dimensional shielding analysis allows detailed modeling of fuel assemblies, basket, and cask shield configuration, including streaming paths. For fuel assembly sources, some fuel assembly detail is homogenized in the model to simplify model input and improve computational efficiency. Thus, the three-dimensional models represent the various fuel assembly source regions as homogenized zones within the fuel tubes in the basket, but explicitly model the axial extent of the source regions. The fuel and hardware source regions of each assembly are therefore homogenized within the volumes defined by the periphery of the fuel assembly and the source region axial extents. The basket details, including support disks, heat transfer disks, and top and bottom weldments are explicitly modeled. Cask body details include the axial extent of the cask shield as described by the License Drawings.

The geometric description of a MCBEND model is based on the combinatorial geometry system embedded in the code. In this system, bodies such as cylinders and rectangular parallelpipeds, and their logical intersections and unions, are used to describe the extent of material zones.

MCBEND employs an automated biasing technique for the Monte Carlo calculation based on a three-dimensional adjoint diffusion calculation. Mesh cells for the adjoint solution are selected based on half value thicknesses for each material.

MCBEND Monte Carlo calculations are performed for each source type present in each source region. This approach entails seven separate analyses, encompassing fuel neutron, fuel gamma, fuel n-gamma (secondary gammas arising from neutron interaction in the shield), fuel region hardware, upper plenum, and upper and lower end-fitting gamma sources. Typically, a total of 5 to 20 million histories are tracked to yield dose rate profiles for each model. These cases are analyzed for both radial and axial detector locations and for normal and hypothetical accident conditions.

5.3.1.1 Directly Loaded Fuel Assembly Model

Based on the fuel parameters provided in Table 5.2-2, homogenized treatments of fuel assembly source regions are developed. The homogenized fuel assembly is represented in the model as a stack of boxes with width equal to the fuel assembly width. The height of each box corresponds to the modeled height of the corresponding assembly region.

The active fuel region homogenizations for the four design basis assemblies are shown in Table 5.3-1. The interstitial material is void under the dry canister conditions of transport. The clad region is Zircaloy (density 6.55 g/cm^3) for all four design basis assemblies. The resulting regional compositions on an atom/barn-cm basis are shown in Table 5.3-2.

Fuel assembly non-fuel regions are homogenized as shown in Tables 5.3-3 and 5.3-4 for stainless steel and Zircaloy materials, respectively. The only material included in the homogenized region is stainless steel for the upper end fitting and combinations of Zircaloy and stainless steel in the upper plenum and lower end fitting regions. Zircaloy in these regions is due to end caps and the portion of the fuel rod cladding in the upper plenum. Volume fractions of material are based on the modeled regional volume and the volume of stainless steel or Zircaloy present as computed from the modeled mass and density (7.92 g/cm^3 for stainless steel and 6.55 g/cm^3 for Zircaloy).

5.3.1.2 Directly Loaded Basket Model

For a given fuel type, the MCBEND description of the basket elements forms a common sub-model employed in the transport cask analysis. The key features of the model are the detailed representation of fuel tubes, basket support and heat transfer disks, and weldment structures.

5.3.1.3 Description of MCBEND NAC-STC Model

The three-dimensional model of the NAC-STC cask containing design basis fuel assemblies is based on the following features:

Normal conditions:
- Radial neutron shield and shield shell (includes heat fins in the neutron shield)
- All balsa upper and lower impact limiters (100% balsa chosen for conservatism)

Accident conditions:
- Top axial lead slump
- Bottom axial lead slump
- Radial lead slump
- Radial neutron shield and shield shell with removal of oxygen, hydrogen, and nitrogen from NS-4-FR material definition
- Loss of upper and lower impact limiters

Features common to both the normal and accident conditions models are the inner lid vent and drain ports, the inner lid neutron shield, the bottom forging neutron shield, the annular expansion foam region below the neutron shield (modeled conservatively as void), radial neutron shield heat fins, and the lower rotating trunnions.

Detailed model parameters used in creating the three-dimensional model are taken directly from the License Drawings. Elevations associated with the transport cask three-dimensional features are established with respect to the center bottom of the NAC-STC cavity for the MCBEND combinatorial model. The three-dimensional NAC-STC model is shown in Figure 5.3-1.

5.3.1.4 Directly Loaded Configuration Shield Regional Densities

Based on the homogenization described in Section 5.3.1.1, the resulting active fuel regional densities are shown in Table 5.3-2 for the design basis directly loaded fuel assemblies. Material compositions for remaining structural and shield materials are shown in Table 5.3-5. Compositions for fuel assembly non-fuel regions are equivalent to the stainless steel and Zircaloy compositions in Table 5.3-5 scaled by the material volume fractions shown in Tables 5.3-3 and 5.3-4.

5.3.2 Yankee-MPC Fuel and GTCC Waste Models

This section provides the radial and axial shielding models used for the Yankee-MPC canistered fuel and GTCC waste.

5.3.2.1 Yankee-MPC Fuel and GTCC Waste Radial Shielding Model

One-dimensional cylindrical SAS1 models are used to evaluate radial midplane dose rates for the NAC-STC containing design basis canister fuel and GTCC waste. In both cases, the source region is transformed into an equivalent cylindrical volume. In the case of the canister fuel region, this volume is based on the periphery of the fuel basket tubes and has an equivalent radius of 30.63 inches (77.80 cm). The fuel assembly source regions are homogenized into the volumes defined by the fuel/basket equivalent radius and the fuel regional elevations defined in Figure 5.1-3. Since the canister basket contains an explicit heat transfer region with aluminum heat transfer disks, an additional middle fuel region is defined with this material for the radial midplane evaluation. The remaining cask body regions are modeled using the exact dimensions of the cask except for the radial neutron shield. Its thickness varies as a result of its polygon shape. An equivalent thickness of 5.52 inches (14.02 cm) is modeled to conserve the actual neutron shield volume (see Figure 5.3-10). To account for axial leakage, an axial buckling equivalent to the active fuel height is applied. In the accident situation, the neutron shield material, NS-4-FR, is voided. An axial peaking factor of 1.15 and 1.80 is applied to the radial midplane gamma and neutron results, respectively, to account for the axial burnup profile as described in Section 5.2.3.

Radial models are also constructed for the Westinghouse, United Nuclear and Exxon fuel types to determine minimum cool time based on shielding constraints. By performing shielding analysis rather than source term magnitude comparisons, spectrum differences are taken into account. Shielding analysis of the Exxon assembly at 10 years cooling is not required, since its neutron and gamma source is lower in each energy group and its mass (and therefore its self-shielding) is identical to the design basis assembly.

In the case of the GTCC waste, the volume is based on the interior periphery of the GTCC basket support wall and has an equivalent radius of 23.47 inches (59.61 cm). The GTCC source region is homogenized into the volume defined by the GTCC basket interior periphery and the container height of 98.25 inches (249.56 cm [See Figure 5.1-4]). This gives a GTCC source volume of 170,023 inches^3 (2.786 x 10^6 cm^3). The GTCC basket support wall is also cylindrical with a 2.5-inch (6.35-cm) thickness. The remaining cask body regions are modeled using the exact dimensions of the cask, except for the radial neutron shield that is again modeled with an equivalent thickness of 5.52 inches (14.02 cm [See Figure 5.3-13]). To account for axial leakage, an axial buckling equivalent to the container height of 98.35 inches is applied. In the accident situation, the neutron shield material, NS-4-FR, is voided. An axial peaking factor of 1.23 is applied to the radial midplane results to account for gamma source peaking as described in Section 5.2.3.

5.3.2.2 Yankee-MPC Fuel and GTCC Waste Axial Shielding Models

One-dimensional slab SAS1 models are used to evaluate top and bottom dose rates for the NAC-STC containing design basis canister fuel and GTCC waste. The top axial model begins at either the active fuel or homogenized GTCC waste canister source midplane and proceeds along the cask centerline to the surface of the top impact limiter. Similarly, the NAC-STC bottom axial model begins at either the canistered fuel or homogenized GTCC waste canister center and ends at the bottom impact limiter. See Figures 5.3-11 and 5.3-12 for the canister fuel axial shielding model and Figures 5.3-14 and 5.3-15 for the canister GTCC waste axial shielding models. To account for transverse radial leakage, radial bucklings equal to the equivalent diameter of the source regions are applied. In the accident situation, the impact limiters are lost, the canister is positioned on either the top or bottom cavity surface, and the NS-4-FR neutron shield material, including that in the bottom forging, is considered to be lost.

5.3.2.3 Cask Regional Material Compositions – Yankee Class Fuel and GTCC Waste

The densities of the materials used in the shielding evaluations for canister fuel and GTCC waste are calculated using the effective fuel radius and source regional elevation. See Figure 5.1-3 for the design basis canistered fuel source zones and elevations. In the case of the canistered Yankee Class fuel, the homogenized source regions include a top fuel, middle fuel (heat transfer zone), bottom fuel, top plenum, bottom plenum, and the top and bottom end-fittings. The structural and heat transfer disks exterior to the fuel/basket region are also homogenized in the one-dimensional radial models. Similarly, the GTCC waste density is based on homogenizing the mass of GTCC waste into the volume defined by the equivalent radius and the height of the container. The homogenized densities and nuclide concentrations are shown in Table 5.3-4.

Figure 5.3-1 Three-Dimensional MCBEND Model for Directly Loaded Fuel

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-2 One-Dimensional Radial Shielding Model for Canistered Fuel

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-3 One-Dimensional Axial Shielding Model for Canistered Fuel

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-4 One-Dimensional Top Axial Model for Canistered Fuel

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-5 One-Dimensional Radial Shielding Model for Canistered GTCC Waste

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-6 One-Dimensional Bottom Axial Model for Canistered GTCC Waste

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 5.3-7 One-Dimensional Top Axial Model for Canistered GTCC Waste

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Table 5.3-1 Directly Loaded Fuel Region Homogenization

Reference Fuel Type	Component	Volume Fraction of Components			
		UO_2	Void	Clad	Interstitial
14×14	Fuel	3.1489E-01	--	--	--
	Gap	--	1.6671E-02	--	--
	Clad	--	--	8.3877E-02	--
	Guide Tube	--	--	1.2662E-02	--
	Instrument Tube	--	--	7.9139E-04	--
	Inside Tubes	--	--	--	3.7699E-02
	Interstitial	--	--	--	5.3341E-01
	Total	3.1489E-01	1.6671E-02	9.7331E-02	5.7111E-01
15×15	Fuel	3.0214E-01	--	--	--
	Gap	--	1.1135E-02	--	--
	Clad	--	--	8.6427E-02	--
	Guide Tube	--	--	6.1920E-03	--
	Instrument Tube	--	--	3.0960E-04	--
	Inside Tubes	--	--	--	4.7622E-02
	Interstitial	--	--	--	5.4618E-01
	Total	3.0214E-01	1.1135E-02	9.2929E-02	5.9380E-01
16×16	Fuel	2.9840E-01	--	--	--
	Gap	--	1.2993E-02	--	--
	Clad	--	--	1.0086E-01	--
	Guide Tube	--	--	5.4230E-03	--
	Instrument Tube	--	--	1.3558E-03	--
	Inside Tubes	--	--	--	6.7633E-02
	Interstitial	--	--	--	5.1334E-01
	Total	2.9840E-01	1.2993E-02	1.0763E-01	5.8097E-01
17×17	Fuel	3.0346E-01	--	--	--
	Gap	--	1.2613E-02	--	--
	Clad	--	--	9.2078E-02	--
	Guide Tube	--	--	7.3113E-03	--
	Instrument Tube	--	--	3.0464E-04	--
	Inside Tubes	--	--	--	5.4568E-02
	Interstitial	--	--	--	5.2967E-01
	Total	3.0346E-01	1.2613E-02	9.9694E-02	5.8424E-01

Table 5.3-2 Directly Loaded Homogenized Fuel Elemental Densities

Element	Density [atom/b-cm]			
	14x14	**15x15**	**16x16**	**17x17**
Cr	7.38366E-06	7.04970E-06	8.16503E-06	7.56288E-06
Fe	1.37490E-05	1.31272E-05	1.52040E-05	1.40828E-05
Hf	2.15094E-07	2.05366E-07	2.37856E-07	2.20315E-07
Ni	6.54113E-07	6.24528E-07	7.23334E-07	6.69990E-07
O	1.46478E-02	1.40545E-02	1.38846E-02	1.40137E-02
Sn	4.85117E-05	4.63175E-05	5.36454E-05	4.96892E-05
U	7.31477E-03	7.01858E-03	6.93178E-03	6.99729E-03
Zr	4.12775E-03	3.94106E-03	4.56457E-03	4.22794E-03

Table 5.3-3 Directly Loaded Fuel Assembly Activated Hardware Region Homogenization

Reference Fuel Type	Region	Mass SS [kg/assy]	SS Volume [cm^3/assy]	Height [cm]	Volume [cm^3/assy]	Volume Fraction
14×14	Lower Nozzle	7.89	9.9659E+02	8.6944	3.3804E+03	2.9482E-01
	Upper Plenum	8.05	1.0164E+03	22.8016	8.8653E+03	1.1465E-01
	Upper Nozzle	9.89	1.2487E+03	8.8900	3.4564E+03	3.6128E-01
15×15	Lower Nozzle	5.68	7.1717E+02	10.7607	4.9559E+03	1.4471E-01
	Upper Plenum	4.12	5.2020E+02	21.2941	9.8070E+03	5.3044E-02
	Upper Nozzle	11.84	1.4949E+03	8.8392	4.0709E+03	3.6723E-01
16×16	Lower Nozzle	5.40	6.8182E+02	11.9685	5.0661E+03	1.3458E-01
	Upper Plenum	10.70	1.3510E+03	25.1358	1.0640E+04	1.2698E-01
	Upper Nozzle	9.50	1.1995E+03	17.3253	7.3336E+03	1.6356E-01
17×17	Lower Nozzle	6.31	7.9634E+02	10.4324	4.7831E+03	1.6649E-01
	Upper Plenum	5.41	6.8308E+02	25.6684	1.1768E+04	5.8043E-02
	Upper Nozzle	7.85	9.9116E+02	8.8392	4.0526E+03	2.4457E-01

Table 5.3-4 Directly Loaded Fuel Assembly Zircaloy Hardware Region Homogenization

Reference Fuel Type	Region	Mass Zirc [kg/assy]	Zirc Volume [cm^3/assy]	Height [cm]	Volume [cm^3/assy]	Volume Fraction
14×14	Lower Nozzle	1.84	2.8103E+02	8.6944	3.3804E+03	8.3137E-02
	Upper Plenum	4.98	7.6064E+02	22.8016	8.8653E+03	8.5800E-02
15×15	Lower Nozzle	2.10	3.2029E+02	10.7607	4.9559E+03	6.4628E-02
	Upper Plenum	6.99	1.0670E+03	21.2941	9.8070E+03	1.0880E-01
16×16	Lower Nozzle	2.59	3.9492E+02	11.9685	5.0661E+03	7.7953E-02
	Upper Plenum	7.97	1.2170E+03	25.1358	1.0640E+04	1.1439E-01
17×17	Lower Nozzle	2.79	4.2540E+02	10.4324	4.7831E+03	8.8938E-02
	Upper Plenum	6.85	1.0462E+03	25.6684	1.1768E+04	8.8896E-02

Table 5.3-5 Regional Densities for Directly Loaded Cask Structural and Shield Materials

Material	Element	Density [atom/b-cm]
Stainless Steel	Cr	1.65112E-02
	Fe	6.31986E-02
	Ni	6.50094E-03
Zircaloy	Cr	7.58615E-05
	Fe	1.41261E-04
	Hf	2.20993E-06
	Ni	6.72052E-06
	O	2.46540E-04
	Sn	4.98421E-04
	Zr	4.24095E-02
Aluminum	Al	6.02626E-02
Lead	Pb	3.20871E-02
NS-4-FR	Al	7.80000E-03
	B	4.27500E-04
	C	2.26000E-02
	H	5.85000E-02
	N	1.39000E-03
	O	2.61000E-02
Heat Fin	Cu	3.62309E-02
	Fe	3.61117E-02
	Cr	9.43448E-03
	Ni	3.71464E-03
Balsa	C	2.78553E-03
	H	4.64261E-03
	O	2.32135E-03

Table 5.3-6 Canistered Fuel and GTCC Material Compositions

Zone/Material	Density (g/cc)	Nuclides	Density (atom/b-cm)
Middle Fuel Zone			
UO_2	2.2769	^{234}U	2.79304E-07
		^{235}U	3.65635E-05
		^{238}U	5.04141E-03
		O	1.01565E-02
Zircaloy	0.6417	Zr	4.23638E-03
SS304	0.3420	Cr	7.52598E-04
		Mn	7.49779E-05
		Fe	2.56318E-03
		Ni	3.33394E-04
Aluminum	0.1091	Al	2.43503E-03
B_4C	0.0101	^{10}B	8.76394E-05
		^{11}B	3.52760E-04
		C	1.10100E-04
Middle Basket/Disk Zone			
SS304	0.9543	Cr	2.10001E-03
		Mn	2.09215E-04
		Fe	7.15218E-03
		Ni	9.30286E-04
Aluminum	0.2920	Al	6.51722E-03
Top Fuel/Basket Zone			
UO_2	2.2769	^{234}U	2.79304E-07
		^{235}U	3.65635E-05
		^{238}U	5.04141E-03
		O	1.01565E-02
Zircaloy		Zr	4.04558E-03
SS304	0.3084	Cr	6.78658E-04
		Mn	6.76116E-05
		Fe	2.31136E-03
		Ni	3.00639E-04
Aluminum	0.0622	Al	1.38826E-03
B_4C	0.0101	^{10}B	8.76394E-05
		^{11}B	3.52760E-04
		C	1.10100E-04
Top Plenum Zone			
Zircaloy	0.5718	Zr	3.77491E-03
SS304	.04821	Cr	1.48517E-03
		Mn	1.05692E-04
		Fe	3.61319E-03
		Ni	4.69968E-04

Table 5.3-6 Canistered Fuel and GTCC Material Compositions (Continued)

Material/Zone	Density (g/cc)	Nuclides	Density (atom/b-cm)
Top End Fitting Zone			
SS304	0.6749	Cr	1.74286E-02
		Mn	1.47961E-04
		Fe	5.05816E-03
		Ni	6.57917E-04
Bottom Fuel/Basket Zone			
UO_2	2.2769	^{234}U	2.79304E-07
		^{235}U	3.65635E-05
		^{238}U	5.04141E-03
		O	1.01565E-02
Zircaloy	0.6128	Zr	4.04558E-03
SS304	0.2350	Cr	6.49170E-04
		Mn	6.46739E-05
		Fe	2.21093E-03
		Ni	2.87577E-04
Aluminum	0.0622	Al	1.38826E-03
B4C	0.0101	^{10}B	8.76394E-05
		^{11}B	3.52760E-04
		C	1.10100E-04
Bottom Plenum Zone			
Zircaloy	0.6128	Zr	4.0455E-03
SS304	1.0529	Cr	2.31699E-03
		Mn	2.30831E-03
		Fe	7.89115E-03
		Ni	1.02640E-03
Bottom Endfitting Zone			
SS304	0.9664	Cr	2.12664E-03
		Mn	2.11867E-04
		Fe	7.24287E-03
		Ni	9.42082E-04
GTCC Waste and Container			
SS304	3.29	Cr	7.2399E-03
		Mn	7.2128E-04
		Fe	2.4658E-02
		Ni	3.2072E-03

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