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2.1 Spent Fuel To Be Stored

The NAC-STC has been designed to store Westinghouse 14 x 14, 15 x 15, 17 x 17, 17 x 17 OFA, Combustion Engineering 14 x 14, 15 x 15, 17 x 17 or Babcock & Wilcox 15 x 15 and 17 x 17 PWR fuel assemblies. The Westinghouse 15 x 15 or 17 x 17 fuel assemblies have been shown to be the bounding standard PWR fuel assemblies (DOE/ET/47912-4, NEDO-10048-2, WCAP-10741). The more limiting source term from two combinations of burnup and cool time for each of the bounding assemblies is used in the shielding and confinement analyses. Both of the bounding assemblies are evaluated for a source term based on an initial enrichment of 3.7 weight percent (w/o) uranium-235, a maximum burnup of 40,000 megawatt days per metric ton of uranium (MWD/MTU), and a minimum cool time of 6.5 years after reactor discharge; and a source term based on an initial enrichment of 4.0 w/o uranium-235, a maximum burnup of 45,000 MWD/MTU, and a minimum cool time of 10 years. Lower enrichments (less than 4.2 w/o) are used to develop the source terms because they are the minimum expected enrichments for which these burnups are likely to be achieved. The lower enrichments result in more limiting source terms for a given burnup.

The maximum decay heat load used for the design basis fuel assemblies is 0.85 kilowatt (kW) per assembly resulting in a maximum decay heat load for the NAC-STC of 22.1 kW. For assemblies that exceed the 0.85 kW decay heat limit, a reduction in burnup or an increase in cool time will be necessary. Thermal analyses in this report are based on a decay heat load of 0.85 kW per assembly.

The type, form and quantity of the bounding design basis fuel assemblies are summarized in Table 2.1-1. The physical characteristics of the design basis Westinghouse 15 x 15 and Westinghouse 17 x 17 fuel assemblies are provided in Table 2.1-2. The design basis decay heat and radiation source strengths are listed in Table 2.1-3.

The SAS2 (Hermann, 1990) sequence of the SCALE-4 system, including ORIGEN-S (Hermann, 1989) was used to calculate the neutron and gamma source terms. The ORIGEN-S computer code also calculates the gamma and neutron spectra.

The concentration of cobalt-60 in the stainless steel end-fittings was also evaluated using the SAS2 sequence. The amount of cobalt-60 in the upper and lower end-fittings was determined based on the mass of each of the end-fittings (Table 2.1-2).

The purpose of the analyses in this report is to demonstrate the adequacy of the NAC-STC for the storage of all of the design basis fuel assemblies (Table 2.1-1). Each of the nuclear analyses evaluates the design basis fuel assembly that is the most limiting for the particular analysis.

2.1.1 Bounding Fuel Evaluation

NAC-STC allowable neutron and gamma sources are shown in Figure 2.1-1, based on the 6.5 year, 40,000 MWD/MTU assembly. The graph also shows that the neutrons can be increased to the 10-year, 45,000 MWD/MTU amount (6.848×10^9 neutrons/sec) provided the fuel gammas remain at or below the 10-year, 45,000 MWD/MTU values (4.397×10^{16} MeV/second). These source terms have been calculated for the Westinghouse 17 x 17 fuel assembly with a conservative uranium loading of 469 kgU. A report entitled, "Fuel Inventory and Afterheat Power Studies of Uranium-Fueled Pressurized Water Reactor Fuel Assemblies Using the SAS2 and ORIGEN-S Modules of Scale with an ENDF/B-V Updated Cross Section Library" (NUREG/CR-2397) concludes that the Westinghouse 17 x 17 fuel assembly is the bouding assembly of the Westinghouse, Babcock & Wilcox and Combustion Engineering assemblies that were evaluated.

The maximum fuel assembly enrichment is 4.2 percent for the NAC-STC (except for 17x17 OFA at 4.1 percent). The burnup and cool time must be limited such that the following criteria are satisfied:

1. Maximum assembly decay heat rate ≤ 0.85 kW.

Table 2.1-1 Type, Form, Quantity and Potential Sources of the Design Basis Fuels (continued)

Fuel Type (3)	- PWR, Westinghouse 15 x 15
	- 4.2 w/o U^{235} maximum initial enrichment (3.7 w/o U^{235} conservatively used for shielding and radiological calculations)
	- 40,000 MWD/MTU maximum burnup
	- 6.5-year cool time after reactor discharge
	- 0.85 kW per assembly maximum decay heat, 22.1 kW per cask
Fuel Form	- Intact assemblies
Quantity	- 26 design basis fuel assemblies per cask
Sources of Fuel	- Commercial PWR nuclear power reactors

Fuel Type (4)	- PWR, Westinghouse 15 x 15
	- 4.2 w/o U^{235} maximum initial enrichment (4.0 w/o U^{235} conservatively used for shielding and radiological calculations)
	- 45,000 MWD/MTU maximum burnup
	- 10-year cool time after reactor discharge
	- 0.85 kW per assembly maximum decay heat, 22.1 kW per cask
Fuel Form	- Intact assemblies
Quantity	- 26 design basis fuel assemblies per cask
Sources of Fuel	- Commercial PWR nuclear power reactors

Table 2.1-2 Design Basis Fuel - Physical Parameters

Parameter	Value	
Assembly Rod Array	17 x 17	15 x 15
Assembly Weight, lb	1,467	1,440
Assembly Length, in	160	160
Active Fuel Length, in	144	144
No. of Fuel Rods	264	204
Rod Pitch, in	0.496	0.563
Cladding Material	Zircaloy-4	Zircaloy-4
Rod Diameter, in	0.374	0.422
Cladding Thickness, in	0.0225	0.0243
Pellet Diameter, in	0.3225	0.3659
Pellet Length, in	0.53	0.60
Pellet Material	UO_2 (sintered)	UO_2 (sintered)
Maximum Fuel Rod Pressure, psia	1,462	1,462
Theoretical Density, percent	95	95
Maximum Initial Enrichment, w/o U^{235}	4.2*	4.2*
Design Basis Burnup, MWD/MTU	40,000**	40,000**
Maximum Weight of U, kg	464	469
Maximum Weight of UO_2, kg	526.4	532.1
Upper End-Fitting, kg/asbl	6.9	6.8
Lower End-Fitting, kg/asbl	5.9	5.7
Plenum Springs, kg/asbl	3.42	3.8
In-Core Grid Spacers, kg/asbl	5.44	8.05

* 3.7 w/o conservatively used for shielding and radiological analyses for 40,000 MWD/MTU burnup. 4.0 w/o conservatively used for shielding and radiological analyses for 45,000 MWD/MTU burnup.

** The design basis fuel may have burnups as high as 45,000 MWD/MTU, but a minimum cool time of 10 years is required.

Table 6.2-2 Fuel Assembly Class Data

Assembly Class	14x14	15x15	16x16	17x17	17x17 (OFA)
Fuel Form	Intact fuel UO_2 pellets	Intact fuel UO_2 pellets	Intact fuel UO_2 pellets	Intact fuel UO_2 pellets	Intact fuel UO_2 pellets
Cladding Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Maximum Uranium Content (kg/assembly)	407	469	426	464	426
Maximum Initial Enrichment (w/o U^{235})	4.2	4.2	4.2	4.2	4.1
Cross-Section (in)					
minimum	7.76	8.20	8.10	8.43	8.43
maximum	8.11	8.54	8.14	8.54	8.43
No. of Fuel Rods Per Assembly					
minimum	176	204	236	264	264
maximum	179	216	236	264	264
Fuel Rod OD (in)					
minimum	0.422	0.418	0.382	0.374	0.360
maximum	0.440	0.430	0.382	0.379	0.360
Min. Clad Thickness (in)	0.023	0.024	0.025	0.023	0.023
Fuel Pellet Diameter Range (in)					
Minimum	0.344	0.358	0.325	0.3225	0.3088
Maximum	0.377	0.390	0.325	0.3232	0.3088
Maximum Active Fuel Length (in)	145.20	144	150	144	144

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6.2 Package Fuel Loading

The NAC-STC can safely transport 26 PWR fuel assemblies. The cask design basis fuel assemblies are the Westinghouse 17 x 17 fuel assembly or the Westinghouse 15 x 15 fuel assembly. As described in Section 6.4.2.1, the design basis Westinghouse 15 x 15 assembly is more reactive than the design basis 17 x 17 assembly. Thus, the 15 x 15 assembly is used in the criticality calculations for the NAC-STC. The design basis fuel assembly characteristics are presented in Table 6.2-1. The Westinghouse 15 x 15 assembly also bounds the Westinghouse 14 x 14 assembly, the Combustion Engineering 14 x 14, 15 x 15 and 16 x 16 assemblies and the Babcock & Wilcox 15 x 15 and 17 x 17 assemblies (DOE/ET/47912-4, NEDO-10048-2, WCAP-10741). An additional evaluation of the Westinghouse 17x17 OFA design in the NAC-STC basket was performed and was shown to be slightly more reactive than the Westinghouse 15x15 fuel assembly. In the case of the OFA design, the maximum allowable enrichment to allow placement in the NAC-STC has been determined to be 4.1 w/o U^{235}. The major fuel classes to be stored in the NAC-STC are presented in Table 6.2-2. Fuel assemblies with zero burnup are used in these analyses. The fresh fuel assumption is conservative because the fuel becomes less reactive as burnup increases.

Table 6.2-1 Characteristics of Design Basis Westinghouse Fuel Assemblies

Parameter	Value	
Assembly Rod Array	15 x 15	17 x 17
Assembly Weight, lbs	1,440	1,467
Fuel Rod Length, in	160	160
Active Fuel Length, in	144	144
Number of Fuel Rods/Assembly	204	264
Fuel Rod Diameter, in	0.422	0.374
Cladding Material	Zircaloy-4	Zircaloy-4
Cladding Thickness, in	0.0243	0.0225
Pellet Diameter, in	0.3659	0.3225
Fuel Cell Pitch, in	0.563	0.496
Pellet Material	UO_2 (sintered)	UO_2 (sintered)
Theoretical Density, percent	95	95
Maximum Initial Enrichment, w/o U^{235}	4.2	4.2
Maximum Uranium Content kgU/assembly	469	464

6.4 Criticality Calculation

This section demonstrates that the NAC-STC cask with the design basis payload meets the criticality requirements of 10 CFR 72 section 72.124 and ANSI/ANS-8.17 and ANSI/ANS-59.7.

Compliance with the requirement of subcriticality (including safety margin and uncertainty factors) in 10 CFR 72 section 72.124 is accomplished by modelling the STC cask according to ANSI/ANS-8.17. The appendix of ANSI/ANS-8.17 states "Paragraph 4.4 requires that consideration be given to normal and credible abnormal conditions and to related uncertainties, including design tolerances, associated with controlled parameters". Fuel rod parameters, fuel unit configuration and fuel array spacing will not be varied during criticality evaluations since the structural analysis in Chapter 2 has shown that no fuel structural damage will occur during a hypothetical accident situation. Moderator conditions will be tested by varying the density of water inside the cask. Reflector/moderator conditions are tested by adjusting water density outside the cask and cask center-to-center spacing (pitch) to determine optimal moderation.

For the hypothetical accident situation, a damaged package is defined as the cask subject to a fire accident followed by moderator intrusion. Under these conditions the cask containment is maintained, and thus, the cavity remains dry. However, the radial neutron shield material is assumed to be lost due to fire and is replaced by external moderator. The fuel remains intact during this scenario. For consistencies sake an evaluation of the cask cavity flooded and loss of neutron shield was done to show that even under this scenario the cask remains subcritical.

6.4.1 Fuel Loading Optimization

The NAC-STC is designed to transport 26 PWR assemblies with an initial enrichment of 4.2 w/o uranium-235. The analyses presented show that the maximum fuel loading along with the most reactive configuration have been analyzed. Loading of fresh fuel into the cask under water with no dissolved boron, and with the cask surrounded by water, is assumed to ensure that the maximum credible reactivity is simulated.

6.4.2 Criticality Results

6.4.2.1 Most Reactive Assembly

A simplified KENO-Va calculation of the two design basis assemblies given in Table 6.2-1 is performed to determine the most reactive. In this simplified model, a unit cell of the NAC-STC basket with the steel and aluminum webbing properly spaced axially is described. Reflecting boundary conditions are imposed on the sides, top and bottom simulating an infinite array of basket cells. In one case, the 15x15 fuel assembly is modelled, and in the other, the 17x17 fuel assembly is modelled. Both fuel assemblies are at the same fuel density, 95% of theoretical and a 4.2 w/o U^{235} initial enrichment. The k-infinity of the 15x15 fuel assembly in the NAC-STC basket is 0.9526 ± 0.0024, and the k-infinity of the 17x17 fuel assembly in the NAC-STC basket is 0.9455 ± 0.0025. Thus, the 15x15 is the most limiting assembly of the two design basis assemblies and is used in subsequent cask criticality analysis.

An additional evaluation of the Westinghouse 17x17 OFA assembly loaded into the NAC-STC basket gave a k-infinity of 0.9575 ± .0024. Lowering the enrichment of the OFA design to 4.1 w/o U^{235} gave a k-infinity of 0.9504 ± .0024 which is below the 15x15 fuel assembly design at 4.2 w/o U^{235} initial enrichment. Thus, in the case of the OFA fuel assembly, the maximum allowable enrichment to allow placement in the NAC-STC is 4.1 w/o U^{235}.

6.4.2.2 Normal Conditions

Criticality results under normal conditions include variations in moderator density from 1.0 g/cc to 0.0001 g/cc (dry) and cask center-to-center spacing from 249.358 cm (touching) to 400 cm. The results are shown in Tables 6.4-1 and 6.4-2. Table 6.4-1 shows the expected reactivity conditions during loading, i.e. wet inside and outside, as well as variation in moderator density due to draining and drying.

Also, shown in Table 6.4-1 is the reactivity penalty from assuming 75% of nominal boron. This is approximately a 2% Δk penalty. Table 6.4-1 shows that cask reactivity is insensitive to variations in cask center-to-center spacing, but that the touching condition is the most reactive. This results in a k_{eff} of 0.9266 ± 0.0023. The CSAS25 input and output for this case is shown in Figures 6.6-1 and 6.6-2, respectively. Simultaneous variation in moderator density inside and outside the cask shows a monotonic decrease in reactivity. There appears to be no optimum at low density conditions. The k_{eff} in the dry situation is 0.4420 ± 0.0009.

Table 6.4-2 shows the expected reactivity conditions during normal storage, i.e. dry inside and possibly wet outside. When the cask cavity is dry, k_{eff} of the package is very low and is insensitive to variations of moderator density outside and cask center-to-center spacing. The maximum k_{eff} for this situation is 0.4470 ± 0.0014.

Including statistical and method uncertainties, all results for the normal condition are below the 0.95 NRC criticality safety limit. Thus, compliance with 10 CFR 72.124 under normal conditions is demonstrated.

6.4.2.3 Hypothetical Accident Conditions

Criticality results under hypothetical accident conditions include variations in moderator density from 1.0 g/cc to 0.0001 g/cc (dry) as well as cask center-to-center spacing from 249.358 cm (touching) to 300 cm. The results are shown in Table 6.4-3. Under accident conditions, moderator is allowed in the neutron shield regions and outside the cask. Again, with the cask cavity dry, the k_{eff} of the package is very low and is insensitive to moderator density and cask spacing variation. The maximum k_{eff} for this situation is 0.5032 ± 0.0014. The CSAS25 input and output for this case is shown in Figure 6.6-3 and 6.6-4, respectively. Even if moderator is allowed in the cavity under accidents, the k_{eff} is 0.9096 ± 0.0047, which is still below 0.95 with uncertainties.

Including statistical and method uncertainties, all results for the accident condition are well below the 0.95 NRC criticality safety limit. Thus, compliance with 10 CFR 72.124 under accident conditions.

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9.1 Fabrication Requirements and Acceptance Tests

This section identifies the fabrication inspection and acceptance requirements, tests, and acceptance criteria established for the NAC-STC to verify, prior to acceptance and marking, that the packaging has been fabricated, assembled, tested, inspected and accepted in accordance with the applicable NAC-STC License Drawings (Section 1.5.2) and the other requirements of this application.

9.1.1 Visual and Non-Destructive Examination (NDE) Inspection Programs

The primary confinement components of the NAC-STC will be fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1 requirements. The nonconfinement components of the NAC-STC, except the fuel basket assembly, will be fabricated per ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 requirements. The fuel basket assembly components will be fabricated per ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. The fabrication and welding requirements for the NAC-STC cask are shown on the NAC-STC License Drawings, Section 1.5.2 of this application.

In the fabrication of the NAC-STC, welding is the method used to join the plates and forgings that comprise the cask body. The welding procedure qualifications and the welding performance qualifications for the fabrication of the NAC-STC will be in accordance with Part QW-Welding, Section IX, ASME Boiler and Pressure Vessel Code. All exposed welds on the NAC-STC will be ground flush to the base metal or to a smooth fillet.

Each selected fabricator for the NAC-STC will establish a detailed written weld inspection plan, in accordance with an approved Quality Assurance program, of visual (VT), dye penetrant (PT), ultrasonic (UT), and radiographic (RT) weld examinations to be performed during fabrication and prior to acceptance of the cask. The weld inspection plan will identify the welds to be examined, the sequence of the examinations, the type of examination method to be used, and the criteria for acceptance of the weld in accordance with the applicable sections of the ASME Boiler and Pressure Vessel Code (ASME Code).

The finished surfaces of all welds on the NAC-STC will be visually examined in accordance with ASME Code, Section V, Article 9, to verify that the components are assembled in accordance with the License Drawings (Section 1.5.2) and that the components are free of nicks, gouges, or other damage. The acceptance criteria for the visually examined welds will be in accordance with ASME Code, Section VIII, Division 1, UW-35 and UW-36.

The NAC-STC primary confinement boundary welds shall be radiographic (RT) examined per ASME Code, Section V, Article 2, to ensure that the welds do not contain any unacceptable imperfections as noted on the NAC-STC License Drawings (Section 1.5.2). Acceptance criteria for RT examinations of primary confinement welds shall be in accordance with ASME Code Section III, Subsection NB, Article NB-5320. Unacceptable imperfections such as a crack, a zone of incomplete fusion or penetration, elongated indications with lengths greater than specified limits, and rounded indications in excess of the limits specified in ASME Code Section III, Division 1, Appendix VI, shall be cause for rejection of the weld. Repaired welds shall be reexamined in accordance with the original acceptance criteria.

The circumferential and longitudinal welds of the outer shell assembly, and the connection welds of the outer shell assembly to the upper forging shall be radiographic (RT) examined per the ASME Code, Section V, Article 2, to ensure that the welds do not contain any unacceptable imperfections. Acceptance criteria for RT examined outer shell welds shall be in accordance with ASME Code, Section VIII, Division 1, UW-51 and repair of unacceptable defects shall be in accordance with UW-38. Repaired welds shall be re-examined in accordance with the original acceptance criteria.

The final closure welds of the bottom ring forging to the bottom forging and to the outer shell, and the closure weld of the bottom plate to the bottom ring forging shall be ultrasonic (UT) examined per ASME Code, Section V, Article 5, to ensure that the welds do not contain unacceptable imperfections. Acceptance criteria for

UT examined welds shall be in accordance with ASME Code, Section VIII, Division 1, UW-53 and Appendix 12, and repair of unacceptable defects shall be in accordance with UW-38. Repaired welds shall be re-examined in accordance with the original acceptance criteria.

Welds that are marked "PT Root and Final Pass" on the NAC-STC License Drawings (Section 1.5.2) will be liquid penetrant (PT) examined per ASME Code, Section V, Article 6. The liquid penetrant (PT) examination method is used to detect discontinuities, such as cracks, seams, laps, laminations and porosity, that are open to the surface of nonporous metals. Acceptance criteria for these liquid penetrant examined welds shall be in accordance with ASME Code Section III, Subsection NB, Article NB-5350. All other noncontainment welds that are marked "PT" on the NAC-STC License Drawings (Section 1.3.2) will be liquid penetrant examined per ASME Code, Section V, Article 6. Acceptance criteria for these liquid penetrant examined welds shall be in accordance with ASME Code Section VIII, Division 1, Appendix 8. Unacceptable indications shall be cause for rejection of the welds. Rejected welds shall be repaired in accordance with approved weld repair procedures prepared in accordance with the applicable provisions of the ASME Code, Section III, NB-4450 for containment welds and Section VIII for non-containment welds. Repaired welds shall be re-examined in accordance with the original acceptance criteria.

All weld inspections shall be performed by qualified personnel in accordance with written procedures. Inspection personnel shall be qualified in accordance with SNT-TC-1A, "Personnel Qualifications and Certification in Nondestructive Testing", current revision at time of fabrication as specified by the ASME Code, Section III, Division 1, Article - NB-5000, Paragraph NB-5520.

9.1.2 Structural and Pressure Tests

9.1.2.1 Lifting Trunnion Load Testing

Each of the two pairs of the cask lifting trunnions shall be load tested in accordance with the requirements of ANSI N14.6 "Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More for Nuclear

Materials." The load test will be performed for one pair and repeated for the other pair. The load test shall be performed in accordance with written procedures prepared by the fabricator and approved by NAC.

The lifting trunnion load test shall consist of applying a vertical load of 375,000 pounds (170.1 MT), which is 150 percent of the maximum service load, +5/-0 percent, to diametrically opposite trunnion pairs. The load will be applied in a vertical direction and equally distributed between the two trunnions and over the length of 2.25 inches of the trunnion/lifting yoke interface areas. The inner and outer lids will be bolted in place for the test. The test may be carried out by the use of calibrated hydraulic rams combined with a load spreading beam, or the cask lifting yoke, attached to the trunnion pair. The load shall be held for a minimum of 10 minutes.

Following completion of each lifting trunnion load test, all trunnion welds and load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Inspections utilizing liquid penetrant examination shall be performed in accordance with "ASME Boiler and Pressure Vessel Code," Section V, Articles 1, 6, 7, 24 and 25. Liquid penetrant acceptance standards shall be as indicated in paragraph NB-5350 of the "ASME Boiler and Pressure Vessel Code," Section III, Division I.

Any evidence of permanent deformation, cracking, galling of load bearing surfaces or unacceptable dye penetrant results shall be cause for rejection of the trunnion or related welds.

9.1.2.2 Load Testing of the Rotation Trunnion Recesses

The rotation trunnion recesses at the lower end of the cask shall be load tested. The load test shall be performed in accordance with written procedures prepared by the fabricator and approved by NAC.

The load test for the recesses shall consist of applying a vertical load of 375,000 pounds (170.1 MT), +5/-0 percent, to the trunnion pair. The load will be applied in a vertical orientation and equally distributed between the two rotation trunnion recesses.

Following completion of the rotation trunnion recesses load test, all trunnion welds and load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Inspections utilizing dye penetrant examination shall be performed in accordance with Section V, Articles 1, 6, 7, 24 and 25 of the "ASME Boiler and Pressure Vessel Code." Liquid penetrant acceptance standards shall be as indicated in paragraph NB-5350 of the "ASME Boiler and Pressure Vessel Code," Section III, Division 1.

Any evidence of permanent deformation, cracking, galling of load bearing surfaces, or unacceptable dye penetrant results shall be cause for rejection of the rotation trunnion recesses or related welds.

9.1.2.3 Hydrostatic Testing

A hydrostatic test shall be performed on the cask confinement vessel - the welded top forging, inner shell, inner shell rings, and bottom inner forging - prior to welding of the outer shell and pouring of the lead, in accordance with the "ASME Boiler and Pressure Vessel Code," Section III, Subsection NB, Article NB6000. The test pressure will be 60 psig, +5/-0 psig which is 150 percent of the Maximum Normal Operating Pressure (MNOP). This test shall be performed in accordance with procedures prepared by the fabricator and approved by NAC.

The hydrostatic test system components, although not part of the cask confinement boundary, will be visually inspected prior to the start of each hydrostatic test. Leakage from valves or connections will be corrected prior to the start of the actual test.

The hydrostatic test pressure gauge will have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for a minimum of 10 minutes for each test after which a visual inspection shall be performed to detect any evidence of leakage. Any evidence of leakage after the minimum hold period will be cause for rejection.

After completion of the ASME Code hydrostatic test, the cask confinement vessel will be dried and prepared for visual and dye penetrant inspections as appropriate. The cask confinement vessel shall be visually inspected. All accessible welds within the cavity shall be liquid penetrant inspected. Any evidence of cracking or permanent deformation is cause for rejection.

9.1.2.4 Pneumatic Bubble Testing of the Neutron Shield Tank

A pneumatic bubble test of the neutron shield tank will be performed in accordance with Section V, Article 10, Appendix I, of the ASME Boiler and Pressure Vessel Code following final closure welding of the bottom closure plates. The pneumatic test pressure shall be 12.5 + 1.5/-0 psig, which is 125 percent of the relief valve set pressure. The test shall be performed in accordance with approved written procedures.

During the test, the two relief valves on the neutron shield tank will be removed. One of the relief valves threaded connections will be used for connection of the air pressure line and test pressure gauge. The other relief valve connection will be plugged with a threaded plug.

Following introduction of pressurized air into the neutron shield, a 15 minute minimum soak time will be required. Following completion of the soak time, approved soap bubble solution will be applied to all fin to shell, shell to end plate, and end plate to outer shell welds. The acceptance criteria for the bubble test will be no air leakage from any tested weld as indicated by continuous bubbling of the solution. If air leakage is indicated, the weld shall be repaired in accordance with approved weld repair procedures and the pneumatic bubble test shall be repeated until no acceptance air leakage is observed.

9.1.3 Leak Tests

Leak tests shall be performed on the storage confinement boundary seals to provide assurance that radioactive material will not be released to the environment. The leak tests to be performed shall be specified in the fabrication specification. The leak tests shall be performed in accordance with procedures prepared by the fabricator and approved by NAC, in accordance with "ASME Boiler and Pressure Vessel Code," Section V, Article 10, Appendix V. The seals to be tested are the interlid and pressure port cover o-rings, the inner lid inner o-ring, the vent port coverplate inner o-ring, the drain port coverplate inner o-ring and the outer lid o-ring.

The series of leak tests to be performed will use helium as a tracer gas and a Helium Mass Spectrometer Leak Detection System, which is capable of being calibrated to a sensitivity of at least 1.1 x 10^{-5} standard cubic centimeters/second. Although confinement is ensured by a minimum of at least two seals, helium tests will be performed to ensure that the leakage past a single seal will not exceed 4.42 x 10^{-5} standard cubic centimeter/second. For each test, the closure will be assembled and torqued to the values specified in Chapter 8. Helium gas will be introduced into the volume to be tested to a minimum pressure of 10 psig. All potential leakage paths shall be tested.

When testing using the helium sniffer method, the probe velocity will not exceed 2 centimeters/second and the probe will be maintained within 3 millimeters of the area of the seal being tested.

The test procedures, equipment and methods will be selected to ensure that the minimum sensitivity of each leak test meets or exceeds 1.1 x 10^{-5} standard cubic centimeters/second.

An indicated leak rate past any confinement seal or closure that exceeds 2.2 x 10^{-5} standard cubic centimeters/second shall be cause for rejection of the item being tested. Seal replacement or other corrective action shall be taken to correct the leak. Then, the item will be retested and inspected in accordance with the above test requirements and acceptance standards.

9.1.4 Components Tests

Tests performed on individual components are designed to ensure that the component meets the design requirements for correct and proper operation of the cask system. Acceptance criteria are established based on the function and design requirements of the component being tested.

9.1.4.1 Valves, Relief Valves, and Fluid Transport Devices

There are no valves that are part of the NAC-STC confinement boundary. Quick-disconnect valved nipples (QDVN) are installed in the vent and drain ports in the inner lid to provide access to the cavity. These valved nipples are used to connect ancillary equipment to the cask cavity for filling, draining, drying, backfilling, gas sampling and leak testing operations. A QDVN is also installed in the interseal test port to access the interspace between the two inner lid metallic o-rings. A description of the quick-disconnect valved nipples is provided in Section 9.3.1.

Upon removal of the external valved coupler, the valve closes automatically. The design and selection of the valves is based on identical equipment and procedures used with other NRC-approved storage and transport casks. The vent and drain port valves are sealed using a bolted coverplate fitted with two metallic o-rings. These seals are leak tested each time they are installed.

The interlid and pressure port covers are based on designs approved for other NAC casks. The port covers are fitted with o-ring seals, which are leak tested each time they are installed.

There are no rupture disks on the NAC-STC. Two self-actuating relief valves are installed 180 degrees apart on the external shell of the neutron shield to provide for venting of the shielding material during transport fire accident conditions. The relief valves remain closed and provide no function for the NAC-STC during normal storage operations.

9.1.4.2 Gaskets

The various cask confinement boundary closures are fitted with gaskets to ensure proper confinement of the radioactive material contents of the cask. The gaskets are o-rings of several different types, materials and sizes. These are described on the License Drawings (Section 1.5.2). The o-ring materials used in the NAC-STC confinement boundary are metallic (stainless steel). The NAC-STC also uses non-confinement boundary o-rings that are BTFE material. In addition, the o-rings and mating sealing surfaces are individually leak tested at the fabricator prior to first use. Proper functioning and acceptable leakage rates are verified by leak tests as part of the cask loading operation prior to acceptance and placement of the cask in storage as detailed in Chapter 8.

9.1.4.3 Miscellaneous

There are no miscellaneous components on the NAC-STC.

9.1.5 Tests for Shielding Integrity

9.1.5.1 Gamma Shield Test

A gamma scan test of the steel and lead shielding of the cask shall be performed to verify the shield integrity and the absence of voids in the lead shielding. The test will be performed on the cask body in accordance with written procedures prepared by the fabricator and approved by NAC.

The gamma scan test shall be conducted by continuous scanning or probing over 100 percent of all accessible areas where lead is located using a 3-inch detector and a cobalt-60 source. The source strength shall be of an intensity sufficient to produce a count rate that equals or exceeds three times the background count rate on the external surfaces of the cask. Scan path spacing will be a maximum of 2.5 inches and the scanning speed will be 4.5 feet per minute or less. All probing will be on a 2-inch grid pattern (when using a 3-inch detector) and the specified count time will be 1 minute minimum.

The acceptance criteria for the gamma shield test will be that the shield contains no lead voids that reduce the lead thickness by more than 0.25 inch. The effectiveness of the cask body shall be equivalent to the shield effectiveness of a lead and steel mock-up, where the material thicknesses are equivalent to the minimum thicknesses specified on the License Drawings less 0.25-inch lead. The shielding mock-up will be produced in accordance with written approved procedures using the same fabrication techniques as those approved for the cask.

Measured count rates that exceed those established by the test mock-up shall cause the component to be rejected. The rejected areas/components shall be evaluated to determine the corrective action to be taken. All repaired areas shall be retested prior to acceptance.

9.1.5.2 Neutron Shielding Test

The neutron shielding of the NAC-STC is provided by a solid layer of NS4FR, which is a hard polymer material developed by BISCO Products, Inc. A 5.5-inch layer of this material is located in the annulus formed by the outer shell and the 0.236-inch (6mm) thick neutron shield shell. The neutron shield is divided in sections by the copper/stainless steel fins. A 2-inch thick layer of neutron shielding material is located in both the inner lid and in the cask bottom.

The installation of NS4FR material in the fabrication of the cask is a special process and, as such, procedures will be prepared and qualified to ensure that the mix ratios, mixing method, degassing, pouring, and curing of the material is properly performed. The neutron shielding material is installed into the annulus between the outer shell and the neutron shield shell by pouring it with the cask in an inverted vertical position. During fabrication, samples of the actual material being poured into the annulus will be taken at regular intervals and tested to ensure that the material is properly mixed and poured and that it provides the minimum neutron attenuation required by the shielding analysis. Samples that do not meet the acceptance criteria or that have voids or excessive porosity will cause the section (between adjacent fins) from which the sample was taken to be rejected. Corrective action will be taken to ensure proper installation of the neutron shielding material.

Following final closure of the neutron shield shell, a neutron scan test shall be performed by continuously scanning or probing over 100 percent of the exterior neutron shield surface using a neutron detector and a neutron source. The source strength shall be of an intensity sufficient to produce a count rate that equals or exceeds three times the background counting time on the external surface of the neutron shield shell. Scan path spacing will be a maximum of 2.5 inches and the scanning speed will be 4.5 feet per minute or less. All probing will be on a 2-inch grid pattern and the counting period will be one minute or greater.

The acceptance criteria for the shield test will be that the shield effectiveness of the external neutron shield shall be equal to or greater than the shield effectiveness of a lead/steel/NS4FR mock-up where the NS4FR thickness is equivalent to the minimum neutron shield thickness shown on the License Drawings (Section 1.3.2) less 3 percent. The shielding mock-up will be produced using the same fabrication techniques as those approved for the cask.

Measured neutron count rates that exceed those established by the test mock-up shall cause the component to be rejected. The rejected areas/components of the cask shall be evaluated to determine corrective actions to be taken. Any repaired areas shall be retested to the original acceptance criteria prior to final acceptance.

An additional neutron shield effectiveness test shall be performed on each cask following first fuel loading. See Section 8.1.5.3 for details on the neutron shield effectiveness test procedures and acceptance criteria.

9.1.5.3 Neutron and Gamma Shield Effectiveness Tests

Following first fuel loading, a neutron and gamma shield effectiveness test shall be performed for each cask prior to storage. The test shall be performed with the cask loaded with fuel, drained, vacuum dried and backfilled with helium. The purpose of the test is to document the effectiveness of the neutron and gamma shielding materials. The test shall be performed in accordance with detailed, approved written test procedures.

Calibrated neutron and gamma dose rate meters shall be used to measure the neutron and gamma dose rate at contact with the outer shell of the neutron shield and at 2.3 meters from the surface (equivalent to 2 meters from the sides of the railcar). Dose measurement points shall be established on the external surface of the shell at 30° intervals and at five points along the height of the shield (a total of 60 measuring points). In addition, neutron and gamma dose rate measurements shall be made of the trunnion areas above the neutron shield, at four points below the neutron shield, and at the edges and center of the cask top (outer lid) and cask bottom surfaces. Dose rates at the top and bottom of the cask shall be measured with the transport impact limiters installed. The dose rates measured at contact and at 2.3 meters shall be recorded on the test data sheet, along with the total power of the loaded fuel assemblies; date, time and location of test; identification and calibration of instrumentation; and identification of test engineer and operators.

To allow an evaluation of the measured dose rates to be completed, the burnup and cool time for the actual fuel assemblies loaded into the cask will be determined and recorded. From this fuel history data, the total actual neutron and gamma source terms will be estimated using ORIGEN or similar calculations.

If the measured dose rates exceed the applicable regulatory limits, the licensee shall notify the USNRC. Appropriate corrective measures will be taken, including fuel unloading and correction of the shielding deficiency. Following corrective actions, the test will be reperformed to the original acceptance criteria prior to final acceptance.

9.1.6 Thermal Test

Prior to acceptance at the facility, a thermal test shall be performed on each fabricated packaging to confirm and verify that the fabricated and assembled cask possesses the heat rejection capabilities predicted by the thermal analyses. The thermal test shall be performed in accordance with approved written procedures.

9.1.6.1 Thermal Test Set-up

The thermal test set-up is shown in Figure 9.1-1(a). As depicted, the thermal test shall be performed with the cask positioned horizontally on a test frame. The transport impact limiter or equivalent insulating material shall be installed on each end of the cask to simulate the transport configuration. The cask will be located in a covered building in a still environment. The cask shall be assembled with the basket installed. A thermal test lid with connections for thermocouple leads and electric heater power cables shall be installed in place of the inner lid. The outer lid will not be installed for the test. The thermal test lid will be provided with an o-ring seal capable of containing the containment cavity helium atmosphere.

Electric heaters shall be installed in each fuel tube. The electric heaters will have an active length of between 120 and 150 inches and be capable of generating a minimum of 22 kilowatts (kw). The heaters will be supported in the basket so as to not be in contact with the wall of the fuel tube. The power supplied to the heater will be recorded throughout the test duration.

Calibrated test thermocouples, with an accuracy of ±2°F, will be installed on the cask basket, inner shell, and outer neutron shield shell surfaces. The location of the test thermocouples are shown in Figure 9.1-1. The specific location of the thermocouples are as follows:

TC1 - basket top steel weldment
TC2 - steel disk at cask basket midpoint
TC3 - aluminum disk at cask basket midpoint
TC4 - basket bottom steel weldment
TC5; TC6; TC7; and TC8 - located at 90° intervals on the inner shell surface at cavity midpoint
TC9 - top of inner shell surface at 30-40 inches from top of cavity
TC10 -bottom of inner shell surface at 30 to 40 inches from base of the cavity
TC11; TC12; TC13; and TC14 - located at 90° intervals on the neutron shield shell surface (at fin tip) at cask midpoint.
TC15 - top of neutron shield shell surface (at fin tip) at 30-40 inches from top of neutron shell.
TC16 - bottom of neutron shield shell surface (at fin tip) at 30-40 inches from bottom of neutron shield shell.
TC17 - top of upper forging
TC18 - outer shell surface at centerline of cask bottom face
TC19 - inner fuel tube wall surface near the center of the cask basket
TC20 - ambient temperature of testing area

The output of the test thermocouples will be recorded throughout the test by a strip chart recorder.

9.1.6.2 Test Procedure

With the cask assembled and instrumented as described above, the cask cavity is evacuated and backfilled to 1.0 atmosphere absolute (14.6 psia) with helium. Power will be applied to the heaters to simulate the cask contents. After initiation of power to the heaters, the temperatures of all thermocouples and heater power levels will be monitored and recorded on data sheets at 60 minute intervals. Power will be maintained to the electrical heaters until the cask has reached thermal equilibrium.

For the purpose of the test, thermal equilibrium is defined as being achieved when over two consecutive hours:

$\Delta t_{TC13} \leq 2°F/hr$, and
$\Delta t_{TC3} \leq 2°F/hr$

Based upon the thermal heat-up evaluation, thermal equilibrium should be achieved in approximately five days.

After verification of thermal equilibrium, final temperature measurements will be recorded for all test thermocouples. The final power readings for the electric heaters will also be recorded. The strip chart will be marked to indicate the time of the final cask measurements. The printout of the strip chart recorder and the completed test data sheets will be incorporated into an approved final thermal test report. The test will be determined to be acceptable if the acceptance criteria of Section 8.1.6.3 are met.

If the acceptance criteria are not met, the cask will not be accepted until appropriate corrective actions are completed. Upon completion of corrective actions, the cask shall be retested to the original test requirements and acceptance criteria.

9.1.6.3 Acceptance Criteria

The purpose of the thermal test is to confirm the heat rejection capabilities of the as-built cask are acceptable and correspond to the temperatures calculated by thermal analyses presented in Chapter 3.0 of this application.

Package heat dissipation acceptance testing assures: 1) maximum material temperatures do not exceed material allowables, and that 2) measured temperature gradients are less than the thermal gradients calculated in the package thermal analyses.

The thermal acceptance test is accepted when the following criteria are met:

1) When corrected for physical test boundary conditions and heat load, the following measured temperatures are not exceeded:

TC No.	Location	Temperature °F
TC1	Top Basket Steel Weldment	435
TC3	Aluminum Fin Center	485
TC2	Steel Support Disk Center	495
TC4	Basket Bottom Steel Weldment	475
TC5-TC8	Cask Inner Shell	330
TC11-TC14	Neutron Shield Shell	240
TC17	Cask Top Forging	200
TC18	Cask Bottom	330
TC19	Tube Wall	540

2) The measured temperature gradient across the central steel disk from TC2 to the average of TC5, TC6; TC7 and TC8 is less than 200°F;

3) The measured temperature gradient across the central aluminum fin from TC3 to the average of TC5; TC6; TC7 and TC8 is less than 190°F; and

4) The measured temperature gradient across the cask body as measured by thermocouple pairs TC5-TC13; TC6-TC14; TC7-TC11; and TC8-TC12 are less than 90°F.

9.1.7 Neutron Absorber Tests

9.1.7.1 General

A neutron absorber verification test shall be performed on test samples from each BORAL sheet pour to verify the presence, proper distribution and areal density of neutron absorbing material. The tests shall be performed in accordance with approved written procedures.

9.1.7.2 Preparation of Samples for Real Time Radiograph of BORAL

Detailed written procedures shall be established by the fabricator and approved by NAC to perform real time radiographic tests of samples from each BORAL pour. For each batch, a 2-inch wide sample shall be taken from each end of a sheet. The samples shall be indelibly marked and recorded for identification. A reference BORAL standard for the appropriate B-10 areal density shall be used as the test acceptance standard.

9.1.7.3 Radiographic Test Performance

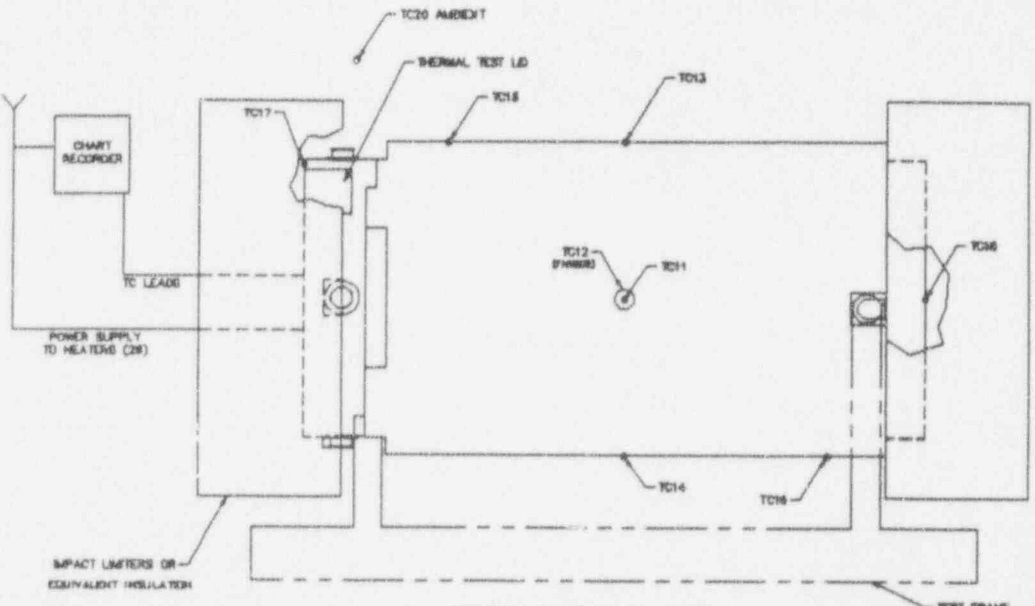
An approved facility with a neutron beam capability shall be selected to perform the described tests. For system calibration, a camera shall be placed in the neutron beam path and the reference BORAL standard plate is placed in a stationary, fixed location between the camera and beamport. A luminance level is then determined at a location near the center of the specimen. The BORAL standard is then replaced by the BORAL test specimen and a luminance level is determined at a location near the center of the specimen.

9.1.7.4 Acceptance Criteria

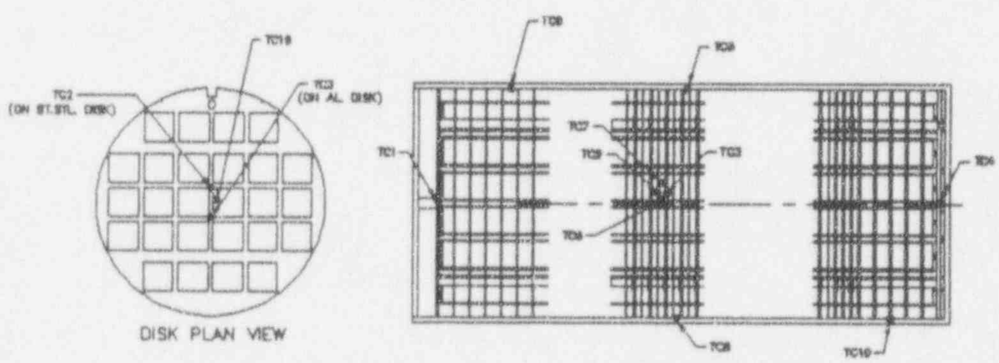
The test shall be considered acceptable if the luminance level determined for each test specimen is equal to or less than that of the BORAL standard. The minimum areal density for BORAL B-10 loading shall be 0.01g/cm^2 B-10.

Any specimen not meeting the acceptance criteria shall be rejected and the sheets from that pour shall be similarly rejected.

Figure 9.1-1 Thermal Test Arrangement



TEST SET-UP AND
(a) EXTERNAL THERMOCOUPLE LOCATIONS



(b) INTERNAL CAVITY AND BASKET THERMOCOUPLE LOCATIONS

The analyses in Chapter 11 demonstrate that the NAC-STC maintains its confinement and safety function for all of the bounding site conditions.

The cask should be stored at a site, which is not adversely affected by nearby industrial, transportation, or military facilities. Any explosion or fire resulting from nearby chemicals, flammable gases, explosives, or munitions must not substantially raise the effective ambient temperature and/or increase the external pressure for the cask environment for an extended length of time. Any projectiles generated by such a fire or explosion must not strike the cask with impact conditions more severe than those of the tornado missiles analyzed in Section 11.2.8. The NAC-STC is limited to use at a site that meets the meteorological, climatological, and diffusion limits listed in Table 12.1-1.

12.1.1.3 Cask Confinement

The NAC-STC has a total of three openings or penetrations into the cask cavity. These are the inner lid, the vent port, and the drain port. The inner lid and each of the ports are sealed using a dual concentric o-ring configuration. For the inner lid and the vent and drain port coverplates, one metallic o-ring forms the primary confinement boundary seal and a second metallic o-ring provides a small region between the two o-rings for testing of the primary confinement boundary seal. The cask will be stored with the outer lid bolted in place with one metallic o-ring acting as the secondary confinement boundary seal. A pressure port is also provided that can be used to continuously monitor the pressure in the interlid volume between the inner and outer lids. An interlid port also penetrates the region between the inner and outer lid. The interlid port is used for testing the outer lid o-ring and for evacuating and backfilling the interlid volume with helium. All confinement seals must demonstrate a leakage rate that does not exceed 2.2×10^{-5} std-cm^3/second. The test procedures to be used after the fuel loading are described in Chapter 7.

The analyses contained in Chapter 11 demonstrate that, under normal, off-normal, and accident conditions, the NAC-STC's confinement function is not compromised.

12.1.1.4 Administrative Controls

The NAC-STC is limited to use at sites that employ administrative controls in accordance with 10 CFR 72.44[c][5].

12.1.2 Bases for Operating Controls and Limits

During the design, manufacture and operation of the NAC-STC, specific controls and limits are enforced. The purpose of this section is to elaborate on the logic involved in selecting these controls and limits.

12.1.2.1 Fuel Characteristic Limit

The NAC-STC is designed to store 26 design basis fuel assemblies as described in Section 2.1, "Spent Fuel to be Stored." Analyses in this report demonstrate the ability of the NAC-STC to safely store design basis fuel under normal operations conditions, off-normal operations conditions and accident event conditions.

Four fuel characteristics are important to the safety functions of the NAC-STC and must be controlled:

1. Initial Enrichment

   Fuel to be stored in the NAC-STC must have an assembly average initial enrichment $\leq$4.2 w/o U^{235}, except for Westinghouse 17x17 OFA where assembly average initial enrichment shall be limited to $\leq$4.1 w/o U^{235}.

2. Burnup

   The maximum burnup is 45,000 MWD/MTU for fuel cooled longer than 10 years. For fuel burnups of 40,000 MWD/MTU or less, the minimum cool time is six and one-half years.

3. Decay Heat

The decay heat of all fuel to be stored in the NAC-STC must not exceed 0.85 kilowatt per assembly; that is, a total of 22.1 kilowatts per cask. For fuel within this limit, the analysis of Section 4.4 documents that the heat rejection capability of the NAC-STC is sufficient to keep the maximum fuel cladding temperatures in a range (below 360°C for 6.5 year cool time), where clad integrity is maintained (PNL-6189/UC-85).

4. Cool Time

As described previously, no fuel may be stored in the NAC-STC unless it has been cooled at least six and one-half years. If the fuel burnup is between 40,000 and 45,000 MWD/MTU, a minimum cool time of ten years is required.

Other aspects considered in selecting fuel characteristic limits include maintaining ALARA dose rates.

Criticality is another concern directly related to the fuel stored. The analyses of Chapter 6 show that k_s is maintained below 0.95 for the most limiting normal operations conditions.

12.1.2.2 Site Limits

The NAC-STC may be sited at any ISFSI site whose natural characteristics meet at least those stated in Section 12.1. These limits are based on 10 CFR 72 requirements and assumed criteria, which provides parameters needed to evaluate the cask for credible operating conditions.

12.1.2.3 Cask Confinement (Leakage) Limits

Limits that apply to the NAC-STC confinement are provided in Section 12.1.1.3. These limits are based on compliance with the allowable leak rates established by ANSI N14.5. Since a repeatable and verifiable test value is required, 2.2×10^{-5} std-cm^3/second is established as the NAC-STC confinement limit.

All cask confinement seals are leak tested and verified to satisfy this established limit. The significance of this limit is that with this cask confinement limit, the radiological analyses in Chapters 7 and 11 demonstrate that the whole body dose rate as a result of normal and off-normal operation of the NAC-STC is much less than 1 mrem/year at the minimum ISFSI controlled area boundary.

12.1.2.4 Other Limits

Other structural design limits that apply to the NAC-STC are Regulatory Guide 7.6, Allowable Stresses; Regulatory Guide 7.8, Loading Conditions; and NUREG-0612, Lifting Requirements.