NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1 a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9235	12	71-9235	USA/9235/B(U)F-96	1	OF	15

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc., application dated February 19, 2009.

UNITED STATES NUCLEAR REGULATORY COMMISSION

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: NAC-STC

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has

5.(a)(2) Description (Continued)

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead.

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material.

The cask is closed by two steel lids which are bolted to the upper end forging. The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominantly balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Connecticut Yankee-multi-purpose canister (MPC) configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC).

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks

5.(a)(2) Description (Continued)

and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

The Yankee Class MPC and Connecticut Yankee MPC TSC assemblies include a vessel shell, bottom plate, and welded shield and structural lids that are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC. The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter. The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister. The structural lid is a 3-inch thick steel plate. The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

The Yankee Class MPC TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick, 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The Connecticut Yankee MPC fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The LaCrosse boiling water reactor multi-purpose canister MPC-LACBWR TSC assembly consists of a vessel shell, a bottom plate and a welded closure lid/closure ring assembly that are fabricated from stainless steel. The MPC-LACBWR TSC bottom stainless steel thickness is 1.25 inches. The shell is ½-inch thick rolled steel plate and 70.6 inches in diameter. The closure lid is a 7.0-inch thick steel plate/forging. The closure lid redundant welded closure is provided by a closure ring. The closure lid is provided with vent and drain penetrations to access the TSC cavity and they are closed by redundant welded port cover

5.(a)(2) Description (Continued)

plates. The MPC-LACBWR TSC fuel basket is designed to hold up to 68 irradiated LACBWR fuel assemblies, including up to 32 damaged fuel assemblies contained in DFCs and up to 36 intact fuel assemblies.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris, dross, or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14-inches high and 70-inches in diameter and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-8, Rev. 15
423-802, sheets 1-7, Rev. 21
423-803, sheets 1-2, Rev. 9
423-804, sheets 1-3, Rev. 9
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-810, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

5.(a)(3) Drawings (Continued)

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 4
455-820, sheets 1-2, Rev. 3
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

[1]Drawing defines the alternate configuration that accommodates the Yankee MPC damaged fuel can.

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos.:

YR-00-060, Rev. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev. D4
YR-00-062, sheet 2, Rev. D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos.:

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 3
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets 1-2, Rev. 2
414-820, Rev.0
414-870, Rev.3
414-871, sheets 1-2, Rev.6
414-872, sheets 1-3, Rev.6
414-873, Rev.2
414-874, Rev.0
414-875, Rev.0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev.4

5.(a)(3) Drawings (Continued)

414-887, sheets1-4, Rev. 4
414-889, sheets 1-3, Rev. 7
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets 1-2, Rev. 3
414-894, Rev. 0
414-895, sheets 1-2, Rev. 4

(vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 1
414-902, sheets 1-3, Rev. 3
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

(viii) For the Dairyland Power Cooperative/LaCrosse BWR transport package and TSC configuration, the TSC, fuel basket, and DFCs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

630045-800, sheets 1-2, Rev. 0
630045-870, Rev. 2
630045-872, sheets 1-2, Rev. 1
630045-877, Rev. 1
630045-881, sheets 1-2, Rev. [illegible]
630045-894, Rev. 1
630045-901, Rev. 0
630045-820, Rev. 0
630045-871, sheets 1-4, Rev. 2
630045-873, Rev. 1
630045-878, Rev. 1
630045-893, Rev. 1
630045-895, sheets 1-3, Rev. 1
630045-902, sheets 1-2, Rev. 1

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.25	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.50	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264[1]
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:
(1) - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

	Fuel Assembly Burnup (BU)															
Uranium Enrichment (wt% U-235)	BU ≤ 30 GWD/MTU				30 < BU ≤ 35 GWD/MTU				35 < BU ≤ 40 GWD/MTU				40 < BU ≤ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7≤E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9≤E<2.1	7	7	5	7	9	8	7	8	12	13	9	11	--	--	--	--
2.1≤E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3≤E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5≤E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7≤E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9≤E<3.1	6	5	5	5	7	7	5	6	8	8	6	8	11	11	8	10
3.1≤E<3.3	5	5	5	5	7	6	6	6	8	8	6	[illegible]	10	10	8	9
3.3≤E<3.5	5	5	5	5	[illegible]	6	[illegible]	6	8	[illegible]	6	[illegible]	10	10	7	9
3.5≤E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7≤E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9≤E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1≤E≤4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5(1)	--	--	--	6(1)	--	--	--	7(1)	--	--	--	9(1)
4.3≤E≤4.5	--	--	--	5(1)	--	--	--	6(1)	--	--	--	7(1)	--	--	--	8(1)

Notes:
(1) - Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950
Maximum Burnup (MWD/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

Notes:

[1] Combustion Engineering (CE) fuel with a maximum burnup of 32,000 MWD/MTU, a minimum enrichment of 3.5 wt. percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.

[2] Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.

[3] Stated enrichments are nominal values (fabrication tolerances are not included).

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity, and does not exceed the mass limits of 10 CFR 71.15.

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397[illegible]	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	[illegible]	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤ 1,600	≤ 1,500	≤ 1,500	≤ 1,600	≤ 1,600
Maximum Burnup (MWD/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:
1. Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.
2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.
3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.
4. Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.
5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.
6. Enrichment of the fuel within each DFC or RFA is limited to that of the basket configuration in which it is loaded.

5.(b)(1) Contents - Type and Form of Material (Continued)

(v) Irradiated undamaged and damaged Dairyland Power Cooperative LACBWR fuel assemblies based on design nominal or operating history record values listed below. Fuel assemblies may contain zirconium alloy shroud compaction debris.

Parameter	Units	Allis Chalmers	Exxon
Number of Assemblies per Canister[1]	---	32	68
Maximum Assembly Weight[6]	lbs	400	400
Assembly Length	in	103	103
Fuel Rod Cladding	---	Stainless Steel	Stainless Steel
Maximum Initial Uranium Mass[2]	kgU	121.4	111.9
Maximum Initial Enrichment	Wt% ^{235}U	3.64/3.94[5]	3.71[3]
Minimum Initial Enrichment	Wt% ^{235}U	3.6	3.6
Maximum Burnup	MWd/MTU	22,000	21,000
Maximum Assembly Decay Heat	W	63	62
Minimum Cool Time	yr	28	23
Assembly Array Configuration	---	10X10	10X10
Number of Fuel Rods	---	100	96
Maximum Active Fuel Length	in	83	83
Rod Pitch	in	0.565	0.557
Rod Diameter	in	0.396	0.394
Pellet Diameter	in	0.350	0.343
Clad Thickness	in	0.020	0.0220
Number of Inert Rods[4]	---	0	4
Inert Rod OD	in	N/A	0.3940

1. Maximum 68 assemblies per canister. Allis Chalmers fuel is restricted to Damaged Fuel Cans (DFCs). Therefore, Allis Chalmers fuel is limited to 32 assemblies per canister.
2. DFCs have been evaluated for 2% additional fuel rod mass.
3. Represents planar average enrichment.
4. Inert rods comprised of stainless steel clad tube containing zirconium alloy slug. Inert rods not required for fuel assemblies located in DFC.
5. Two Allis Chalmers fuel types: Type 1 at an enrichment of 3.64 wt% ^{235}U and Type 2 at 3.94 wt% ^{235}U.
6. Not including weight of DFC. DFCs may contain optional inner container subject to maximum weight and fissile material limits in this table.

5.(b)(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i): 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii): Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs. with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod. Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 22,340 lbs. with a maximum decay heat of 2.9 kW.

(v) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

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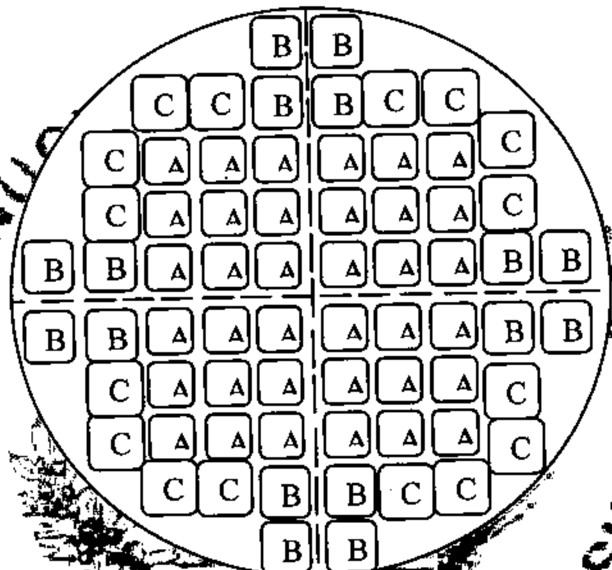
CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9235	12	71-9235	USA/9235/B(U)F-96	13 OF	15

5.(b)(2) Maximum quantity of material per package (Continued)

(vi) For the contents described in 5.(b)(1)(v): Up to 68 LACBWR assemblies, including up to 32 damaged fuel assemblies contained in DFCs, may be transported in the MPC-LACBWR TSCs.

Total weight of contents within the MPC-LACBWR TSC is 28,870 lbs., including the weight of 32 DFCs. The maximum decay heat is 4.5 kW per package. LACBWR undamaged fuel assemblies and LACBWR DFCs must be loaded in accordance with the following loading pattern:



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U, up to four slots maximum, B and C combined. Damaged Allis Chalmers fuel maximum enrichment 3.64 wt% ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt% ^{235}U, up to four slots maximum, B and C combined. Damaged Allis Chalmers fuel maximum enrichment 3.94 wt% ^{235}U.

LACBWR DFCs are allowed to contain an additional 2% fissile material to account for loose pellets, not necessarily associated with the as-built fuel assembly.

NOTE: The above sketch is not to scale. It is a depiction of the loading pattern.

5.(c) Criticality Safety Index (CSI):

(1) CSI=0.0 for contents described in 5.(b)(1)(i), 5.(b)(1)(ii), 5.(b)(1)(iii), and 5.(b)(1)(iv) (i.e., Yankee Class and CY Fuel and GTCC Waste).

(2) CSI=100 for contents described in 5.(b)(1)(v) (i.e., LACBWR fuel).

6. Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii).

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii), and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0 X 10[illegible] cm^3/sec (helium) and shown to have a leak rate no greater than 8.0 X 10[illegible] cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least [illegible].0 X 10[illegible] cm^3/sec (helium) and shown to have a leak rate no greater than 2.0 X 10^{-7} cm^3/sec (helium).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report.

(c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report.

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

12. Transport by air is not authorized.

13. Packagings must be marked with Package Identification Number USA/9235/B(U)F-96.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15. Revision No. 11 of this certificate may be used until October 31, 2011.

16. Expiration date: May 31, 2014.

REFERENCES

NAC International, Inc., application dated: February 19, 2009.

As supplemented June 3 and December 17, 2009; February 3, April 28, June 3, August 19, and September 1, 2010.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Chris Staab

Chris Staab, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 5, 2010

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9235
Model No. NAC-STC Package
Certificate of Compliance No. 9235
Revision No. 12

SUMMARY

By letter dated December 17, 2009, as supplemented February 3, April 28, June 3, August 19, and September 1, 2010, NAC International (NAC) submitted a revised application in accordance with 10 CFR Part 71 for an amendment to Certificate of Compliance (CoC) No. 9235 for the Model No. NAC-STC package to incorporate the Dairyland Power Cooperative's LaCrosse boiling water reactor (LACBWR) fuel as authorized contents.

Eight NAC International Drawings were revised and fourteen new drawings were added to update the CoC for this revision request.

Accordingly, CoC No. 9235 has been amended based on the statements and representations in the application, and staff agrees that the changes do not affect the ability of the package to meet the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 71. NRC staff evaluated the Model No. NAC-STC package and documented the security assessment review separately, as it contains sensitive information that cannot be made publicly available. The security assessment should be reviewed prior to approval of any amendment to this application.

EVALUATION

The submittal was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages, standards for fissile material packages, and performance standards under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). Staff reviewed the application using the guidance in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

Based on the statements and representations in the application, as supplemented, and the conditions listed in the CoC, the staff has reasonable assurance that the design has been adequately described and evaluated and meets the requirements of 10 CFR Part 71.

REFERENCES

NAC International, application dated December 17, 2009.

NAC International, supplements dated February 3, April 28, June 3, August 19, and September 1, 2010.

1.0 GENERAL INFORMATION

1.1 Package Description

The NAC International Storage Transport Cask (NAC-STC) is designed to store and transport spent nuclear fuel. The Transportable Storage Canister (TSC) is a component of the NAC International Multi-Purpose Canister (NAC-MPC) dry storage system. The NAC-MPC system is being amended to include the long-term storage of Dairyland Power Cooperative (DPC) LACBWR spent nuclear fuel.

1.2 Packaging Drawings

The applicant submitted eight revised drawings. The revised drawings include:

423-800, sheets 1-3, Rev. 15	Cask Assembly, NAC-STC Cask
423-802, sheets 1-7, Rev. 21	Cask Body, NAC-STC Cask
423-803, sheets 1-2, Rev. 9	Lid Assembly, Inner, NAC-STC Cask
423-804, sheets 1-3, Rev. 9	Details-Inner Lid, NAC-STC Cask
455-801, sheets 1-2, Rev. 4	42 MTR Element Top Module
455-820, sheets 1-2, Rev. 3	Spacer, Transport Cask, MPC-LACBWR
423-843, Rev. 2	Transport Assembly, Balsa Impact Limiters, NAC-STC
414-801, sheets 1-2, Rev. 2	Cask Assembly, NAC-STC, CY-MPC

The drawings were revised to incorporate changes due to the CoC revision to incorporate the LACBWR fuel and minor editorial corrections.

The applicant submitted fourteen new license drawings:

630045-800, Rev. 0 (sheets 1-2)	Assembly, Transport Cask, MPC-LACBWR
630045-820, Rev. 0	Spacers, Transport Cask, MPC-LACBWR
630045-870, Rev. 2	Shell Weldment, Canister (TSC), MPC-LACBWR
630045-871, Rev. 2 (sheets 1-4)	Details TSC, MPC-LACBWR
630045-872, Rev. 1 (sheets 1-2)	Assembly, Transportable Storage Canister (TSC), MPC-LACBWR
630045-873, Rev. 1	Assembly, Drain Tube TSC, MPC-LACBWR

630045-877, Rev. 1	Bottom Weldment, Fuel Basket, MPC-LACBWR
630045-878, Rev. 1	Top Weldment, Fuel Basket, MPC-LACBWR
630045-881, Rev. 1 (sheets 1-2)	Fuel Tube, MPC-LACBWR
630045-893, Rev. 1	Support Disk, Fuel Basket, MPC-LACBWR
630045-894, Rev. 1	Heat Transfer Disk, Fuel Basket, MPC-LACBWR
630045-895, Rev. 1 (sheets 1-3)	Fuel Basket Assembly, 68 Element BWR, MPC-LACBWR
630045-901, Rev. 0	Assembly, Damaged Fuel Can (DFC), MPC-LACBWR
630045-902, Rev. 1 (sheets 1-2)	Details, Damaged Fuel Can (DFC), MPC-LACBWR

The drawings were added to incorporate the LACBWR fuel as authorized contents.

1.3 Contents

The requested canistered fuel configuration is designed to store up to 68 LACBWR spent fuel assemblies, including up to 36 standard tube assemblies and up to 32 damaged fuel cans (DFCs) or fuel assemblies that exhibit slight physical effects (e.g., twist or bow). The DFCs may contain fuel assemblies defined as damaged or fuel debris. Four of the DFCs may contain undamaged Exxon fuel.

The LACBWR DFC is designed to hold a complete fuel assembly or fuel debris with or without a separate internal debris container. LACBWR damaged fuel and fuel debris may be placed in a separate drainable damaged fuel rod or debris container to facilitate handling of the fuel/debris and for placement in a LACBWR DFC. The DFC has a square cross-section that is slightly larger than a standard LACBWR fuel assembly. Consequently, loading of the DFC into the MPC-LACBWR canister basket is restricted to one of the 32 peripheral DFC tube assembly basket positions.

The calculated cavity contents weight of the NAC-STC LACBWR configuration of the NAC-STC is 55,835 pounds. The maximum gross transport weight of the NAC-STC spent fuel shipping package is calculated to be 243,235 pounds for LACBWR canistered fuel with the balsa impact limiters.

2.0 STRUCTURAL EVALUATION

The staff reviewed the application to revise the Model No. NAC-STC package structural design and evaluation to assess whether the package will remain within the allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 2 (Structural

Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

2.1 Structural Design – NAC-STC LACBWR

2.1.1 Discussion

The NAC MPC-LACBWR TSC is the primary enclosure for LACBWR fuel assemblies and DFCs. The TSC has the capability to hold up to 68 fuel assemblies, which may include up to 32 DFCs. The general construction is that of a right circular cylinder with a welded bottom plate and a welded closure lid. The TSC provides a leaktight containment barrier in addition to the primary containment provided by the NAC-STC.

2.1.2 Design Criteria

The design criteria for the NAC-STC LACBWR configuration are unchanged from the previously approved configuration presented in Table 2.1.2-1 of the Safety Analysis Report (SAR).

2.2 Weights and Centers of Gravity

The SAR summarizes the NAC-STC LACBWR configuration with varying weights and center of gravity locations as illustrated in Table 2.11.2-1.

2.3 Mechanical Properties of Materials

2.3.1 Materials Description

This revision to CoC No. 9235 for the NAC-STC transport system allows Lacrosse BWR spent fuel as an approved content. There is no change in the transport packaging design and materials, or impact limiter materials, from that already approved for other fuels. The Lacrosse fuel will be in a canister that is placed in the transport cask. Materials used in the canister design are the same as in previously used canisters except: 1) aluminum top and bottom spacers are used to limit the motion of the shorter fuel assemblies, and 2) there are more basket slots. The gamma and neutron shielding materials for the LACBWR MPC are in the transport packaging , the same as in the previously approved directly loaded configuration. Because the packaging is the same as previously used, and the materials of construction, codes used, and welds used for the cask are the same as previously approved, only the canisters, its internals, and the fuel were reviewed.

2.3.2 Materials Evaluation

Drawings for the Lacrosse BWR canister and interior components were provided by the applicant. Materials of construction, weld specifications, and weld and examination codes are on the drawings (Drawing Nos. 630045-870, 630045-872, 630045-871, 630045-902, 630045-895, and 630045-881). All weld types, inspection tests, and ASME acceptance tests are given in SAR Table 4.1-2 and are the same as has been previously approved. Weld procedures, examinations, and acceptance testing are also detailed in operating procedures (SAR Section 8.1.1). Details of the weld testing for the TSC are given in SAR Section 8.1.9 and are similar to those approved for the Yankee-MPC and CY-MPC canisters.

The canister is made of ASME SA240 stainless steel (304/304L). An aluminum spacer (type 6061) is bolted to the underside of the closure lid to provide movement of the fuel assemblies (SAR Section 1.4.1). Stainless steel (type 304) fuel tubes are in stainless steel (type 304) support disks, with aluminum (6061-T651) heat transfer disks intertwined. The seals are provided by metallic and Viton O-rings. These are the same materials with the same materials properties that have been previously approved for use for canistered fuel (SAR Section 2.11.3). The staff audited the stainless steel type 304 thermal expansion, Young's modulus, yield, and ultimate strength, along with the aluminum thermal expansion in various sections of the SAR and found them to be correct. The staff agrees with the SAR Section 2.11.1.3 that "because Type 304, and Type 304L, stainless steel are austenitic stainless steel, they do not undergo a ductile-to-brittle transition in the temperature range of interest for a spent fuel transport package and/or storage cask. Therefore, brittle fracture is not a concern." The staff verified the safe operating ranges for the lead gamma shield, solid neutron shield, aluminum heat transfer disks and both metallic and Viton O-rings stated in the SAR Section 3.6.3.2 against alternative sources and found them to be acceptable. The neutron absorber material is Boral clad with 304 stainless steel. The Boral has a ^{10}B loading of 0.015 g/cm^2. A 75% efficiency factor is used. The staff finds that the acceptance plan for the absorber material (SAR Section 8.1.7) is acceptable.

Lacrosse BWR fuel manufactured by Exxon and Allis Chalmers that consists of UO_2 pellets in a stainless steel cladding is being added as a new content. The fuel properties in SAR Tables 6.8.4-15, 6.8.3-1, 6.8.1-1, and 5.6.2-1 were reviewed by the staff and found to be correct and consistent. A canister will hold 68 LACBWR spent fuel assemblies, with up to 32 assemblies in damaged fuel cans (SAR Section 1.4.1). A DFC will hold complete assemblies or debris defined as fuel in the form of particle, loose pellets, and fragmented rods or assemblies (SAR Section 1.4.1). The staff found the definition of damaged fuel to be acceptable.

The staff finds that the drying procedure is adequate so that no water will be present to allow galvanic action between the stainless steel rod cladding and both canister components and the aluminum used in the heat transfer disks or spacers. The staff agrees with the SAR Section 2.11.4.1 statement "Significant neutron radiation damage does not occur for neutron fluences below 10^{19} n/cm^2. That value is much greater than the neutron fluence exposure that is experienced by the TSC components. Significant gamma radiation damage to metals only occurs for doses of $l0^{18}$ rads or more. This value is much higher than the gamma dose produced by spent nuclear fuel in the TSC." The staff expects no chemical interaction between any of the fuel, absorber, aluminum, or steel components of the canister system.

Many of the MPCs may be constructed, loaded, and be on a storage pad for a considerable number of years. The materials properties used for the evaluation of the safety systems and contents of the MPCs that have already been in storage service must be representative of the conditions at the time of transport, not at the time of the loading of the MPC. All mechanical and thermal properties of the materials of construction of the MPC used in this Part 71 analysis are for pristine materials.

The staff finds that the conclusion in the RAI response that stainless steel licensed for 40 years in-reactor use will survive the rigors of normal or off-normal storage conditions, and as such, these MPC materials will have essentially the same properties after a maximum 40 year storage period as acceptable. No conclusion is made for periods beyond 40 years. The staff concluded that the material properties may or may not remain the same after an accident and thus does not support the analysis for accident conditions.

The staff agrees that since the fuel burnup will be <45 GWd/MTU and be stored in an inert dry configuration that the condition of the fuel is not expected to change. The staff based their conclusion on the conditions of ISG-11, Rev. 3, being met, and the results of the examination of a cask containing low burnup fuel in Idaho after approximately 15 years storage.

Justification that the neutron absorbers (poisons) in the fuel basket will still function as required after the storage duration, accounting for, at a minimum, boron depletion, thermal and neutron embrittlement, and potential blistering was provided in the 2nd RAI responses, as was a justification that the aluminum casing will not become brittle. The staff agrees that the justifications, based on the dose received by the absorber over a 50 year period, will not decrease the effective B-10 content or embrittle the aluminum. Arguments were also presented in the RAI response that, based on Generic Safety Issues, Issue 196: Boral Degradation (NUREG-0933, Main Report with Supplements 1-32), blistering was not an issue. Staff reviewed the generic safety issue and supporting documentation and agreed with the conclusion that blistering, if it occurs during storage, should not affect the efficiency of the neutron absorber, if the canister is flooded.

To ensure that the properties of the materials and condition of the components has not significantly deteriorated, SAR Section 7.1.3.2 addresses the potential damage that may occur to the MPC during its removal from storage by requiring monitoring of the loading process and the need to evaluate any event experienced during transfer of the canister from the storage cask to the transport package. In addition, "Canisters that are retrieved from storage will be evaluated to ensure that site-specific ambient conditions and potential canister-specific exposure to 10 CFR Part 72 normal, off-normal and accident conditions meet the design and licensing requirements for a canister to be transported in the NAC-STC package...." This will ensure that the canister meets the design basis functions as authorized content of the NAC-STC packaging. Staff acceptance of this method is limited to low burnup fuel, in canisters that have been in storage for less than 40 years, and have the same materials of construction and neutron absorbers as the MPC in this amendment.

2.3.3 Conclusion

The staff finds no material issues with the requested amendment. All materials are expected to function in a manner that will meet the regulations for maintaining sub-criticality, heat dispersion, containment, and shielding.

- The staff agrees with the applicant that no galvanic, chemical, corrosive or radiological interactions take place and that the system meets the requirements of 10 CFR 71.43(d)(3).
- The staff expects that the welds, and seals will behave as expected over both the temperature and stress ranges expected for normal and hypothetical accident conditions (HAC) and not compromise the containment of radionuclides as required in 10 CFR 71.43(f), 10 CFR 72.51(a)(1), and 10 CFR 71.51(a)(2).
- A complete chemical and physical description of the contents of the DFCs was given, including a definition of damaged fuel meeting 10 CFR 71.33(b)(3), and 10 CFR 71.55(e)(1).
- The staff found that the fuel and properties of the cladding were sufficiently accurate to analyze the behavior of the contents during NCT and HAC and meet the requirements of 10 CFR 55(d)(2). Based on these properties the most credible configuration consistent with the fuel form could be determined meeting 10 CFR 71.55(b)(1).
- The staff has checked the mechanical and thermal properties of the materials of construction and found them to be accurate and suitable for analysis of the behavior of the system over the expected ranges of temperature and stress to maintain containment, and shielding, thus meeting the requirements of 10 CFR 71.33(a)(5), 10 CFR71.43(f), 10 CFR 71.47(a), 10 CFR 71.55(d)(4), 10 CFR 71.55(e).
- The staff found that the absorbers were described in sufficient detail, with reasonable quality assurance and acceptance plans to meet the requirements of 10 CFR 71.33(a)(5)(iii), 10 CFR 71.111, and 10 CFR 71.123.
- Applicable codes or defensible code alternates are listed meeting 10 CFR 71.3(c). The drawings are complete containing lists of the materials of construction, weld specifications and acceptance codes meeting requirement 10 CFR 71.107(a)
- All the materials stayed below their temperature limits for both NCT and HAC thus meeting the requirements of 10 CFR 71.33(a)(5), 10 CFR71.43(f), 10 CFR 71.47(a), 10 CFR 71.55(d)(4), and 10 CFR 71.55(e).

2.4 General Standards for All Packages (10 CFR 71.43)

The general standards for all packages for the NAC-STC LACBWR remain unchanged from the previously approved configurations and include Minimum Package Size, a Tamper-Proof Feature, Positive Closure, and Chemical and Galvanic Reactions.

2.4.1 Effects of Radiation on Materials

See Section 2.3.2 of this SER.

2.5 Lifting and Tie-Down Standards for All Packages (10 CFR 71.45)

The lifting and tie-down devices for the NAC-STC LACBWR remain unchanged from the previously approved configurations and the evaluations performed bound the current configuration.

2.6 Normal Conditions of Transport (NCT) (10 CFR 71.71)

The applicant notes that the LACBWR content total weight is bounded by the Yankee MPC content weight used to demonstrate regulatory compliance. Therefore, the Yankee MPC analyses bound those of the LACBWR configuration of the NAC-STC for the one-foot NCT and 30-foot HAC drop evaluations.

2.6.1 Heat

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Heat Condition analysis for the directly loaded and Yankee MPC configurations NAC-STC cask, respectively.

The requirements of 10 CFR 71.71(c) (1) are satisfied.

2.6.2 Cold

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Cold Condition analysis for the directly loaded and Yankee MPC configurations NAC-STC cask, respectively.

The requirements of 10 CFR 71.71(c)(2) are satisfied.

2.6.3 Reduced External Pressure

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Reduced External Pressure evaluation for the previously approved versions of the NAC-STC cask.

The requirements of 10 CFR 71.71(c)(3) are satisfied.

2.6.4 Increased External Pressure

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Increased External Pressure evaluation for the previously approved versions of the NAC-STC cask.

The requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5 Vibration

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the vibration analysis for the previously approved versions of the NAC-STC cask.

The requirements of 10 CFR 71.71(c)(5) are satisfied for vibration.

2.6.6 Water Spray

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Water Spray evaluation for the previously approved versions of the NAC-STC cask.

The requirements of 10 CFR 71.71(c)(6) are satisfied.

2.6.7 Free Drop

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Free Drop analysis for the directly loaded and Yankee MPC configurations NAC-STC cask, respectively.

The requirements of CFR 71.71(c)(7) are satisfied.

2.6.8 Corner Drop

The corner drop test does not apply since the gross weight of the package exceeds 110 lb (50 kg), in accordance with 10 CFR71.71(c)(8).

2.6.9 Compression

The compression drop test does not apply since the gross weight of the package exceeds 11000 lb (5000 kg), in accordance with 10 CFR 71.71(c)(9).

2.6.10 Penetration

The NAC-STC and the NAC-STC LACBWR configuration are bounded by the Penetration analysis for the previously approved versions of the NAC-STC cask.

The requirements of 10 CFR 71.71(c)(10) is satisfied.

2.6.11 MPC – LACBWR Fuel Basket Analysis (NCT)

2.6.11.1 Detailed Analysis

The applicant evaluated a fully detailed fuel basket analytically, which considered 5 orientations of the basket relative to the horizontal plane including 0, 11.2, 15.2, 37, and 45 degrees. In addition, the applicant evaluated the basket structure for an end drop condition.

2.6.11.2 Finite Element Model Description

The applicant utilized two separate finite element models for the end and side drops, respectively. The end drop evaluation was of a single maximally loaded support disk comprised of two dimensional shell elements. The side drop model was a detail three-dimensional half symmetry model that included the basket support disks and the canister shell. The model also incorporated gap elements to simulate contact between distinct parts.

2.6.11.3 Thermal Expansion Evaluation of MPC-LACBWR Support Disk

A thermal evaluation was performed to evaluate the possibility of loads due to differential thermal expansion as well as to provide an initial condition state for two of the load cases evaluated in conjunction with the one-foot drop. The results demonstrated that no differential thermal expansion stresses occur and the thermal stresses incorporated into the load combinations cited previously also do not adversely affect the performance under NCT.

2.6.11.4 Stress Evaluation of Support Disk – One-Foot End Drop

A static finite element analysis was performed with a 20 g load amplification factor to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 5.49.

2.6.11.5 Stress Evaluation of Support Disk – Thermal plus One-Foot End Drop

A static finite element analysis was performed with a 20 g load amplification factor plus thermal loads to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 5.31.

2.6.11.6 Stress Evaluation of Tie Rods and Spacers – One-Foot End Drop

A closed-form classical hand calculation was performed to demonstrate that the bottom spacers will be able to withstand the inertial loading from spacer disks, heat transfer disks, spacers, washers, and top or bottom weldment(s) due to a one-foot drop event. The analysis demonstrated a Margin of Safety of 1.82.

2.6.11.7 Stress Evaluation of Support Disk – One-Foot Side Drop

A static finite element analysis was performed with a 20 g load amplification factor to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 0.06.

2.6.11.8 Stress Evaluation of Support Disk – Thermal plus One-Foot Side Drop

A static finite element analysis was performed with a 20 g load amplification factor plus thermal loads to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 1.70.

2.6.11.9 Support Disk Shear Stresses –One-Foot Drops

The applicant evaluated the maximum shear stresses developed during the primary loading, side drop, and end drop load conditions. These stresses were compared against bounding material allowables at a temperature of 400°F and showed a positive Margin of Safety.

2.6.11.10 Stress Evaluation of Weldment – One-Foot End Drop

The applicant utilized finite element analysis results to demonstrate that the top and bottom weldments of the fuel basket assembly under a 20 g dynamic load due to an end drop would not exceed the allowable stresses of the weldment material. The results showed a minimum Margin of Safety of 0.85.

2.6.11.11 Bearing Contact with Canister Shell

The applicant utilized finite element analysis results to demonstrate that the canister, under a 20 g dynamic load due to a side drop, would not exceed the bearing stress of the canister inner shell.

2.6.11.12 Buckling Evaluation – Support Disk (One-Foot Drops)

The applicant utilized finite element analysis results to demonstrate that the canister under a 20 g dynamic load would not be subject to buckling of the outer canister shell. The results illustrate that a positive Margin of Safety was achieved.

2.6.12 MPC-LACBWR Transportable Storage Canister Analysis - NCT

2.6.12.1 MPC-LACBWR Canister – Canister Analysis Description

The applicant analyzed the canister, which is a right circular cylinder comprise of ½ inch stainless steel plate with a 1.25 inch bottom plate and a 7 inch welded closure lid. The finite element model used a loading of 20 g for both the end drop and side drop, which corresponds to a 1-foot drop.

2.6.12.2 Finite Element Model Description - MPC-LACBWR Canister

The finite element model of the canister was a three-dimensional half symmetry model comprised of solid elements and was analyzed with the Ansys finite element software. Where necessary, supporting structures, such as basket support disks, were included to accurately simulate structural performance. The model also used contact gap elements to simulate interaction between discreet parts of the package.

2.6.12.3 Thermal Analysis Evaluation of MPC-LACBWR Canister

A thermal evaluation was performed to evaluate the possibility of loads due to differential thermal expansion as well as to provide an initial condition state for two of the load cases evaluated in conjunction with the one-foot drop. The results demonstrated that the differential thermal expansion stresses and the thermal stresses incorporated into the load combinations cited previously do not adversely affect the performance under NCT.

2.6.12.4 Stress Evaluation for MPC-LACBWR Canister for One-Foot End Drop

A static finite element analysis was performed with a 20 g load amplification factor applied plus internal pressure to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the canister and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 1.71.

2.6.12.5 Stress Evaluation for MPC-LACBWR Canister for Thermal plus One-Foot End Drop

A static finite element analysis was performed with a 20 g load amplification factor applied, internal pressure, and thermal loads to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the canister and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 4.29.

2.6.12.6 Stress Evaluation for MPC-LACBWR Canister for One-Foot Side Drop

A static finite element analysis was performed with a 20 g load amplification factor applied plus internal pressure to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the canister and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 0.27.

2.6.12.7 Stress Evaluation for MPC-LACBWR Canister for Thermal plus One-Foot Side Drop

A static finite element analysis was performed with a 20 g load amplification factor applied, internal pressure, and thermal stresses to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the canister and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 1.66.

2.6.12.8 MPC-LACBWR Canister shear stresses for One-Foot Side Drop and One-Foot End Drop

The applicant evaluated the maximum shear stresses developed during the pressure, side drop, and end drop load conditions. These stresses were compared against bounding material allowables at a temperature of 300 F and showed a positive Margin of Safety.

2.6.12.9 MPC-LACBWR Canister bearing stresses for One-Foot Side Drop

The applicant utilized finite element analysis results to demonstrate that the canister under a 20 g dynamic load due to a side drop would not exceed the bearing stress of the cask inner shell.

2.6.12.10 Canister Buckling Evaluation – One-Foot End Drop

The applicant performed a closed-form classical hand calculation to demonstrate that the canister under a 20 g dynamic load would not be subject to buckling of the outer canister shell.

2.6.12.11 Canister Lifting Evaluation

The canister was evaluated for a static lifting condition and included self-weight, internal pressure of 20 psig, and a dynamic load factor of 1.1. The applicant reported stress intensity results from a 3D finite element analysis and demonstrated that all Margins of Safety at critical sections exceeded 0.0 with a minimum Margin of Safety of 0.43.

2.6.12.12 MPC-LACBWR Canister Closure Weld Evaluation – NCT

The canister closure weld was evaluated with a weld stress reduction factor applied to the allowable stress for the weld. Stress intensities derived from the finite element analysis were compared with the reduced allowable stresses. The results illustrated that in all cases, the stress intensity combinations had a Margin of Safety exceeding zero.

2.6.13 MPC-LACBWR Damaged Fuel Can Analysis – NCT

The damaged fuel can was evaluated for lifting and handling loads as well as side and end drops subjected to a 20 g dynamic amplification. The applicant performed classical closed-form calculations to evaluate tear out, bending, direct tension, and weld performance during lifting and handling operations. Similarly, the applicant evaluated the effects of a one-foot drop on the compressive strength of the DFC body, lid strength, unsupported sections during the side drop, and welds.

2.6.14 Cavity Spacer Evaluation for LACBWR canistered fuel - NCT

The MPC-LACBWR Canister is 6 inches shorter than the Yankee MPC canister and required an additional aluminum honeycomb spacer to account for the gap. Since the additional space is of the same material and construction as the previously approved spacers, previous analysis of compressive stress and the conclusions thereof, still apply.

2.7 Hypothetical Accident Conditions (HAC) (10 CFR 71.73)

The applicant notes that the LACBWR content total weight of 55,837 lbs is bounded by the Yankee MPC content weight used to demonstrate regulatory compliance. Therefore, the Yankee MPC analyses bound those of the LACBWR configuration of the NAC-STC for the one-foot NCT and 30-foot HAC drop evaluations.

2.7.1 Thirty-Foot Free Drop

2.7.1.1 Thirty-Foot End Drop - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

2.7.1.2 Thirty-Foot Side Drop - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

2.7.1.3 Thirty-Foot Corner Drop - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation for the 30-foot side drop.

2.7.1.4 Thirty-Foot Oblique Drop - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation for the 30-foot side drop.

2.7.1.5 Lead Slump - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

2.7.1.6 HAC Closure Analysis - The NAC STC cask evaluation in the STC-LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

The requirements of 10 CFR 71.73(c)(1) are satisfied.

2.7.2 Puncture

The NAC STC cask evaluation in the NAC-STC LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

The requirements of 10 CFR 71.73(c)(3) are met.

2.7.3 Thermal

The NAC STC cask evaluation in the NAC-STC LACBWR configuration is bounded by the NAC STC cask evaluation in the directly loaded and Yankee MPC configurations.

The requirements of 10 CFR 71.73(c)(4) are met.

2.7.4 Crush

This evaluation is not applicable due to the package mass exceeding 500 kg (1100 lbs) per 10 CFR 71.73(c)(2).

2.7.5 Immersion - Fissile

This requirement is satisfied by the demonstration of containment after the HAC sequence of free drop, puncture, and fire preceding the immersion evaluation. In addition, the criticality evaluation considered water in leakage.

The requirements and intent of 10 CFR 71.73(c)(5) are satisfied.

2.7.6 Immersion - All Packages/Deep Water Immersion

The immersion analysis presented in Section 2.7.7 is applicable for all NAC-STC configurations presented previously and in this amendment

The requirements of 10 CFR 71.73(c)(6) are met.

2.7.8 MPC – LACBWR Fuel Basket Analysis (HAC)

2.7.8.1 Detailed Analysis

See Section 2.6.11.1.

2.7.8.2 Finite Element Model Description

See Section 2.6.11.2.

2.7.8.3 Stress Evaluation of Support Disk – 30-Foot End Drop

A static finite element analysis was performed with a 60 g load amplification factor to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the worst case end drop of 2.53.

2.7.8.4 Stress Evaluation of Support Disk – 30-Foot Side Drop

A static finite element analysis was performed with a 55 g load amplification factor to simulate the dynamic one-foot drop event. Stress intensities were reported at critical sections in the support disk and the results demonstrate that all Margins of Safety exceed 0.0, with the minimum Margin of Safety for the Bottom End drop of 0.05.

2.7.8.5 Stress Evaluation of Support Disk – 30-Foot Off Angle Drop

As discussed previously, the end and side drop events bound this orientation.

2.7.8.6 Stress Evaluation of Tie Rods and Spacers – 30-Foot End Drop

A closed-form classical hand calculation was performed to demonstrate that the bottom spacers will be able to withstand the inertial loading from spacer disks, heat transfer disks, spacers, washers, and top or bottom weldment(s) due to a 30-foot drop event. The analysis demonstrated a Margin of Safety of 1.12.

2.7.8.7 Buckling Evaluation – Support Disk (30-Foot Drop)

The applicant performed a closed-form classical hand calculation using results from a finite element analysis to demonstrate that the canister under a 30-foot drop dynamic load would not be subject to buckling of the outer canister shell. The applicant demonstrated that the minimum Margin of Safety against buckling was +0.26.

2.7.8.8 MPC-LACBWR Fuel Tube Analysis

The applicant evaluated the fuel tube under accident conditions to confirm its ability to maintain positioning of Boral plates within the fuel basket structure. The tube itself provides no structural function with respect to fuel assembly support. The applicant used a combination of classical closed-form hand calculations and finite element analysis to demonstrate positive Margins of Safety for stresses as well as maintaining an operational requirement minimizing deflections.

2.7.8.9 Stress Evaluation of Weldment – 30-Foot End Drop

The applicant utilized finite element analysis results to demonstrate that the top and bottom weldments of the fuel basket assembly under a 60 g dynamic load due to an end drop would not exceed the allowable stresses of the weldment material. The results showed a minimum Margin of Safety of 0.63.

2.7.9 MPC-LACBWR Transportable Storage Canister Analysis - HAC

2.7.9.1 MPC-LACBWR Canister – Accident Analysis Description

The applicant analyzed the canister which is a right circular cylinder comprise of ½ inch stainless steel plate with a 1.25-inch bottom plate and a 7-inch welded closure lid. The finite element model use is the same as that used for NCT with the exception of modified loadings of 48 g and 55 g for the end drop and side drop, respectively, corresponding to a 30-foot drop. See Section 2.6.12.2 for a description of the finite element model.

2.7.9.2 MPC-LACBWR Canister – Accident Analysis Results

The applicant summarized the stress results and associated Margins of Safety for each orientation/configuration considered for HAC. The summary results showed that positive Margins of Safety were achieved at all critical sections for all evaluations with the minimum Margin of Safety of 0.37 for the side impact with internal pressure.

2.7.9.3 Canister Buckling Evaluation – 30-Foot End Drop

The applicant performed a closed-form classical hand calculation to demonstrate that the canister under a 48g dynamic load would not be subject to buckling of the outer canister shell.

2.7.9.4 MPC-LACBWR Canister Closure Weld Evaluation – Accident Conditions

The canister closure weld was evaluated with a weld stress reduction factor applied to the allowable stress for the weld. Stress intensities derived from the finite element analysis were compared with the reduced allowable stresses. The results illustrated that in all cases, the stress intensity combinations had a Margin of Safety exceeding zero.

2.7.10 MPC-LACBWR Damaged Fuel Can Analysis - HAC

The applicant performed closed-form classical hand calculations to determine the structural performance of the Damaged Fuel Can (DFC) during HAC conditions.

The DFC was evaluated for compressive stress, buckling, side drop loadings, weld performance, lifting conditions, and end drop impacts. All Margins of Safety exceeded 0.0.

2.8 Special Form – NAC-STC LACBWR

Not applicable.

2.9 Fuel Rods Buckling Assessment for LACBWR Canistered Fuel

Staff reviewed the classical closed-form hand calculations, finite element analysis methodologies, and tabulated stress results and subsequent Margins of Safety presented by the applicant. Staff also evaluated bounding conditions presented by the applicant and found them credible.

2.10 Conclusions

Evaluation Findings, Normal Conditions of Transport – The staff has reviewed the packaging structural performance under the normal conditions of transport and concludes that there will be no substantial reduction in the effectiveness of the package.

Evaluation Findings, Hypothetical Accident Conditions – The staff has reviewed the packaging structural performance under the hypothetical accident conditions and concludes that the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The staff reviewed the application to revise the Model No. NAC-STC package thermal design and evaluation to assess whether the package temperatures will remain within their allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 (Thermal Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

3.1 Thermal Design

The design basis heat load for the LACBWR spent fuel is 0.66 watts per assembly, and for the 68 assembly transportation package the total heat load is 4.5 kW. This heat load is significantly lower than the previously approved loadings for the STC, where the uncanistered limit was 22.1 kW (Yankee Class Fuel), and for canistered fuel it was 17.0 kW (Connecticut Yankee Fuel). For the LACBWR fuel, 32 of the assemblies are allotted for damaged fuel which are loaded into damaged fuel cans (DFCs) and located on the perimeter of the basket. The remaining 36 assemblies are allotted for intact fuel assemblies.

3.2 Thermal Evaluation

NAC performed a specific thermal analysis of the La Crosse fuel package and demonstrated that the fuel and materials of the package stay within their allowable temperature limits. Since the previously licensed configurations for the Yankee and

Connecticut Yankee fuels have a significantly higher thermal loading than the La Cross STC, and the configurations are similar, it is reasonable to conclude that the thermal loading of the La Crosse is also acceptable. A specific thermal analysis was performed for the NAC-STC LACBWR to assist in their structural evaluation of the canister. This thermal analysis continues to demonstrate that all STC materials are below their allowable temperature limit, including the metallic containment seals that are required to be utilized for the NAC-STC LACBWR package. The pressure calculations provided demonstrate that the maximum pressure calculated for the package for NCT (6.2 psig for canister) and HAC (38.7 psig (2.63 atm) for the canister) are below the package's maximum normal operating pressure (MNOP) of 50.6 psig. Note that the cask is hydrotested to 76 psig as stated in SAR Section 8.1.2.3.

Compaction of the damaged fuel was not specifically evaluated for this application. However, it was evaluated for the storage of LACBWR and was shown to have minimal impact on component temperatures, which would also be valid for the NAC-STC LACBWR package.

3.3 Conclusion

The staff finds that the NAC-STC LACBWR meets the requirements of 10 CFR Part 71.

4.0 CONTAINMENT

The staff reviewed the application to revise the Model No. NAC-STC package to verify that the package containment design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

4.1 Containment System Design

The containment boundary of the NAC-STC LACBWR package is defined under containment condition B of the NAC-STC transport system and its components are detailed in Table 4.1-1 of the application under containment condition B using metallic O-rings. These components are an inner shell, upper and lower shell rings transitional sections, bottom inner forging, top forging, inner lid, inner lid O-ring, vent port cover plate, vent port cover plate inner O-ring, drain port cover plate and drain port cover plate inner O-ring. In this system, the inner shell is 1.5 inches thick and has a 74.0 inch outer diameter, the inner lid is 79.0 inches in diameter and the port cover assemblies have 3.75 inches diameters.

4.2 Containment Evaluation

The staff has reviewed the evaluation of the containment system under NCT and concludes that the package satisfies the containment requirements of 10 CFR 71.43(f), and 10 CFR 71.51(a)(1) with no dependence on filters or a mechanical cooling system. The package has been designed and tested as "leak tight" per ANSI N14.5 (i.e., for a leak rate of 1x 10^{-7} ref cm^3/s). The maximum operating pressures for NCT are within design limits according to Section 3.6.4.4 of the Thermal Evaluation chapter.

For the containment evaluation under HAC, the package shall satisfy the requirements of 10 CFR 71.51(a)(2). According to Section 4.6.3, the containment boundary and the leaktight conditions are maintained after structural and thermal HAC tests. The maximum operating pressures for HAC are within design limits according to Section 3.6.5 of the Thermal Evaluation chapter.

4.3 Conclusion

The staff has reviewed the Containment Evaluation section of the SAR and concluded that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71, and that the package meets the containment criteria of ANSI 14.5 for NCT and HAC.

5.0 SHIELDING EVALUATION

NAC submitted an application for an amendment to the NAC-STC CoC to incorporate Dairyland Power Cooperative LACBWR spent fuel assemblies as approved contents for transport in the NAC-STC system. This amendment is the 12th revision to CoC No. 9235.

The NAC-STC system has previously been approved to store fuel assemblies from Yankee Nuclear Power Plant or fuel assemblies from the Connecticut Yankee Atomic Power Company. Both are decommissioned pressurized water reactors.

The staff reviewed the addition of the new contents using the guidance in Section 5 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2000. The staff's evaluation of the applicant's changes to the shielding evaluation follows.

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The staff reviewed the general information chapter in the NAC-STC SAR as well as any additional information on the shielding design in Chapter 5 of the NAC-STC SAR, "Shielding." The staff determined that all figures, drawings, and tables describing the shielding features are sufficiently detailed to support an in-depth evaluation. The shielding design features of the NAC-STC include multi-walled shielding materials that completely surround the fuel. This includes stainless steel and lead for gamma shielding and a borated polymer (NS-4-FR) for neutron shielding.

5.1.2 Codes and Standards

The NAC-STC SAR identifies the appropriate regulations in 10 CFR Part 71 throughout Section 5. The staff also verified that the NAC-STC SAR appropriately identifies the ANSI/ANS 6.1.1 1977 version for the flux to dose rate conversion factors in Table 5.6.4-2.

5.1.3 Summary Table of Maximum Radiation Levels

The staff examined the summary tables in Tables 5.6.1-1 and 5.6.1-2 of the NAC-STC SAR. The staff reviewed these tables to ensure that the NAC-STC meets the requirements in 10 CFR 71.47 and 10 CFR 71.51. Since the NAC-STC SAR states that the NAC-STC will be operated under "exclusive use," the staff verified that the evaluated radiation levels do not exceed those specified in 10 CFR 71.47(b).

The staff verified that the summary table states that the limit of 200 mrem/h will not be exceeded on the external surface of the package. This meets the regulatory limits in 10 CFR 71.47(b)(1).

Although the applicant did not show calculated dose rates at the outer vehicle surface in the summary table, the table does show that there will be less than 200 mrem/h dose rate at the surface of the package, therefore 10 CFR 71.47(b)(2) is also met. This regulation requires that the dose rate be limited to 200 mrem/h at the vehicle surface.

The staff verified that the summary table states that the limit of 10 mrem/h will not be exceeded at any point 2 meters from the outer lateral surface of the vehicle. The staff finds that this meets the requirement in 10 CFR 71.47(b)(3).

The staff verified that the summary table states that the external radiation dose during hypothetical accident conditions (HAC) does not exceed 1 rem/h at 1 meter from the external surface of the package. The staff finds that this meets the requirement of 10 CFR 71.51(a)(2).

5.1.4 Transport Index

The applicant calculated a maximum transport index of 2 for the NAC-STC with LACBWR fuel based on the shielding analysis results. The NAC-STC SAR states that the maximum dose rate at 1 meter from the surface of the package during NCT is 1.3 mrem/hr based on the results of the calculations presented in Tables 5.6.1-1 and 5.6.1-2 of the NAC-STC SAR. The staff finds this value to be reasonable. The staff finds that the maximum transport index has been appropriately determined and that it meets the requirements of 10 CFR 71.4. The actual transport index will be determined upon shipment.

5.2 Source Specification

In addition to previously approved contents, the applicant is adding Allis Chalmers and Exxon fuel to the NAC-STC. The fuel assemblies are 10x10 with stainless steel cladding. The minimum enrichment, maximum burnup, and cooling time that were used in the shielding analysis are as follows:

Fuel Type	Minimum Enrichment [wt % U-235]	Maximum Burnup [MWD/MTU]	Minimum Cooling Time
Allis Chalmers	3.6%	22,000	28
Exxon Nuclear Company	3.6%	21,000	23

Information about the fuel parameters used in the shielding analysis is located in Table 5.6.2-1 of the NAC-STC SAR. The staff verified that these parameters are consistent with those listed Table 1.4-4 of the general information section in the NAC-STC SAR.

The applicant modeled the central 36 fuel assemblies as Exxon Nuclear Company assemblies, and the outer 32 fuel assemblies as Allis Chalmers fuel. This information is specified in Figure 5.6.1-1 of the NAC-STC SAR.

The applicant determined the source term for the fuel and fuel assembly hardware using SAS2H as part of the SCALE 4.3 code package. The staff finds that the SCALE 4.3 code package is acceptable for use in this application.

The applicant is using a 27 group library composed primarily of ENDF/B-IV cross sections with some pre-released ENDF/B-V data for a large number of fission product isotopes. The staff finds that this cross section set is appropriate.

The applicant gives the reactor operating conditions assumed for the fuel when generating the source terms in Table 5.6.2-2 of the NAC-STC SAR. The applicant uses a single uniform moderator density based on the average core outlet density when generating the axial source profile. They use the core inlet density for the core bypass region (channel to channel region). Since during operation there is varied density along the height of the fuel, the applicant performed a comparison of the uniform versus a varied density axial profile from the assembly with the largest peaking and demonstrated that the uniform moderator density produces more conservative or a roughly equivalent source term in terms of neutrons or gammas per second. The applicant also compared the energy spectra of the source calculated using the uniform water density versus the node specific water density. The results of this comparison are shown in Tables 5.6.2-11, 5.6.2-12, and 5.6.2-13 of the NAC-STC SAR. These tables show that the energy comparison is roughly the same with the largest difference being at the lower energy groups where there is a lower contribution to dose. Therefore, the staff finds the operating conditions used to generate the source terms acceptable.

5.2.1 Gamma Source

The staff verified that the applicant specified the gamma source term as a function of energy for both the fuel and the hardware. These values are listed for a single assembly in Table 5.6.2-4 of the NAC-STC SAR, and for the fuel and hardware in Table 5.6.2-5 of the NAC-STC SAR.

The staff verified that the applicant appropriately considered the Co-60 contained in the fuel assembly hardware. The applicant states that they used a 2 g/kg Co-60 impurity within the stainless steel and that this is the maximum impurity allowed per manufacturer specifications. The staff finds this value to be reasonable and acceptable.

The staff reviewed the energy group spectra of the gamma source to determine if it is appropriate. The applicant used SAS2H to determine a source term and a grouped energy spectra for the source term. This was then used in the MCNP code for the shielding calculation which uses continuous energy cross sections. Although the applicant is mixing a grouped energy source with a continuous energy shielding calculation, the staff accepts that using continuous energy cross sections is always more accurate and finds this combination acceptable.

5.2.2 Neutron Source

The staff verified that the applicant specified the neutron source as a function of energy. This is listed in Table 5.6.2-3 of the NAC-STC SAR for a single assembly. The applicant used the MCNP code for the shielding calculation and the sub-critical multiplication was not accounted for within the code. The applicant used a scaled factor based on the system multiplication factor (as discussed in the Criticality evaluation, Chapter 6 of the NAC-STC SAR) to account for this effect. This method is discussed in Section 5.6.2.2 of the NAC-STC SAR and is based on a multiplication factor of 0.4 for a dry cask. This results in the neutron source being scaled up by 1.67 for a dry cask. This approach has been used for the NAC-STC for previously approved contents and for other approved applications (NAC MAGNASTOR Dry Cask Storage System, Docket No. 72-1031). The staff finds this acceptable for the MPC-LACBWR application.

5.3 Model Specification

The staff reviewed Section 2 (structural evaluation) and 3 (thermal evaluation) of the NAC-STC SAR to determine the effects of the NCT and HAC on the packaging and its contents. Section 5.6.3.3 of the NAC-STC SAR has a summary of the NAC-STC cask features assumed during NCT. The applicant includes the radial neutron shield and shield shell and the upper and lower impact limiters with reduced dimensions. The impact limiter is assumed to be all balsa.

Chapter 2 of the NAC-STC SAR shows that NCT tests required by 10 CFR 71.71 do not impact the geometry of the package. However the impact limiters experience some crush and deformation, which is why the applicant truncates the dimensions of the impact limiters. The effects of the NCT tests are bounded by the dimensions assumed in the model. The staff finds that the shielding model is consistent with the effects of the tests performed in compliance with 10 CFR 71.71.

During HAC the applicant does not include the upper and lower impact limiters as part of the shielding model. The applicant does include the radial neutron shield and shell, however it removes hydrogen, oxygen and nitrogen from the neutron shield material.

The applicant does account for the lead slump resulting from the 30-foot drop. They account for this in the top and bottom (axial) to account for a package dropped on its end and in the radial/angular direction for a package dropped on its side. The axial and the radial/angular slump are applied simultaneously. The amount of lead slump is determined based on the volume of the gap formed due to the lead pour and the cool down. The applicant determined that this corresponds to an axial slump of 5.95cm or an angular slump across 0.438 radians. This is consistent with the information presented in Section 2.7.1.5 of the NAC-STC SAR. The staff finds the amount of lead slump assumed by the applicant acceptable.

5.3.1 Configuration of Source and Shielding

The staff examined the sketches of the shielding model as represented by Figures 5.6.3-1 through 5.6.3-5 of the NAC-STC SAR. The staff verified that the dimensions were consistent with the cask drawings presented in Section 1.4.3.2 of the NAC-STC SAR. The applicant used nominal dimensions for modeling the cask. The staff finds this acceptable given the dose rates that are calculated with the LACBWR fuel. In all models the cask and canister shield thicknesses and axial extents are explicitly represented, including streaming paths.

The applicant modeled the Allis Chalmers fuel in the 32 peripheral basket locations and the Exxon Nuclear Company fuel in the 36 interior basket locations as described in Figure 5.6.1-1 of the NAC-STC SAR. The applicant performed an analysis in Section 5.6.4.6 of the NAC-STC SAR where there is a single Exxon Nuclear Company assembly placed in the peripheral locations and found there is a slight increase (about 1%) in the dose at the azimuthal location of the assembly. This value is within the uncertainty of the calculation. Therefore, the staff finds that having up to 4 damaged or undamaged Exxon Nuclear Company assemblies in the peripheral locations, as allowed by Figure 1.4-4 of the NAC-STC SAR, would give a minimal increase in gamma dose rates and is acceptable.

The fuel and hardware are homogenized and placed into regions defined by the fuel assembly width and height which is subdivided axially into source regions for the active fuel, upper plenum, and upper and lower end fittings. This is shown in Figure 5.6.2-1 of the NAC-STC SAR.

The applicant has assumed that there is no absorber material (BORAL) within the basket. The staff finds this assumption is conservative.

To account for the effects of magnitude and spectrum flux variations on the hardware activation, the applicant used regional flux factors to adjust the assembly hardware mass. This was done because the source strength of the hardware was calculated per mass. The adjusted mass was multiplied by the source term results for the light elements from the SAS2H calculation to get the appropriate source terms for the hardware regions. This is shown in Table 5.6.2-6 of the NAC-STC SAR. The staff reviewed this information and finds that this is acceptable.

The applicant created an axial profile based on measured burnup profile data. The applicant's burnup profile is shown in Figure 5.6.2-2 of the NAC-STC SAR. The staff viewed the axial burnup profiles as well as the axial gamma and neutron source profiles. The staff finds that they are acceptable and would provide representative results for the dose rates where there is axial peaking.

The staff verified that the dimensions of the transport vehicle are included. The applicant assumes a railcar width of 124 inches to determine the 2-meter surface. This is conservative compared to the diameter of the balsa impact limiters which is 128 in. The staff finds this acceptable.

The staff verified that the dose point locations include all locations prescribed by 10 CFR 71.46(b) and 71.51(a)(2). This includes the package surface, 1 meter from the package surface, and 2 meters from the railcar (transportation vehicle) edge. The applicant provided figures showing the dose rate profiles for various locations axially, radially, and azimuthally around the cask in Figures 5.6.4-1 through 5.6.4-30 of the NAC-STC SAR. The staff confirmed visually that the dose rates reported in Tables 5.6.1-1 through 5.6.1-4 of the NAC-STC SAR generally agreed with the locations where the dose rate is highest. This occurs along the cask radial and axial centerline.

The staff verified that the applicant has considered potential streaming effects of the trunnions and neutron shield heat fins. Streaming due to trunnions was found to be bounded by the centerline dose rates. Although the neutron dose showed peaking where the neutron shield heat fins were located, this was compensated for by a lower gamma dose therefore the heat fins are not accounted for in the reported dose rates. This is discussed in Section 5.6.3.4 of the NAC-STC SAR. The staff finds this acceptable.

5.3.2 Material Properties

The NAC-STC is made of steel, lead and polymer (NS-4-FR). The staff verified that the applicant identified the materials and mass densities of the homogenized fuel assembly, shield and structural materials and impact limiter. These are specified in Tables 5.6.3-5 and 5.6.3-6 of the NAC-STC SAR. The staff finds that the values used are typical values for the commonly used materials and are reasonable for use in the shielding analysis. For the NS-4-FR the applicant performs neutron shielding material testing as specified in Section 8.1.5.3 of the NAC-STC SAR to ensure that the NS-4-FR density and uniformity are in accordance with that assumed in the shielding model. The staff finds this provides reasonable assurance that the material properties specified for NS-4-FR in the NAC-STC shielding model are representative of actual package conditions.

Section 8.1.5.3 of the NAC-STC SAR states that "Each lot (mixed batch) of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density, meet the requirements specified in Chapters 1 and 3 and the License Drawings." Chapter 3 of the NAC-STC SAR (Figure 3.3-2) states the properties of NS-4-FR and the staff has confirmed that the properties used in the shielding model conforms to those listed in Chapter 3 of the NAC-STC SAR. Furthermore, the density used for the shielding analysis is 1.63g/cm^3. This is conservative in relation to the actual density of the NS-4-FR listed in Figure 3.3-2 of the NAC-STC SAR as 1.68g/cm^3. The staff finds this acceptable.

The applicant assumes that the oxygen, hydrogen and the nitrogen are removed from the NS-4-FR neutron shielding material during accident conditions. Chapter 3 of the NAC-STC SAR states that the neutron shield will exceed its safe operating range during fire accident conditions. Section 3.5.1.1.3 of the NAC-STC SAR states that "At the end of the fire transient, the neutron shield is considered to be voided of NS-4-FR." The shielding analysis still includes the boron, carbon, and aluminum left in the neutron shield. In response to staff questions, the applicant submitted information demonstrating that the neutron shield material would survive during a 10 CFR 71.73 design basis fire. Voiding the shield during the fire transient was done because it was considered a conservative assumption with respect to the thermal analysis. Therefore the staff finds that it is acceptable to assume some presence of the neutron shield for the HAC shielding analysis. In addition, the staff recognizes that without the hydrogen to moderate neutrons leaving the shield that the boron would be inefficient at absorbing them. Therefore the staff finds that the inclusion of boron within the neutron shield would not cause the calculation to under predict dose rates to where the package would exceed regulatory limits.

Section 3.3.2 of the NAC-STC SAR states that the maximum expected weight loss of the NS-4-FR neutron shield material will be less than 2% after a 20 year period if the NAC-STC package is maintained within normal operating temperatures. The staff finds, based on the calculated neutron doses in Tables 5.6.1-1 through 5.6.1-4 of the NAC-STC SAR that this is acceptable for the proposed additional contents of the LACBWR fuel and would not cause the package to exceed any regulatory dose rate limits.

5.4 Evaluation

5.4.1 Methods

For the shielding analysis the applicant uses the MCNP5 code with default neutron and photon cross sections that are from various releases of the ENDF/B-V and ENDF/B-VI libraries and the MCNPLIB04 photoatomic data set. MCNP is a three dimensional code that employs the Monte Carlo method. It is widely used and recognized for shielding analyses. The staff has previously accepted the use of MCNP for similar shielding evaluations and finds that its use is acceptable for this application.

5.4.2 Key Input and Output Data

The staff verified that key input data for the shielding calculations are identified and that information about the source and shielding were properly input into the codes. The staff viewed the output files provided to the staff and determined that they have proper convergence and that the calculated radiation levels from the output files agree with those reported in the text.

5.4.3 Flux-to-Dose-Rate Conversion

The NAC-STC SAR states that the shielding evaluation uses the ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors in all the cask shielding evaluations. The staff finds this acceptable.

5.4.4 Radiation Levels

The staff viewed the calculated radiation levels as displayed in Figures 5.6.4-1 through 5.6.4-28 of the NAC-STC SAR. The staff confirmed that the calculated radiation levels under both NCT and HAC for undamaged and damaged fuel are in agreement with the summary tables and that they satisfy the limits in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2). The staff verified that the analysis showed that the locations selected are those of maximum radiation levels and include any radiation streaming paths.

The staff also verified that the applicant's evaluation addresses damage to the shielding under NCT and HAC.

5.4.5 Confirmatory Analysis

During the review of Revision 6 of the NAC-MPC Storage System (Docket No. 72-1025), the staff performed an independent analysis of the source term generated from the LACBWR fuel. Staff conducted confirmatory source-term analyses using SAS2H with ENDF-V cross-section libraries in the SCALE 5.1 package. Inlet and outlet moderator density were analyzed to capture the bounding flux spectra expected within the assembly during operation. Power, cycle time and down time were calculated as presented by the applicant. ORIGEN-S was used to determine the combined source term at the minimum cooling time for each LACBWR assembly type. Staff also re-ran the bounding Combustion Engineering 16x16 Type A Connecticut Yankee assembly source-term calculation for comparison.

It was not possible to conduct a direct confirmation. The applicant used a version of SAS2H with a cross-section library unavailable to the staff. The changes required to the input and to conform to the newer versions would significantly change the problem being investigated. The staff's own conclusions were compared separately, and an identical comparison was made using the provided output files. The trend among each group of analyses with a single variable change is the same for both the Allis Chalmers and Exxon fuel assemblies.

Comparing the staff's source term calculation for the CE Type A fuel with the staff's analysis of the LACBWR fuel indicate that the applicant's analysis has resulted in a reasonable estimate of the LACBWR source term.

The staff did not perform an independent shielding calculation to confirm the dose rates calculated by the applicant. The NAC-STC shield design has been previously reviewed and approved by the NRC staff for Yankee Class and Connecticut Yankee fuel. The staff finds that the source terms for these two fuel types are greater than that of the LACBWR fuel and therefore the shielding characteristics of the NAC-STC are bounding for the LACBWR fuel.

5.5 Evaluation Findings

The staff's evaluation of the NAC-STC results in the following evaluation findings for the addition of LACBWR fuel:

- As documented in Section 5.1 of this SER, the staff finds that the package description and evaluation satisfies the shielding requirements of 10 CFR Part 71
- As documented in Section 5.2 of this SER, the staff finds that the source specification used in the shielding evaluation is sufficient to provide a basis for evaluation of the package against the shielding requirements of 10 CFR Part 71

- As documented in Section 5.3 of this SER, the staff finds that the models used in the shielding evaluation are described in sufficient detail to permit an independent review and independent calculations of the package shielding design
- As documented in Section 5.4 of this SER, the staff finds that the external radiation levels satisfy the requirements of 10 CFR 71.47 for packages transported by an exclusive-use vehicle
- As documented in Section 5.3 of this SER, the staff finds that the radiation levels will not significantly increase during NCT consistent with the tests specified in 10 CFR 71.71
- As documented in Section 5.3 of this SER, the staff finds that the maximum external radiation level at one meter from the external surface of the package will not exceed 1 rem/hr during HAC consistent with the tests specified in 10 CFR 71.73

6.0 CRITICALITY REVIEW

NAC submitted an application for an amendment to the NAC-STC CoC to incorporate Dairyland Power Cooperative LACBWR spent fuel assemblies as approved contents for transport in the NAC-STC system. This amendment is the 12th revision to CoC No. 9235.

The NAC-STC system has previously been approved to store fuel assemblies from Yankee Nuclear Power Plant and fuel assemblies from the Connecticut Yankee Atomic Power Company. Both are decommissioned pressurized water reactors.

The staff reviewed the addition of the new contents using the guidance in Section 6 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March, 2000. The staff's evaluation of the applicant's changes to the criticality evaluation follows.

6.1 Description of Criticality Design

The staff reviewed the General Information section in Chapter 1 of the NAC-STC SAR as well as any additional information in the Criticality Section, Chapter 6, of the NAC-STC SAR. The staff verified that the information is consistent as well as all descriptions, drawings, figures and tables are sufficiently detailed to support an in-depth staff evaluation.

6.1.1 Packaging Design Features

The applicant provided drawings of the package in Section 1.4.3.2 of the NAC-STC SAR. The staff reviewed these drawings and found that they sufficiently describe the locations, dimensions and tolerances of the containment system, basket, and neutron absorbing material. Therefore the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5) with respect to the criticality evaluation.

For criticality control, the NAC-STC design relies upon neutron absorbing material (BORAL) in the basket as well as fixed geometry of the fuel assemblies and a restricted loading pattern.

6.1.2 Codes and Standards

The applicant identified the regulations in 10 CFR Part 71 that are applicable to the criticality design of the package. The staff finds that these regulations are appropriately identified. The applicant also identifies standards applicable to the validation and application of the computer codes used in Section 6.8.5 of the NAC-STC SAR. The applicant has described and justified the basis and rationale used to formulate the package quality assurance program in Section 1.3.1 of the NAC-STC SAR. The quality assurance program provides control over all activities designated important to safety that are applicable to the design, fabrication, assembly, testing, maintenance, repair, modification, and use of the packaging for transportation of radioactive materials. This program meets 10 CFR Part 71 Subpart H requirements. The staff finds that this meets the requirements of 10 CFR 71.31(c), with respect to criticality safety, and finds it acceptable.

6.1.3 Summary Table of Criticality Evaluations

The applicant provided a summary table of the criticality evaluations in Table 6.8.1-2 of the NAC-STC SAR. The applicant performed an evaluation for an infinite array under NCT. The single package evaluation is under HAC conditions and bounds that of the NCT conditions.

The applicant shows that the limiting conditions for the NAC-STC with LACBWR fuel loading is the single cask under hypothetical accident conditions with flooded damaged fuel canisters in the exterior fuel locations, no moderation for the interior undamaged fuel, no moderation in the spaces between the canister and the cask and full external moderation. This analysis gives a maximum k-eff of 0.93738 and includes two times the standard deviation (2σ).

The applicant shows that the maximum k-eff for the limiting configuration is less than the upper subcriticality limit (USL) of 0.9376. The staff finds that this meets the requirements of 10 CFR 71.55(b), (d) and (e).

6.1.4 Criticality Safety Index

Section 6.8.1 of the NAC-STC SAR states that the Criticality Safety Index (CSI) is 100. Per 10 CFR 71.59(b) the value of N is 0.5. The applicant calculated an infinite array for NCT (5N) and a single cask for HAC (2N). The staff finds that these array sizes are acceptable and meet the requirements of 10 CFR 71.59(a)(1) and 10 CFR 71.59(a)(2). In addition, the staff finds that the licensee meets 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Spent Nuclear Fuel Contents

The applicant proposes to add new fuel types as part of this amendment request. These are Allis Chalmers 10x10 fuel with a maximum enrichment of either 3.64% or 3.94%, and Exxon Nuclear Company 10x10 fuel with a maximum enrichment of 3.71%. Both fuel types are stainless steel clad. Table 6.8.1-1 of the NAC-STC SAR lists the nominal design parameters for each fuel type and Table 6.8.2-1 of the NAC-STC SAR has additional fuel assembly characteristics. The staff verified that these are consistent with the fuel parameters listed in Table 1.4-4 of the NAC-STC SAR.

The staff finds that this meets the requirements of 10 CFR 71.31(a)(1), 10 CFR 71.33(b)(1), 10 CFR 71.33(b)(2), and 10 CFR 71.33(b)(3) because the package and contents are adequately defined.

The applicant evaluated each of the new fuel types individually and determined that the 3.94% enriched Allis Chalmers fuel has the highest reactivity. In addition the applicant evaluated the effect of assuming a "homogenized" planar average enrichment of the Exxon type fuel rather than the varied radial enrichments. The applicant determined that using the "homogenized" planar averaged enrichment for the Exxon type fuel is statistically equivalent to having discrete radial enrichments. The staff finds this acceptable. The results of these studies are shown in Table 6.8.4-1 of the NAC-STC SAR.

For the calculations presented in the summary table of the criticality evaluations in Table 6.8.1-2 of the NAC-STC SAR, the applicant states that Exxon fuel is located in the 36 interior locations and damaged Allis Chalmers fuel is located in the exterior 32 locations. Based on the results of the reactivity calculations for the individual fuel types and the allowed loading pattern in Figure 6.8.1-1 of the NAC-STC SAR, the staff finds that the analyzed loading pattern is acceptable and that it bounds all other possible loading patterns.

The applicant assumes 96% theoretical density for the UO_2 pellets. The staff finds that this is conservative and acceptable because it bounds LACBWR fuel assembly material mass.

The applicant does not take credit for burn-up. All assemblies are assumed to be fresh fuel. The staff finds this conservative and acceptable.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The staff verified that for the criticality analyses that the applicant determined and used appropriate fuel and package dimensions. The applicant uses nominal design dimensions for the cask components. The staff finds that the cask does not significantly contribute to reactivity and therefore nominal dimensions are acceptable. The applicant performed calculations to determine the most reactive configuration considering manufacturing tolerances of the basket as well as fuel shifts. The applicant created a "combined shifted/tolerance" model. This is described in Section 6.8.4.2.1 of the NAC-STC SAR. The staff finds that the applicant has appropriately considered the manufacturing tolerances and component shifts.

The staff viewed Section 2 (structural evaluation) and 3 (thermal evaluation) of the NAC-STC SAR to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents. Under hypothetical accident conditions, the applicant assumes that there is no impact limiter and no neutron shield. The applicant states that structural analyses for normal and accident conditions demonstrate that no operating condition induces geometry variations in the system beyond those allowed by the manufacturing tolerances.

The staff examined the sketches of the model used for the criticality calculations and verified that the dimensions and materials are consistent with those in the drawings of the actual package. The applicant discusses the differences between their model and the package in Section 6.8.3.2.2 of the NAC-STC SAR. The staff reviewed these differences and finds that they will not cause any substantial effect on the criticality analysis and finds them acceptable.

The applicant found an optimum moderator density both inside the canister and within the gap between the cask and the canister. The applicant also found the optimum moderation with respect to pellet to clad gap flooding, preferential flooding and partial flooding. The staff finds this acceptable.

The staff verified that the applicant has a heterogeneous model of each fuel rod. The staff finds this acceptable.

6.3.2 Material Properties

The staff verified that the appropriate mass fractions and densities are provided for all materials used in the models of the packaging and contents. The applicant provided this information in Tables 6.8.3-7 and 6.8.3-8 of the NAC-STC SAR. The staff finds that the values used are typical values for the commonly used materials and are reasonable for use in the criticality analysis. The only material in the cask that was adjusted for accident conditions is the NS-4-FR used for the neutron shield. The applicant assumes that this material is not present during accident conditions. The staff finds that the material properties are consistent with the package under the tests of 10 CFR 71.71 and 10 CFR 71.73 and finds this acceptable.

The applicant reduced the density of the neutron absorber material (BORAL) to 75%. The staff finds that this is acceptable. The applicant tests the BORAL to ensure the presence, proper distribution, and minimum weight percent of Boron-10. These tests are described in Section 8.1.8 of the NAC-STC SAR. The staff finds this acceptable.

Section 3.3.2 of the NAC-STC SAR states that the maximum expected weight loss of the NS-4-FR neutron shield material will be less than 2% after a 20 year period if the NAC-STC package is maintained within normal operating temperatures. Since the neutron shield is not necessary for criticality control, this is acceptable for the proposed additional contents of the LACBWR fuel.

6.3.3 Computer Codes and Cross Section Libraries

The applicant performs the criticality evaluations using the MCNP5 Release 1.30 three-dimensional Monte Carlo code and continuous energy cross sections. The MCNP5 code is widely used in these types of applications and the staff finds it is appropriate for this application.

The applicant is using cross section data from various revisions of the ENDF/B-VI library with the exception of tin, where ENDF data is not available and so ENDL data is used. The applicant states that these are the same cross sections used to perform the validation of the code. The staff finds the cross sections used are appropriate for use with the NAC-STC LACBWR application.

The staff verified that the applicant provided representative input files. The staff also verified that the information regarding the model configuration, material properties and cross sections is properly represented in the input files. The staff reviewed the key input data for the criticality calculations specified in the input files and finds them acceptable. The staff viewed the output files provided and determined that they have proper convergence and that the calculated k-eff values from the output files agree with those reported in the text.

6.3.4 Demonstration of Maximum Reactivity

The staff reviewed the NAC-STC SAR and determined that both types of spent nuclear fuel contents are considered in the criticality models. The applicant demonstrates that the Allis Chalmers fuel type is more reactive than the Exxon fuel, however they model both fuel types because loading is restricted as shown in Figure 6.8.1-1 of the NAC-STC SAR.

Damaged fuel is allowed in the periphery of the basket where there are damaged fuel canisters (DFCs). This is mostly restricted to Allis Chalmers fuel but up to 4 Exxon fuel assemblies are allowed. The Exxon fuel is less reactive and therefore will not adversely affect the criticality safety when placed in the peripheral locations. This is demonstrated in Table 6.8.4-14 of the NAC-STC SAR.

The applicant performs calculations of a homogenous mixture of fuel to simulate rubbelized fuel in the DFCs. The results of these calculations are shown in Table 6.8.4-11 of the NAC-STC SAR. The applicant shows that the reactivity of the homogenous cases is lower than that of the heterogeneous fuel array.

For damaged fuel evaluations, the applicant assumes the maximum square pitch allowed within the DFC opening. Table 6.8.4-10 of the NAC-STC SAR shows that the larger pitch gives the most reactive conditions. The applicant also assumes a bare array (i.e. no cladding). This is conservative because it removes any absorption by the cladding. The staff finds that the damaged fuel evaluations are conservative.

The applicant determined the optimum internal moderation. They performed calculations varying the moderator density conditions both inside and outside the cask and also performed calculations assuming partial flooding. They found that the most reactive moderator density is where the interior portion of the transportable storage canister (TSC) is voided and the exterior (DFC locations) is fully flooded. The results of the applicant's analyses are shown in Figures 6.8.4-1, 6.8.4-2, and 6.8.4-3 of the NAC-STC SAR. The applicant also performed partial drain down studies to show that when steel from the lid is acting as a reflector rather than the water above the top of the fuel, the steel reflector does not produce a significant change in reactivity.

The applicant modeled a single cask and an infinite array of casks simulated by modeling a reflective square, cylindrical and hexagonal boundary conditions around each cask. Single cask models have 20 cm of water reflector around them. Under normal conditions, the applicant modeled an infinite array of casks and shows that the exterior (interstitial) flooding of the casks makes no difference on the reactivity of the system. This is shown in Table 6.8.4-15 of the NAC-STC SAR. In addition this table shows that the single cask (CSI=100) evaluation produced higher reactivity under accident conditions.

The staff finds that the applicant's analysis demonstrated that they have found the maximum reactivity per the requirements of 10 CFR 71.55(b).

6.3.5 Confirmatory Analysis

The staff performed independent calculations to verify the k-eff of the NAC-STC with the LACBWR fuel. The staff constructed its model using design information found in the NAC-STC SAR. The staff used the KENO6 code with the 238-group cross section library derived from ENDF-VI data.

The staff used the fuel loading pattern as described in Figure 6.8.1-1 of the NAC-STC SAR. In Slot A the staff assumed undamaged Exxon Fuel assemblies. In Slot B the staff assumed damaged Allis Chalmers Fuel assemblies with an enrichment of 3.64%. In Slot C, the staff assumed damaged Allis Chalmers fuel assemblies with an enrichment of 3.94%.

The damaged fuel assemblies were modeled as a bare array (no cladding material) of UO_2 rods. Based on the sensitivity studies from the applicant, the staff assumed the following for the damaged fuel assemblies:

- Heterogeneous array of rods
- Pitch of 0.6 inches (Table 6.8.4-10 of the NAC-STC SAR shows that reactivity is increased with increased pitch, and this is the maximum pitch allowed in the DFC space)
- All rods present within the damaged fuel rod assemblies (Table 6.8.4-9 of the NAC-STC SAR shows that reactivity increased with one missing rod. The increase was not substantial so for simplicity, the staff assumed all rods were present)

The positioning of the assemblies of the staff's model was based on Drawing No. 630045-893 in the NAC-STC SAR. The staff assumes void within the empty spaces within the undamaged (Exxon) fuel assemblies, and basket. The staff assumed void within the gap between the fuel rod and the cladding. The staff ran a sensitivity study and found that this condition was more reactive than a flooded gap. The empty spaces within the damaged (Allis Chalmers) fuel assembly locations are assumed to be flooded with full density water. The staff assumes the cask gap (area between the canister and cask that is filled with the axial spacers) is voided. The staff performed an additional calculation and confirmed that this is more conservative than flooding this area. The staff assumed no external reflection. The staff performed a calculation and shows that although the results are statistically similar, this gave slightly more conservative results.

The staff used several simplifying assumptions similar to that of the applicant. The staff assumed no structural materials within the basket besides the fuel tubes. The staff assumed that all assembly hardware could be modeled as a top and bottom cap of equivalent stainless steel. The staff assumed no aluminum or additional absorber sheets associated with the enlarged and damaged fuel cans. The staff's model has no impact limiter.

The k-eff from the staff's calculation is 0.9313. This is within 1% of the applicant's calculated k-eff value. The staff finds that this helps to demonstrate that the features important to criticality are sufficiently described and that the applicant has addressed the most reactive conditions and that the reported k-eff is conservative and that the applicant has appropriately modeled the cask geometry and materials.

6.4 Single Package Evaluation

6.4.1 Configuration

The staff verified that the applicant's evaluation demonstrates that a single package is subcritical under both normal conditions of transport and hypothetical accident conditions. Under HAC the applicant assumes no impact limiter and no neutron shield and performs calculations with no moderation for the interior fuel and full flooding of the exterior fuel housed in DFCs.

The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents.

The staff determined that water moderation is in the most reactive extent as required by 10 CFR 71.55(b).

The NAC-STC SAR states that all single package analyses include full reflection of 20 cm water on all sides. The staff finds that this meets the requirement in 10 CFR 71.55(b)(3).

6.4.2 Results

6.4.2.1 NCT

The staff confirmed that the results of the applicant's criticality calculations are consistent with the information presented in the summary table discussed in Section 6.1.3 of this SER. The maximum k-eff for a single package is 0.93738. The single package model uses modeling assumptions consistent with HAC and is bounding for NCT.

Since k-eff is less than the USL of 0.9376 under the tests specified in 10 CFR 71.71, the staff verified that this meets the requirements of 10 CFR 71.55(d)(1) which requires that the contents be subcritical.

Since the applicant performs evaluations using reasonably bounding geometry of the fuel and basket to perform the criticality calculations, the staff verified that the geometric form of the package contents could not be altered in such a way that would affect the conclusions from the criticality safety analyses. The staff finds that the applicant meets 10 CFR 71.55(d)(2).

The applicant performed calculations where moderation is present to such an extent to cause maximum reactivity consistent with the chemical and physical form of the material. The staff finds that this meets 10 CFR 71.55(d)(3).

Under the tests specified in 10 CFR 71.71, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention including (1) the total volume of the packaging will not be reduced on which the criticality safety is assessed, (2) the effective spacing between the fissile contents and the outer surface of the packaging is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm cube. The staff finds that this meets the requirements in 10 CFR 71.55(d)(4).

6.4.2.2 HAC

The staff confirmed that the results of the applicant's criticality calculations are consistent with the information presented in the summary table discussed in Section 6.1.3 of this SER.

Since k-eff is less than the USL of 0.9376 under the tests specified in 10 CFR 71.73, the staff verified that this meets the requirements of 10 CFR 71.55(e) which requires that under HAC the contents be subcritical.

The staff verified that (1) the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents, (2) water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and (3) there is full reflection by water on all sides, as close as is consistent with the damaged condition of the package. This meets the requirements of 10 CFR 71.55(e)(1) through (3).

6.5 Evaluation of Package Arrays

6.5.1 Configuration

The applicant specified a CSI of 100 therefore the array calculations are for a single package for HAC. For NCT the applicant modeled an infinite array using several different boundary conditions to simulate an infinite square or hexagonal array. For NCT, the applicant assumes no internal moderation or external moderation. The staff finds this consistent with the results of the tests specified in 10 CFR 71.71. The applicant assumes a cask separation of 0.5 cm and 20 cm (which is considered fully reflected). The applicant did not present an evaluation in the NAC-STC SAR that demonstrates that 0.5 cm is the most reactive cask separation for the array configurations. The applicant bases the 0.5 cm separation on previous NAC-STC content evaluations. The staff finds that since the results of Tables 6.8.4-15 and 6.8.4-16 of the NAC-STC SAR show that the reactivity of the cask is not highly sensitive to the separation of the casks that this is a reasonable assumption and is acceptable. In addition, the applicant has other conservative modeling assumptions (e.g., fresh fuel) that would negate any uncertainty due to this calculation assumption.

The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents.

6.5.2 Results

6.5.2.1 NCT

The maximum k-eff for the NCT array analyses is 0.36965. Since k-eff for an infinite array is less than the USL of 0.9376 under the tests specified in 10 CFR 71.71, the staff verified that this meets the requirements of 10 CFR 71.59(a)(2) which requires that an array size 5N of undamaged packages be subcritical.

6.5.2.2 HAC

The applicant did not perform calculations for an array size greater than one. The staff finds this acceptable because the CSI is 100. The staff verified that this meets the requirements of 10 CFR 71.59(a)(2) which requires that an array size 2N of packages under HAC be subcritical.

6.6 Benchmark Evaluations

The applicant performs the criticality evaluations using the MCNP5 Release 1.30 three-dimensional Monte Carlo code and continuous energy cross sections. The applicant performed benchmarks with the same computer code and cross section set.

6.6.1 Experiments and Applicability

The applicant performed benchmark comparisons and determined a USL based on the guidance published in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages." The staff finds the use of this guidance acceptable.

The staff verified that the following important design parameters for the NAC-MPC-LACBWR system were within the benchmark experiments cited by the applicant.

- Enrichment
- Type of fissile material
- Fuel rod pitch and diameter
- B-10 plate loading
- EALF
- H/U-235 ratio

The fuel pellet outer diameter range of the benchmarks is larger than that of the Exxon fuel. In addition the applicant does not have any benchmark comparisons for stainless steel clad fuel. The staff does not find these differences significant and that any potential negative effects would be compensated for by conservative assumptions within their analysis (i.e., fresh fuel assumption, etc.).

6.6.2 Bias Determination

The applicant calculated a USL of 0.9376 using the USLSTATS code. This includes the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k-eff less than the USL is less than 0.95. The staff finds this acceptable.

6.7 Burnup Credit

The applicant does not request credit for burnup.

6.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the subcriticality requirements of 10 CFR Part 71.

7.0 PACKAGE OPERATIONS

Chapter 7 of the SAR provides procedures for package loading, unloading, and preparation of the empty package for transport. Sections 7.1.2.2, 7.1.3.2, 7.2.1, 7.2.2, 7.3.2, 7.3.3.2, 7.4.2, 7.4.3, and 7.6 provide revised operating procedures for this revision request.

The staff reviewed the Operating Procedures in Chapter 7 of the SAR to verify that the package will be operated in a manner that is consistent with its design evaluation. On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the proposed additional contents of the LACBWR fuel and other clarification changes in accordance with 10 CFR Part 71. Further, the CoC is conditioned such that the package must be prepared for shipment and operated in accordance with the Operating Procedures specified in Chapter 7 of the Safety Analysis Report.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff reviewed the revisions to Chapter 8 of the application to verify that the revised acceptance tests for the packaging meet the requirements of 10 CFR Part 71.

To support this revision request, Sections 8.0, 8.1.1, 8.1.7, 8.1.8, and 8.1.9 of the SAR were revised to describe the requirements for acceptance testing and maintenance to support this revision request.

Based on the statements and representations in the application, the staff concludes that the revised acceptance tests for the packaging meet the requirements of 10 CFR Part 71. Further, the CoC is conditioned to specify that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

CONDITIONS

The CoC has been revised as follows:

Condition No. 5(a)(2):

Specific descriptions of the Yankee-MPC and CY-MPC TSCs were added throughout. The description of the MPC-LACBWR TSC was added.

Condition No. 5(a)(3):

Eight drawings were revised and fourteen new drawings were added.

Condition No. 5(b)(1)(v):

This condition was added to detail the type and form of the LACBWR fuel contents.

Condition No. 5(b)(2):

An editorial change was made on page 12 of 15.

Condition No. 5(b)(2)(vi):

Condition was added to provide for the maximum quantity of material per package for the additional contents of the LACBWR fuel.

Condition No. 15:

Allows the use of Revision 11 of this certificate for one year.

New supplements were added to the references.

CONCLUSION

Based on the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. NAC-STC package design has been adequately described and evaluated and that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9235, Revision No. 12, on October 5 , 2010.