SAFETY EVALUATION REPORT

Docket No. 71-9270
Model No. UMS Universal Transport Cask Package
Certificate of Compliance No. 9270
Revision No. 0

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SUMMARY

By application dated November 30, 1997, as supplemented, NAC International (NAC), requested that the Nuclear Regulatory Commission approve the Model No. UMS Universal Transport Cask Package (UMS) as a Type B(U)F-85 package. Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concludes that the package meets the requirements of 10 CFR Part 71.

References

NAC application dated April 30, 1997.

Supplements dated June 18, 1999, May 31, June 29, August 8, and September 20, 2000, February 28, March 14, March 31, June 1, and November 16, 2001, and January 31, March 13, August 12, September 27, and October 21, 2002.

1.0 GENERAL INFORMATION

1.1 Packaging

The UMS is a canister-based system for the storage and transportation of spent nuclear fuel. The transportation component of the UMS system, designated the Universal Transport System, consists of a Universal Transport cask (UTC) body with a closure lid and energy-absorbing impact limiters loaded with a Transportable Storage Canister (TSC) containing either spent Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) nuclear fuel or Maine Yankee site specific contents including Greater than Class C (GTCC) waste.

The NAC-UMS is designed to transport up to 24 intact PWR spent fuel assemblies, 56 intact BWR spent fuel assemblies, GTCC waste, or site specific spent nuclear fuel with associated component hardware. Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Class 4 and 5 BWR fuel assemblies include the Zircaloy channels. The loading of site specific fuels that include control component hardware may require the use of a TSC that is longer than if the hardware were excluded. The spent fuel is loaded into a TSC which contains a stainless steel grid work referred to as a basket.

The cask body of the UMS is a right-circular cylinder of multi wall construction which consists of 304 stainless steel inner and outer shells separated by lead gamma radiation shielding which is poured in place. The inner and outer shells are welded to a 304 stainless steel top forging which mates to the cask lid. The inner shell is also welded to a 304 stainless steel bottom forging and the outer shell is welded to the bottom plate. The cask bottom consists of the bottom forging and bottom plate with neutron shield material sandwiched between them. Layers of 4.5 inches thick 304 stainless steel ring and two 0.75 inch stainless steel disks are located at the bottom lead annulus between the bottom forging and the outer shell.

Neutron shield material is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates.

The containment boundary of the UMS consists of the inner shell; bottom forging; top forging; cask lid and lid inner O-ring; vent port cover plate and vent port cover plate inner O-ring; and drain port cover plate and drain port cover plate inner O-ring.

There are five TSCs of different lengths, each to accommodate a different class of PWR or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and the lengths vary from about 175 to 192 inches long. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The TSC contains the basket and fuel assemblies or GTCC waste. Spacers are placed below each Class 1, 2, 4 or 5 canisters to locate and support the canister in the cask cavity.

The spacers are free standing structures that are confined in place by the bottom of the canister and the cask bottom inner surface. The spacer(s) ensure that the canister lid is laterally supported by the cask top forging when the cask is horizontal and minimizes axial movement of the canister. Each class 1 or class 2 PWR canister is positioned by a stainless steel spacer that is 16.75 inches or 7.65 inches in length, respectively. The class 5 BWR canister is located with a 1.5 inch aluminum spacer and the class 4 BWR canister is located by four 1.5 inch aluminum spacers. No spacers are used with the class 3 PWR canister.

The spent fuel basket design uses a series of high strength stainless steel PWR or carbon steel BWR support disks to support the fuel assemblies in stainless steel tubes. The PWR fuel tubes contain BORAL neutron poison on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing BORAL on two sides of the tubes; (2) tubes containing BORAL on one side; and (3) tubes containing no BORAL. Aluminum heat transfer disks are provided in both the PWR and BWR fuel baskets to enhance thermal performance of the basket. The heat transfer disks are supported by stainless steel tie rods and split spacers that maintain the basket assembly configuration.

The GTCC waste canister is essentially identical to the class 1 TSC, except for the placement of lifting lugs and the placement of a key way within the canister. The GTCC basket is constructed of Type 304 stainless steel and consists of primarily a cylinder with a 3-inch thick wall closed at the bottom end with a 3-inch thick plate. The cylinder is centered in the GTCC waste canister by 14 Type 304 stainless steel support plates along its length. A 3-inch thick 304 stainless steel separator fixture divides the cylinder into two vertically stacked components, each 77 inches deep with a diameter of 47.8 inches.

The package has impact limiters at each end of the cask body. The impact limiters consist of a combination of redwood and balsa wood encased in Type 304 stainless steel. The impact limiters limit the g-loads acting on the cask during a transport drop load condition due to crushing of the redwood and balsa wood. The upper and lower impact limiters are bolted to the cask body by 16 equally spaced attachment rods with nuts.

The approximate dimensions and weights of the package are as follows:

Overall length (with impact limiters, in)	273.3
Overall length (without impact limiters, in)	209.3
Impact Limiter Outside diameter (in)	124.0
Outside diameter (without impact limiters, in)	92.9
Cavity diameter (in)	67.6
Cavity length (in)	192.5
Cask lid thickness (in)	6.5
Bottom thickness (in)	10.3
Inner shell thickness (in)	2.0
Outer shell thickness (in)	2.75
Gamma shield thickness (in)	2.75
Radial neutron shield thickness (in)	4.50

Transportable Storage Canister

Shell thickness (in)	0.625
Shell bottom (in)	1.75
Shield lid thickness (in)	7
Structural lid thickness (in)	3
Outer diameter (in)	67
Internal cavity diameter (in)	65.8
Internal fuel cavity length (in), depending on class	163-180
Overall length (in), depending on class	175-192

Fuel Basket

Basket assembly length (in), depending on class	162-180
Basket assembly diameter (in)	65.5
Number of support disks, depending on class	30-41
Number of heat transfer disks, depending on class	17-33

Total weight (pounds) including cask, basket, impact limiters, fuel, canister with lids, cask lid, and spacers for each fuel class is approximately:

Class 1 (PWR)	251,000
Class 2 (PWR)	252,000
Class 3 (PWR)	249,000
Class 4 (BWR)	256,000
Class 5 (BWR)	255,000

1.2 Type and Form of Material

The NAC-UMS package is designed to transport four types of contents as listed below:

a. 24 intact irradiated PWR fuel assemblies within a TSC;
b. 56 intact irradiated BWR fuel assemblies within a TSC;
c. 24 irradiated intact or damaged PWR fuel assemblies and fuel debris from Maine Yankee within a TSC; or
d. GTCC waste from Maine Yankee within a TSC.

Each type of package contents is described in detail below.

1.2.1 Intact PWR assemblies

The NAC-UMS package is designed to transport 24 irradiated intact PWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. An empty fuel rod position must be filled with a solid filler rod, fabricated from either Zircalloy or Type 304 stainless steel, which displaces an equal to or greater volume that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with Zircaloy cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 1.2-1 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2-2 below. PWR fuel assemblies may include standard inserts such as guide tube thimble plugs and burnable poison rods.

The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.2 wt% ^{235}U respectively. Unenriched fuel assemblies are not authorized for loading into the TSC. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is 5 years. The maximum weight of UO_2 is 11.53 MTU per cask.

Table 1.2-1, Intact PWR Fuel Assembly Characteristics													
TSC Class[1]	Vendor[2]	Array	Max. Length (in)	Max. Width (in)	Max. Assembly Weight	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)	Min Guide Tube Thickness (in)
1	CE	14x14	157.3	8.11	1292	0.404	176[4]	0.590	0.438	0.024	0.380	137.0	0.040
1	Ex/ANF	14x14	160.2	7.76	1271	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14x14	159.8	7.76	1177	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14x14	159.8	7.76	1302	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15x15	159.8	8.43	1472	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17x17	159.8	8.43	1348	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17x17	159.8	8.43	1482	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17x17	160.1	8.43	1373	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15x15	165.7	8.54	1515	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17x17	165.8	8.54	1505	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16x16	178.3	8.10	1430	0.442	236[4]	0.506	0.382	0.023	0.3255	150.0	0.035
1	Ex/ANF[3]	14x14	160.2	7.76	1215	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15x15	147.5	8.20	1360	0.432	216	0.550	0.418	0.026	0.358	132.0	---
1	Ex/ANF[3]	15x15	148.9	8.25	1339	0.431	216	0.550	0.417	0.030	0.358	131.8	---
1	CE[3]	16x16	158.2	8.10	1300	0.403	236[4]	0.506	0.382	0.023	0.3255	136.7	0.035

[1] Minimum and Maximum initial Enrichments are 1.9 wt% ^{235}U and 4.2 wt% ^{235}U, respectively. All fuel rods are Zircaloy clad.

[2] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits in not restricted to the vendor(s) listed.

[3] 14x14, 15x15, and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively, These are not generic fuel assemblies provided to multiple reactors.

[4] Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 1.2-2, Loading Table for Intact PWR Fuel										
Minimum Initial Enrichment wt% ^{235}U (E)	Burnup $\leq$ 30 GWD/MTU Minimum Cooling Time (years)					30 < Burnup $\leq$ 35 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 $\leq$ E < 2.1	6	8	8	7	8	8	10	11	9	10
2.1$\leq$ E < 2.3	6	7	8	6	7	7	10	10	8	10
2.3$\leq$ E < 2.5	6	7	7	6	7	7	9	10	8	9
2.5 $\leq$ E < 2.7	6	7	7	6	7	7	9	9	7	8
2.7$\leq$ E < 2.9	6	7	7	6	7	6	8	9	7	8
2.9$\leq$ E < 3.1	5	7	7	6	6	6	8	8	7	8
3.1$\leq$ E < 3.3	5	6	7	6	6	6	8	8	7	7
3.3 $\leq$ E < 3.5	5	6	6	6	6	6	7	8	6	7
3.5 $\leq$ E < 3.7	5	6	6	6	6	6	7	7	6	7
3.7 $\leq$ E$\leq$ 4.2	5	6	6	6	6	6	7	7	6	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup $\leq$ 40 GWD/MTU Minimum Cooling Time (years)					40 < Burnup $\leq$ 45 GWD/MTU Minimum Cooling Time (years)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 $\leq$ E < 2.1	11	15	15	13	15	18	20	21	20	20
2.1$\leq$ E < 2.3	10	13	14	12	13	15	19	19	18	19
2.3$\leq$ E < 2.5	9	12	13	11	12	14	17	19	17	17
2.5 $\leq$ E < 2.7	9	12	12	10	11	12	16	18	15	17
2.7$\leq$ E < 2.9	8	11	11	9	11	11	15	18	14	17
2.9$\leq$ E < 3.1	8	10	10	9	10	10	14	18	13	15
3.1$\leq$ E < 3.3	7	10	10	9	10	10	13	17	13	15
3.3 $\leq$ E < 3.5	7	9	10	8	9	9	12	17	13	15
3.5 $\leq$ E < 3.7	7	9	10	8	9	8	11	17	12	15
3.7 $\leq$ E$\leq$ 4.2	7	8	10	8	8	8	11	15	12	14

1.2.2 Intact BWR assemblies

The NAC-UMS package is designed to transport 56 irradiated intact BWR fuel assembles within the TSC. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks.

For BWR fuel, the initial enrichment limit (the enrichment of the as-delivered fresh fuel assembly) represents the maximum peak planer-average enrichment allowed for loading into the TSC. The peak planar-average enrichment is defined to be the maximum planar-average enrichment at any height along the axis of the fuel assembly.

The fuel assemblies consist of uranium dioxide pellets with Zircaloy cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 1.2-3 below. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2-4.

BWR intact fuel assemblies are authorized with or without channels based on a maximum channel width of 120 mils. The minimum and maximum allowable assembly average enrichment for loading is 1.9 wt% ^{235}U and 4.0 wt% ^{235}U respectively. The maximum burn up of the spent fuel assemblies is 45,000 MWD/GTU and the minimum cool time is six years. The maximum weight of UO_2 is 11.08 MTU per cask. Unenriched fuel assemblies are not authorized for loading into the TSC. BWR fuel assemblies with unenriched axial blankets must have an enriched central fuel region and are acceptable for loading into a TSC if the minimum fuel enrichment of the central region is 1.9 wt% ^{235}U. Any empty fuel position must be filled with a solid filler rod fabricated from either Zircalloy or Type 304 stainless steel.

Table 1.2-3, Intact BWR Fuel Assembly Characteristics

Canister Class[1]	Vendor[4]	Array	Max. Length (in)	Max. Assembly Width (in)[5]	Max. Assembly Weight (lb)[6]	Max MTU	No of Fuel Rods	Max Pitch (in)	Min Rod Dia (in)	Min Clad Thick (in)	Max Pellet Dia (in)	Max Active Length (in)[2]
4	Ex/ANF	7x7	171.3	5.51	620	0.196	48	0.738	0.570	0.036	0.490	144
4	Ex/ANF	8x8	171.3	5.51	563	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9x9	171.3	5.51	557	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7x7	171.1	5.51	681	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7x7	171.2	5.51	681	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8x8	171.1	5.51	639	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8x8	171.1	5.51	681	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8x8	176.1	5.51	588	0180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9x9	176.1	5.51	576	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[6]	Ex/ANF	9x9	176.1	5.51	576	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7x7	175.9	5.51	683	0.198	49	0.738	0.563	0.032	0.487	144.0
5	GE	8x8	176.1	5.51	665	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8x8	175.9	5.51	681	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9x9	176.1	5.51	646	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9x9	176.1	5.51	646	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

[1] Maximum Peak Planar Average Enrichment 4.0 wt%^{235}U. Minimum enrichment is 1.9 wt%^{235}U. All fuel rods are Zircalloy clad.

[2] 150 inch active fuel length assemblies contain 6 inch natural uranium blankets on top and bottom.

[3] Shortened active fuel length in some rods.

[4] Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is not restricted to the vendor(s) listed.

[5] Assembly width including channel. Unchanneled or channeled may be loaded based on a maximum channel thickness of 120 mils.

[6] Exxon/ANF assembly weights are listed without channel.

Table 1.2-4, Loading Table for Intact BWR Fuel						
Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cooling Time (years)			30 < Burnup ≤ 35 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	8	8	8	14	13	15
2.1≤ E < 2.3	7	7	8	12	12	13
2.3≤ E < 2.5	7	7	7	11	10	11
2.5 ≤ E < 2.7	7	6	7	9	9	10
2.7≤ E < 2.9	6	6	6	9	8	9
2.9≤ E < 3.1	6	6	6	8	8	8
3.1≤ E < 3.3	6	6	6	7	7	8
3.3 ≤ E < 3.5	6	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	7	7	7
3.7 ≤ E≤ 4.0	6	6	6	7	7	7
Minimum Initial Enrichment wt% ^{235}U (E)	35 < Burnup ≤ 40 GWD/MTU Minimum Cooling Time (years)			40 < Burnup ≤ 45 GWD/MTU Minimum Cooling Time (years)		
	9x9	8x8	7x7	9x9	8x8	7x7
1.9 ≤ E < 2.1	24	23	25	34	33	35
2.1≤ E < 2.3	21	20	22	31	30	32
2.3≤ E < 2.5	19	18	20	29	28	29
2.5 ≤ E < 2.7	17	16	17	26	25	27
2.7≤ E < 2.9	14	14	15	24	23	24
2.9≤ E < 3.1	13	12	13	21	20	22
3.1≤ E < 3.3	11	11	12	19	18	20
3.3 ≤ E < 3.5	10	10	11	17	16	18
3.5 ≤ E < 3.7	10	9	10	15	14	16
3.7 ≤ E≤ 4.0	10	9	10	14	13	15

1.2.3 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The NAC-UMS package is designed to transport 24 irradiated intact or damaged PWR fuel assemblies, canistered fuel debris, and GTCC waste within the TSC from the Maine Yankee Reactor. The standard Maine Yankee fuel assembly is the CE 14x14 as discussed in Section 1.2.3.1 of the Safety Evaluation Report (SER).

In the course of reactor operations, some of the 14x14 assemblies were modified to change the standard configuration. These modifications included a) the removal of fuel rods without replacement; b) the replacement of removed fuel rods or burnable poison rods with rods of a different material, such as stainless steel, or with fuel rods of a different enrichment; and c) the insertion of control elements, or instruments or plug thimbles, in guide tube positions. In addition to the modified fuel assemblies, there are fuel assemblies that were designed with variable enrichment and axial blankets. These fuel assemblies are not modified, but differ from the cask design basis fuel assemblies.

Stainless steel spacers may be used in canisters to axially position PWR intact fuel assemblies that are shorter than the available cavity length. The minimum length of the PWR intact fuel assembly internal structure and bottom end fitting and/or spacers will ensure that the minimum distance to the fuel region for the base of the canister is 3.2 inches.

Unenriched fuel assemblies are not authorized for loading.

The following are the allowable Maine Yankee site specific contents:

1.2.3.1 Maine Yankees' site specific contents not requiring preferential loading patterns:

a) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly characteristics in Table 1.2-1. The maximum fuel assembly weight, including other associated hardware is 1,515 pounds. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

b) Irradiated Maine Yankee CE 14 x 14 PWR intact fuel assemblies may contain inserted control element assemblies (CEA), in-core instrument (ICI) thimbles or CEA plugs. CEAs or CEA plugs may not be inserted in damaged fuel assemblies, consolidated fuel assemblies or assemblies with irradiated stainless steel replacement rods. Fuel assemblies with a CEA or CEA plug inserted must be loaded in a Class 2 canister and cannot be loaded in a Class 1 canister. Fuel assemblies without an inserted CEA or CEA plug, including those with inserted ICI Thimbles, must be loaded in a Class 1 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5 except for those assemblies containing ICI thimbles which must meet the specifications of Table 1.2.6.

c) PWR intact fuel assemblies with fuel rods replaced with stainless steel or Zircaloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt%. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.7.

d) PWR intact fuel assemblies with fuel rods having variable enrichments with a maximum rod enrichment up to 4.21 wt% ^{235}U and that also has a maximum planar average enrichment up to 3.99 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

e) PWR intact fuel assemblies with annular axial end blanket enrichments up to 2.6 wt% ^{235}U. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

f) PWR intact fuel assemblies with burnable poison rods or solid filler rods may occupy up to 16 of 176 fuel rod positions. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

g) PWR intact fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs, as long as the assembly can be handled safely by normal means. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

1.2.3.2 Maine Yankee site-specific allowable contents requiring preferential loading based on shielding, criticality, or thermal constraints (Maine Yankee CE 14 x 14 intact PWR fuel assemblies). A PWR basket fuel diagram can be found on Figure 1.2.1.

a) Maine Yankee CE 14 x 14 PWR intact fuel assemblies with a burn up between 45,000 and 50,000 MWD/MTU meeting the following requirements for verification of the oxide layer thickness and high burn up fuel requiring preferential loading in the peripheral PWR fuel basket positions:

A verification program is required to determine the oxide layer thickness on high burn up fuel by measurement or by statistical analysis. A fuel assembly having a burn up between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burn up. The verification program shall be capable of classifying high burn up fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

I. A HIGH BURN UP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

II. A HIGH BURN UP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.

b) PWR intact fuel assemblies with up to 176 fuel rods missing from the fuel assembly lattice. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5. These assemblies must be placed in a corner PWR fuel basket loading position.

c) PWR intact fuel assemblies with a burnable poison rods replaced by a hollow Zircaloy rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.   These assemblies may be placed in a corner PWR fuel basket loading position.

d) Intact fuel assemblies with a start-up source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 canister. Only one start-up source may be loaded in any fuel assembly or any canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5.   These assemblies may be placed in a corner PWR fuel basket loading position.

e) PWR intact fuel assemblies with CEA ends (fingertips) and/or an ICI segment inserted in corner guide tube positions. The assembly must also have a CEA plug installed. The assembly must be loaded in a PWR fuel basket corner position and must be loaded in a Class 2 canister. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5. CEA fingertips are not considered as CEA's for determination of minimum cool times.

1.2.3.3 Maine Yankee CE 14 x 14 PWR fuel enclosed in a Maine Yankee fuel can (MYFC).

All Maine Yankee CE 14 x 14 PWR fuel enclosed in an MYFC must be loaded in a Class 1 fuel canister in corner position of the PWR fuel basket. Up to 4 MYFC may be loaded into a TSC.   Intact Maine Yankee CE 14 x 14 PWR fuel may be loaded into a Maine Yankee fuel can. The contents that must be loaded in the Maine Yankee fuel can are:

a) PWR fuel assemblies with up to two intact or damaged fuel rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5 for intact fuel rods and Table 1.2.8 for damaged fuel rods.

b) A damaged fuel assembly with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A damaged fuel assembly cannot have an inserted CEA or other non-fuel component. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.8 for damaged fuel rods.

c) Individual intact or damaged fuel rods in a rod type structure, which may be a guide tube, to maintain configuration control. The combined

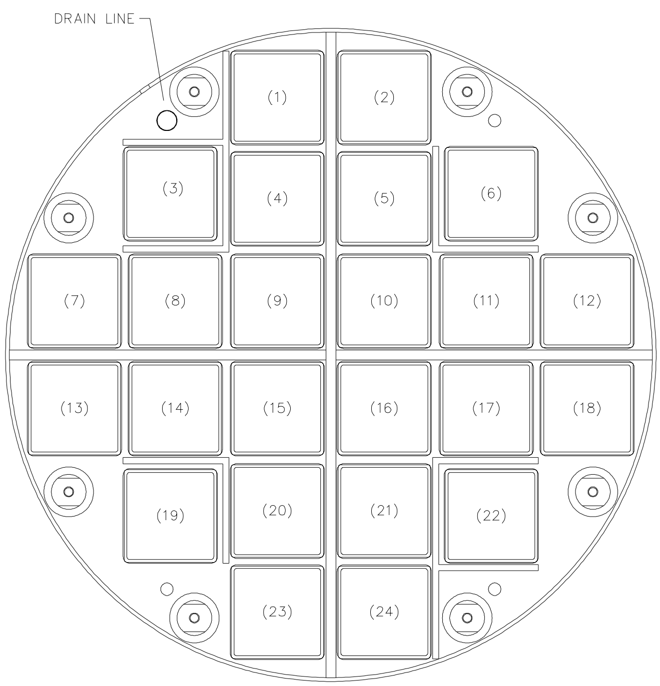
maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.5 for intact fuel rods and Table 1.2.8 for damaged fuel rods.

d) Fuel debris consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2.8 for damaged fuel rods.

e) Consolidated Fuel lattice and structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are 289 fuel rods having a total lattice weight less than or equal to 2,100 pounds. A consolidated fuel lattice cannot have an inserted CEA or other non-fuel component. Only one consolidated fuel lattice may be stored in any TSC. Fuel must be cooled a minimum of 24 years.

f) High burn up fuel assemblies not meeting the oxide layer thickness criteria previously defined in section 1.2.3.2 (a). The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 1.2-8.

PWR Basket Fuel Loading Position Diagram, Figure 1.2.1



1. Basket corner positions are positions 3, 6, 19, and 22 in the figure. Corner positions are also periphery positions.
2. Basket periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24 in the figure. Periphery positions include the corner positions.

Table 1.2-5, Loading Table for Maine Yankee CE 14x14 Fuel With and Without CEA Cooled to Indicated Time						
Burnup 30 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	6	6	7	6	6	6
2.1≤ E < 2.3	6	6	7	6	6	6
2.3≤ E < 2.5	6	6	6	6	6	6
2.5 ≤ E < 2.7	6	6	6	6	6	6
2.7≤ E < 2.9	6	6	6	6	6	6
2.9≤ E < 3.1	5	6	6	6	6	6
3.1≤ E < 3.3	5	5	6	6	6	5
3.3 ≤ E < 3.5	5	5	6	6	5	5
3.5 ≤ E < 3.7	5	5	6	5	5	5
3.7 ≤ E≤ 4.2	5	5	6	5	5	5

Loading Table for Maine Yankee CE 14x14 Fuel With and Without CEA Cooled to Indicated Time						
Burnup 35 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	8	8	9	8	8	8
2.1≤ E < 2.3	7	7	9	8	8	8
2.3≤ E < 2.5	7	7	8	7	7	7
2.5 ≤ E < 2.7	7	7	8	7	7	7
2.7≤ E < 2.9	6	7	7	7	7	7
2.9≤ E < 3.1	6	6	7	7	6	6
3.1≤ E < 3.3	6	6	7	6	6	6
3.3 ≤ E < 3.5	6	6	7	6	6	6
3.5 ≤ E < 3.7	6	6	6	6	6	6
3.7 ≤ E≤ 4.2	6	6	6	6	6	6

Table 1.2.5, Loading Table for Maine Yankee CE 14x14 Fuel With and Without CEA Cooled to Indicated Time						
Burnup 40 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	11	12	14	13	12	12
2.1≤ E < 2.3	10	10	13	11	11	11
2.3≤ E < 2.5	9	9	12	10	10	10
2.5 ≤ E < 2.7	9	9	10	9	9	9
2.7≤ E < 2.9	8	8	10	9	8	8
2.9≤ E < 3.1	8	8	9	8	8	8
3.1≤ E < 3.3	7	7	8	8	8	8
3.3 ≤ E < 3.5	7	7	8	7	7	7
3.5 ≤ E < 3.7	7	7	8	7	7	7
3.7 ≤ E≤ 4.2	7	7	7	7	7	7

Loading Table for Maine Yankee CE 14x14 Fuel With and Without CEA Cooled to Indicated Time						
Burnup 45 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	18	18	21	19	18	18
2.1≤ E < 2.3	15	16	19	17	17	16
2.3≤ E < 2.5	14	14	18	16	15	15
2.5 ≤ E < 2.7	12	13	16	14	14	13
2.7≤ E < 2.9	11	12	14	13	12	12
2.9≤ E < 3.1	10	11	13	12	11	11
3.1≤ E < 3.3	10	10	12	11	10	10
3.3 ≤ E < 3.5	9	9	11	10	10	10
3.5 ≤ E < 3.7	9	9	10	10	10	10
3.7 ≤ E≤ 4.2	9	9	10	10	10	10

Table 1.2.5, Loading Table for Maine Yankee CE 14x14 Fuel With and Without CEA Cooled to Indicated Time						
Burnup 50 GWD/MTU		Minimum Cool Time (Years) for				
Enrichment	No CEA (Class 1)	No CEA (Class 2)	5 Yr CEA	10 Yr CEA	15 Yr. CEA	20 Yr. CEA
1.9 ≤ E < 2.1	27	27	29	27	27	27
2.1≤ E < 2.3	24	24	27	25	24	24
2.3≤ E < 2.5	22	22	25	23	22	22
2.5 ≤ E < 2.7	19	19	23	21	20	20
2.7≤ E < 2.9	17	17	21	19	18	18
2.9≤ E < 3.1	15	16	19	18	18	18
3.1≤ E < 3.3	15	15	18	17	17	17
3.3 ≤ E < 3.5	15	15	17	17	17	17
3.5 ≤ E < 3.7	14	14	15	15	15	15
3.7 ≤ E≤ 4.2	14	14	15	15	15	15

Table 1.2-6, Loading Table (Years) for Maine Yankee CE 14x14 Fuel Containing ICI Thimbles					
Minimum Initial Enrichment wt% ^{235}U (E)	Burnup ≤ 30 GWD/MTU	30 < Burnup ≤ 35 GWD/MTU	35 < Burnup ≤ 40 GWD/MTU	40 < Burnup ≤ 45 GWD/MTU	45 < Burnup ≤ 50 GWD/MTU
1.9 ≤ E < 2.1	6	8	11	18	27
2.1≤ E < 2.3	6	7	10	16	24
2.3≤ E < 2.5	6	7	9	14	22
2.5 ≤ E < 2.7	6	7	9	13	19
2.7≤ E < 2.9	6	6	8	11	17
2.9≤ E < 3.1	5	6	8	10	15
3.1≤ E < 3.3	5	6	7	10	15
3.3 ≤ E < 3.5	5	6	7	9	15
3.5 ≤ E < 3.7	5	6	7	9	14
3.7 ≤ E≤ 4.2	5	6	7	9	14

Table 1.2-7, Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods					
Assy Number	Burnup (GWD/MTU)	Enrichment (wt %)	SSR Source (g/s/assy)	Cool Time (years)	Earliest Transportable
N420	45	3.3	2.1602E+13	10	Jan 2001
N842	35	3.3	3.1396E+12	6	Jan 2001
N868	40	3.3	5.2444E+12	7	Jan 2001
R032	45	3.5	1.4550E+13	9	Jan 2005
R439	50	3.5	1.3998E+13	14	Jan 2010
R444	50	3.5	5.5993E+13	19	Jan 2015

Table 1.2-8, Cool Time (years) for Maine Yankee CE 14x14 Damaged Fuel					
Minimum Initial Enrichment wt% ^{235}U (E)	Burnup $\leq$ 30 GWD/MTU	30 < Burnup $\leq$ 35 GWD/MTU	35 < Burnup $\leq$ 40 GWD/MTU	40 < Burnup $\leq$ 45 GWD/MTU	45 < Burnup $\leq$ 50 GWD/MTU
$1.9 \leq E < 2.1$	7	11	19	28	37
$2.1 \leq E < 2.3$	6	9	16	26	34
$2.3 \leq E < 2.5$	6	8	14	23	32
$2.5 \leq E < 2.7$	6	8	12	21	30
$2.7 \leq E < 2.9$	6	7	11	19	27
$2.9 \leq E < 3.1$	6	7	10	17	25
$3.1 \leq E < 3.3$	5	7	9	15	23
$3.3 \leq E < 3.5$	5	6	8	13	21
$3.5 \leq E < 3.7$	5	6	8	12	19
$3.7 \leq E \leq 4.2$	5	6	7	11	17

1.2.4 Greater Than Class C Waste from Maine Yankee

The NAC-UMS package is designed to transport Maine Yankee Greater Than Class C Waste within a TSC. Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53. The maximum curie inventory shall not exceed the values shown in Table 1.2-9.

Table 1.2-9, Maine Yankee GTCC Curie Inventory Limits per TSC

Radionuclide	Curie Inventory (Ci)/ TSC
H-3	3.00E+02
C-14	1.50E+02
Mn-54	3.50E+02
Fe-55	2.00E+05
Co-58	1.00E+01
Co-60	2.90E+05
Ni-59	8.20E+02
Ni-63	9.00E+04
Nb-94	1.00E+01
Tc-99	1.00E+01

1.2.4.1 Weight

The maximum content's weight shall not exceed 77,500 pounds. The total weight of the PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, shall not exceed 38,500 pounds. The total weight of the BWR assemblies shall not exceed 39,000 pounds. The total weight of the GTCC waste per canister shall not exceed 20,000 pounds in total or 10,000 pounds per compartment.

1.2.4.2 Decay Heat Limit

The maximum decay heat limit per package for PWR fuel for 24 assemblies is limited to the values in Table 1.2-10. The individual PWR assembly decay heat is limited to 0.83 kW. The maximum decay heat limit per package for BWR fuel is 16 kW for up to 56 BWR fuel assemblies. The individual BWR assembly decay heat is limited to 0.29 kW. The maximum decay heat for the GTCC canister is 4.5 kW.

Table 1.2-10, PWR Decay Heat Limits				
Cool Time (Years)	PWR Decay Heat Limit (kW) Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000[1]
5	20.0	20.0	19.9	19.3
6	19.5	19.3	19.2	18.7
7	17.8	17.8	17.7	17.2
10	17.4	17.3	17.2	16.8
15	16.8	16.8	16.7	16.5

[1]Maine Yankee PWR fuel assemblies

1.3 Conclusions

The applicant adequately described the contents of the package as required by 10 CFR 71.33(b). The staff agrees with the applicant's conclusion that the package meets the requirements of 10 CFR Part 71.

1.4 Transport Index for Criticality Control (Criticality Safety Index)

Minimum transport index to be shown
on label for nuclear criticality control: 0.0

1.5 Drawings

The package is constructed and assembled in accordance with NAC drawings:

790-209, Rev. 1	790-210, Rev. 1	790-500, Rev. 2	790-501, Rev. 3
790-502, Rev. 6	790-503, Rev. 2	790-504, Rev. 2	790-505, Rev. 1
790-508, Rev. 2	790-509, Rev. 2	790-516, Rev. 2	790-519, Rev. 1
790-520, Rev. 2	790-570, Rev. 4	790-571, Rev. 2	790-572, Rev. 4
790-573, Rev. 7	790-574, Rev. 3	790-575, Rev. 7	790-581, Rev. 5
790-582, Rev. 7	790-583, Rev. 4	790-584, Rev. 11	790-585, Rev. 8
790-591, Rev. 2	790-592, Rev. 5	790-593, Rev. 4	790-594, Rev. 2
790-595, Rev. 5	790-605, Rev. 8	790-611, Rev. 3	790-612, Rev. 3
412-501, Rev. 2	412-502, Rev. 2		

2.0 STRUCTURAL EVALUATION

The objective of this review is to verify that the structural performance of the package has been adequately evaluated for the tests and conditions specified under normal conditions of transport (NCT) and hypothetical accident conditions (HAC) and the

package design has adequate structural integrity to meet the requirements of 10 CFR Part 71.

2.1 Description of Structural Design

2.1.1 Descriptive Information, Including Weights and Centers of Gravity

The UMS system consists of four principal components: (1) the UTC, (2) the TSC, (3) the fuel basket, and (4) the impact limiters. Considering variable length of fuel assemblies, the system is configured for five classes of length combinations for the canister and basket to transport three classes of PWR spent fuel and two classes of BWR fuel. By replacing the basket inside the PWR Class 1 canister with a GTCC basket assembly, the system is used to transport Maine Yankees site-specific GTCC waste. A MYFC is also provided to accommodate Maine Yankee damaged fuel. Major structural features of these components are as follows.

2.1.1.1 Universal Transport Cask

The major features of the UTC include a multi-walled cask body of inner and outer stainless steel shells and an intermediate shell of lead gamma shielding, a bottom forging, a cask lid, two primary and two secondary lifting trunnions, and a pair of rotation pockets.

The containment boundary of the cask consists of: (1) a 4.3 inch-thick bottom forging, (2) a 2 inch-thick inner shell with an inside diameter of 67.6 inches, (3) a top forging, (4) a 6.5 inch-thick cask lid with 48 SB-637, Grade N07718, closure bolts, (5) vent and drain port cover plates, and (6) lid inner O-rings. The forgings, inner shell, and cask lid are all of Type 304 stainless steel construction.

The cask is equipped with four 6.5 inch-diameter lifting trunnions fabricated from the Type 17-4 PH stainless steel. The two diametrically opposite primary trunnions are welded into the 2.0 inch recesses in the top forging, and the two secondary trunnions are bolted to the top forging, each spaced at 90 degrees away from the primary trunnions. Two rotation pockets of the Type XM-19 stainless steel are welded to the outer shell of the cask body at about 17.6 in above the cask bottom. They are aligned with the two primary lifting trunnions with a 3.0 inch offset from the cask centerline to facilitate cask rotation during the up and down-ending operations. The trunnions and rotation pockets also function as tie-down devices for the package.

2.1.1.2 TSC

The TSC features a circular cylindrical shell welded to a bottom closure and closed at the top with a shield lid and a structural lid.

The TSC shell is fabricated from 5/8 inch-thick Type 304L stainless steel plates. The bottom closure is a 1.75 inch-thick Type 304L stainless steel plate joined to the shell by full penetration welds. The shield and structural lids are welded in place during the spent fuel loading operation. The TSC provides confinement for the spent fuel or GTCC waste during storage. No credit is taken for the TSC for its containment function during transport operations for intact fuel. However, the TSC serves as a separate inner container, per 10 CFR 71.63(b), for the transport of damaged fuel. Table 1.2-2 of the

SAR lists major physical design parameters of five TSC classes, including a common outside diameter of 67.06 inches, and a variable overall length ranging from 175.1 inches for the Class 1 PWR fuel application to 190.4 inches for the Class 5 BWR fuel.

2.1.1.3 Fuel Basket

The fuel basket is an assembly of fuel tubes laterally supported by a number of support disks which are axially retained by the spacers aligned on either eight radially located tie rods for the PWR or six for the BWR fuel. A top and a bottom weldment are placed at the basket ends to position and support the fuel tubes. Aluminum heat transfer disks, spaced midway between support disks, are provided to facilitate heat conduction from spent fuel assemblies to the TSC shell. Cutouts in the heat transfer disks are sized to allow fuel tubes and tie rods to pass through to prevent the disks from becoming load-bearing members for any structural loads other than dead loads. Type 304 stainless steel is used to construct the fuel tubes, which, together with the encased Boral sheets on tube external surfaces, provide spent fuel criticality control in the basket. Table 1.2-1 of the SAR lists major design parameters of the five classes of fuel baskets, including the Type 17-4 PH stainless steel for the 0.5 inch-thick PWR support disks and the SA-533, Type B, Class 2 carbon steel for the 0.625 inch-thick BWR support disks.

2.1.1.4 MYFC

The 162.8 inch-long MYFC is a 18-gauge, Type 304 stainless steel, square tube sized to fit within a standard PWR fuel tube. The can has a minimum internal width of 8.52 inches to allow it to hold an intact fuel assembly, a damaged fuel assembly, or the consolidated fuel in the Maine Yankee fuel inventory. The fuel can has a 0.63 inch-thick bottom plate and a 0.88 inch-thick top closure, each provided with four 1.4 inch-diameter wire screened holes to permit water to be drained out during loading operations. The assembly and details of the MYFC are presented in NAC Drawings 412-501 and 412-502.

2.1.1.5 Greater-Than-Class-C Waste Basket

The GTCC waste basket assembly is constructed of the Type 304 stainless steel. It consists primarily of a 3 inch-thick cylindrical shell welded to a 3 inch-thick bottom plate and closed at the top with a 3 inch-thick cover plate. The basket assembly is centered in a PWR Class 1 TSC, also called a GTCC waste container, by 14 Type 304 stainless rib ring plates, each 1-in thick and 65.3 inches in diameter. The basket cavity measures 48.8 inches in diameter and 157 inches in length. The basket cavity is divided into two axially stacked loading compartments, each 77 inches long, using a support weldment as a separator fixture. It is designed to transport up to 20,000 lbs. of GTCC waste with the maximum weight of GTCC waste in each compartment limited to 10,000 lbs. NAC Drawings 790-611 and 790-612 present design details for the waste basket and its TSC loading configuration, respectively.

2.1.1.6 Impact Limiters

Two impact limiters made of redwood and balsa wood are used to protect the cask for impact loading in a cask drop. NAC Drawings 790-506 and 790-507 presents design details, including the shell enclosure, gusset partitions, and specifications for densities, preparation, and bonding of wood blocks, for the upper and lower impact limiters of the

cask, respectively. The impact limiters, which overlap the cask ends, measure 124 inches in outside diameter and 43 inches long. The inside diameter of the upper impact limiter is slightly larger than that of the lower impact limiter to accommodate the cask lid.

2.1.1.7 Weights and Centers of Gravity

Tables 2.2.1 and 2.2.2 of the SAR list the calculated weights and centers of gravity of the major components and total systems for the three classes of PWR and two of BWR fuels, respectively. The center-of-gravity locations are identified along the cask vertical axis. For the transport ready configurations, with impact limiters attached, the largest system weight is 255,022 lbs., which is bounded by the design basis transport weight of 260,000 lbs. considered in the structural evaluation for cask lifting, tie-down, and free drops. The centers of gravity of the transportation ready configurations are at about the cask mid-height for all five fuel load classes.

The SAR states that the GTCC waste basket is designed to transport up to 20,000 pounds of the GTCC waste with a maximum weight of 10,000 lbs. in each of the two basket compartments. Table 2.2.3 of the SAR lists the weight of the loaded GTCC waste canister at 70,454 lbs., which is less than the loaded Class 1 canister of 70,705 lbs. The center of gravity of the transport ready cask containing the GTCC waste is calculated at a height of 108.54 inches above the cask bottom. It deviates insignificantly from that of the corresponding height of 107.99 inches for the cask containing the PWR Class 1 fuels.

2.1.2 Design Criteria

Section 2.1.2 of the SAR summarizes the structural design criteria for the UMS system, including applicable codes and standards as well as load combinations. These design criteria are reviewed as follows.

2.1.2.1 Codes and Standards

Section 2.1.2.1 of the SAR presents the applicable codes and standards for the structural design of the UMS system components. Additional details on allowable stress limits and performance criteria for miscellaneous structural failure modes are described in Sections 2.1.2.4 and 2.1.2.5 of the SAR, respectively.

The cask containment boundary and the TSC shell are designed per the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NB, stress requirements, which are essentially identical to those of Section III, Division 3, of ASME Code. Structural stability of the cask inner shell and the canister shell is evaluated in accordance with the interaction equations of ASME Code Case N-284-1. The fuel basket is evaluated in accordance with ASME Code, Section III, Subsection NG, for stresses and NUREG/CR-6322 for buckling. The cask lifting trunnions and lid lifting rings are evaluated per ANSI N14.6 and NUREG-0612, which envelop the requirements of 10 CFR 71.45(a). There exists no industry standard for the impact limiters, which are designed to absorb the impact energy by crushing energy-absorbing materials. Table 2.1.2-2 and 2.1.2-3 of the SAR lists stress allowables for containment and non-containment structures of the cask system, respectively. Table 2.1.2-1 of the SAR provides the justifications for exceptions to the ASME Section III, including those to Subsection NF for the GTCC waste basket.

The use of the codes and standards meets the intent of Regulatory Guide 7.6 and NUREG-1617, and is acceptable.

2.1.2.2 Load Combinations

The applicant performed stress analysis of the UMS system for the NCT and HAC of 10 CFR 71. Each load condition as characterized by a combination of initial conditions is evaluated for the most limiting total load effects on structural performance. Table 2.1.2-2 of the SAR lists initial conditions, including ambient temperature, solar insolation, decay heat, internal pressures, and fabrication stresses, for all individual load conditions. The load combination approaches follow Regulatory Guide 7.8 and are acceptable.

2.2 Material Properties

2.2.1 Materials and Material Specifications

The applicant provided a general description of the materials of construction in SAR Sections 1.2 and 2.1. Additional information regarding the materials, fabrication details, and testing programs can be found in SAR Sections 4.1 and 8.1. The staff reviewed the information contained in these sections and the information presented in the drawings to determine whether the UMS meets the requirements of 10 CFR 71.31(a)(1) and (c); 71.33; 71.35(a); 71.41; and, 71.43(d). In particular, the following aspects were reviewed: materials selection; applicable codes and standards; weld design and specification; bolt fabrication and preparation; chemical and galvanic reactions; and coatings compatibility.

2.2.1.1 Structural Materials

The main structural components of the UTC, including the top and bottom forgings, the port cover plates, the inner and outer shells, the cask lid and the bottom plate, are fabricated with Types 304 austenitic stainless steel. These types of steels were selected because of their high strength, ductility, resistance to corrosion and metallurgical stability. Because there is no ductile-to-brittle transition temperature in the range of temperatures expected to be encountered for these steels, their susceptibility to brittle fracture is negligible. The trunnions and rotation pockets are fabricated from Type 630, 17-4PH (i.e., precipitate hardened) stainless steel, and Grade XM-19 stainless steel, respectively. Type 17-4 precipitate hardened, stainless steel is heat treated to produce higher strengths (e.g., yield and ultimate tensile strengths) than those of Type 304 steel without a significant loss of corrosion resistance, while Grade XM-19 stainless steel is strengthened by nitrogen additions. The ductile-to-brittle transition temperatures of Type 630 and Grade XM-19 stainless steels are below the expected operating temperatures, so brittle fracture of these materials is not expected. The 48 cask lid bolts are fabricated from Type SB-637, Grade N07718, nickel alloy. This nickel alloy also has high strength, ductility and corrosion resistance and is not susceptible to brittle failure at the expected service temperatures.

Most of the structural components of the TSC and the GTCC Waste Canister are fabricated from Types 304 or 304L austenitic stainless steel. The support disks of the TSC basket used to store and transport PWR fuel are fabricated from Type 630, 17-4PH stainless steel. As noted above, Types 304 and 630 stainless steels do not experience a ductile-to-brittle transition, so they are not susceptible to brittle fracture. Further, since Type 304L has essentially the same structure and composition as Type 304, except it

has a lower carbon content, it is also not susceptible to brittle fracture. The support disks of the TSC basket used to store and transport BWR fuel are fabricated from ASME SA 533, Type B carbon steel. An electroless nickel metallic coating is applied to the surfaces of these support disks to protect the carbon steel from corrosion prior to, and during, immersion in the spent fuel pool. Brittle fracture of the carbon steel is not expected since the ductile-to-brittle transition temperature is below the expected operating temperatures.

The UMS impact limiters used during spent fuel transport are made of ASTM A 240 Type 304 stainless steel and wood. As noted above, Type 304 stainless steel is an austenitic material that is not susceptible to brittle fracture under the temperature conditions of transport.

The staff concludes that the material properties and characteristics needed to satisfy the functional safety requirements of the UMS will be maintained during transport. The staff concludes that the selection of these materials is acceptable for use in the TSC and the GTCC Waste Canister.

2.2.1.2 Non-Structural Materials

The neutron absorbers and gamma shields for UMS are fabricated from materials that perform satisfactorily under NCT and HAC. Chemical-copper grade lead (ASTM B29, 1992) is poured into the annulus of the UTC for gamma shielding. The thermal analyses of SAR Section 3 and SAR Tables 3.4-1 through 3.4-3 show that the temperatures of the lead during NCT are well below the melting point of this material (e.g., 600°F). Therefore, the staff concludes that the lead will undergo minimal slumping and will perform its intended function of gamma shielding. The cask also utilizes NS-4-FR neutron shielding. The NS-4-FR material is a high-hydrogen content, durable, fire-resistant material that has been used reliably in several storage cask systems. As SAR Tables 3.4-1 through 3.4-3 show, the temperatures experienced by the NS-4-FR (assuming a helium backfill gas which is the design basis configuration of the cask) are less than the temperature limit. The staff concludes that the chemical lead and the NS-4-FR neutron shielding material are suitable shielding materials for the UMS.

Criticality control in the basket is achieved by surrounding each fuel assembly with stainless steel clad Boral sheets. Boral has a long, proven history in worldwide nuclear service and has been used in other spent fuel storage and transportation casks. The Boral sheets are held in place with stainless steel cladding which are stitch-welded to the fuel tubes. The stitch weld design provides a vented sheath configuration to facilitate fuel retrievability. Since the TSC will be loaded dry into the UMS, no reactions between the aluminum-containing Boral sheets and spent fuel pool water are anticipated during fuel loading operations. In accordance with Section 8.1.5 of the SAR, a verification test using real-time radiography will be performed to ensure that the Boral sheets have a minimum ^{10}B loading of 0.011 grams/cm^2 ^{10}B in the Boral plates of the BWR fuel tubes and 0.025 grams/cm^2 ^{10}B in the Boral plates of the PWR fuel tubes.

The staff concludes that the material properties and characteristics of the lead, NS-4-FR and Boral that are needed to satisfy the shielding and subcriticality requirements of 10 CFR Part 71 during NCT and HAC, will be maintained during transport.

2.2.1.3 Welds

The UMS materials of construction (e.g., stainless, carbon, low alloy steels, etc.) are readily weldable using commonly available welding techniques. Circumferential and longitudinal welds are used to fabricate the inner shell and to attach it to the top and bottom forgings of the UTC. These containment vessel welds are full penetration bevel or groove welds that are radiographed and acceptance tested in accordance with ASME Section III NB-5320. To ensure integrity of the welds, the containment boundary is hydrostatically tested in accordance with ASME Code requirements, visually inspected by the dye penetrant examination method in accordance with the ASME Code and helium leak tested in accordance with ANSI N14.5-1997. The cask welds were well-characterized on the engineering drawings, and standard welding symbols and notations, in accordance with AWS Standard A2.4, "Standard Symbols for Welding, Brazing, and Nondestructive Examination," were used.

The staff concludes that the welded joints of the UMS meet the requirements of the ASME, ANSI and AWS Codes, as applicable.

2.2.1.4 Bolting Materials

The bolts that are used to fasten the UMS lid to the top forging of the cask body are fabricated from ASME SB-637, Grade N07718, nickel alloy. The ductile-to-brittle transition temperature of this steel is below the expected operating temperatures, so brittle fracture of the bolts is not expected. Procurement of the bolts in accordance with the ASME SB-637 specification will ensure that the material receives the proper heat treatment and possesses the required mechanical properties. The staff finds the bolting material acceptable. The staff also independently verified the tabulated design values and found them acceptable.

The TSC is an all-welded canister. Therefore, there are no bolts to evaluate.

2.2.1.5 Materials and Materials Selection Conclusion

The staff reviewed packaging materials of construction and their specifications and found them to be acceptable. Material specifications and properties for structural materials are consistent with the design code selected (ASME Section III). Relevant material properties for structural materials are listed in detail in SAR Section 2.3. The staff concluded that the material properties used are appropriate for the load condition (e.g., static or dynamic impact loading, hot or cold temperature, wet or dry conditions, etc.), and that appropriate temperatures at which allowable stress limits are defined are consistent with those temperatures expected in service.

2.2.2 Prevention of Chemical, Galvanic, or Other Reactions

The staff reviewed the materials and coatings of the UMS to verify that significantly adverse chemical or galvanic reaction among packaging components, among packaging contents, or between the packaging components and the packaging contents will not occur. There are no zinc, zinc compounds, or zinc-based coatings in the UMS or the TSC. The TSC will be loaded dry into the UTC, and the remaining void space in the cask cavity will be backfilled with helium gas. In this configuration, no credible chemical or

galvanic reactions are expected to occur. Therefore, there will be no corrosion of the UMS materials of construction or the generation of combustible gases during transport.

2.2.3 Effects of Radiation on Materials

The staff considered any potential damaging effects of radiation on the packaging materials, including degradation of the seals and sealing materials and degradation of the properties of the structural materials. The materials of construction of the UMS and TSC are not subject to radiation embrittlement during spent fuel transportation due to the low radiation dose and energy levels as compared to reactor vessel service.

2.2.4 Maine Yankee High Burnup fuel

With regard to transporting Maine Yankee high burn up fuel (i.e., spent fuel with burn up levels exceeding 45 Gwd/MTU) in the UMS, Section 4.5.1 of the SAR indicates that high burn up fuel assemblies that meet the oxide thickness limits of ISG-15 (or ISG-11, Revision 1) will be considered as damaged fuel and will, therefore, be placed in MYFC's. Specifically, the criteria of ISG-15 are:

A. A Maine Yankee high burn up fuel assembly may be stored as intact fuel provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise intact fuel.

B. A Maine Yankee high burn up fuel assembly not meeting the cladding oxide thickness criteria for intact fuel or that has an oxide layer that is detached or spalled from the cladding is classified as damaged fuel.

The ISG-15 acceptance criteria and requirements are specified in the loading procedures of the approved Final Safety Analysis Report (FSAR) for the NAC-UMS Universal Storage System (Docket 72-1015), which provide for the loading of the canister with high burn up fuel. At the time of loading the dual-purpose canister into the UMS transport cask (which will be conducted as a dry operation as identified in the Operating Procedures of the SAR), the criteria and requirements of ISG-15 for high burn up fuel evaluation and loading will have been satisfied.

2.2.5 Materials Conclusion

The staff concludes that the materials of construction of the UMS are acceptable for the described structural, thermal, shielding, criticality, and confinement functions.

2.3 Fabrication and Examination

Fabrication and examination specifications of the package are prescribed mainly by Section III of the ASME Boiler and Pressure Vessel Code. Specific sections, divisions, subsections, and articles of the ASME Code for fabrications and examinations of different transportation components are delineated on the engineering drawings and are listed in Section 2.1.2 of the SAR. The staff reviewed the applicant's evaluation and agreed with

the applicant’s conclusion that the package meets the regulatory requirements of 10 CFR 71.31(c).

2.4 General Standards for All Packagings (10 CFR 71.43)

2.4.1 Minimum Package Size

The transverse length of the UMS is 92.11 inches, and the longitudinal dimension is 209.25 inches. Both of these dimensions are greater that 10 cm. The staff reviewed the applicant’s evaluation and agreed with the applicant’s conclusion that the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamperproof Feature

A crimped wire seal is looped through a hole in the end flange of the lifting trunnion and through a hole in an adjacent corner of the upper impact limiter. The upper impact limiter must be removed to obtain access to the cask closure lid and bolts. An additional crimped wire seal is also placed between the lower impact limiter and the rear cask support. The presence of these seals demonstrates that unauthorized entry into the package has not occurred. The staff reviewed the applicant’s evaluation and agreed with the applicant’s conclusion that the package meets the requirements of 10 CFR 71.43(b) for a tamperproof feature.

2.4.3 Positive Closure

The large pre-load applied to the lid bolts prevents inadvertent opening of the cask closure lid from loads such as shock, vibration, thermal expansion, internal loads or external loads. The staff reviewed the applicant’s evaluation and agreed with the applicant’s conclusion that the package meets the requirements of 10 CFR 71.43(c).

2.5 Lifting and Tie-Down Standards for All Packages

2.5.1 Lifting Devices

The UMS is equipped with two primary and two secondary lifting trunnions located on the top forging near the top of the cask. The two primary trunnions, in a non-redundant lift, are capable of supporting six and ten times the cask design weight, 260,000 lbs., without producing stresses in the trunnions or cask greater than the material yield and ultimate strengths, respectively. The two secondary trunnions, which are bolted to the cask, are used in a redundant, four-trunnion, lifting operation. They are sized to support three and six times the cask design weight without stressing the trunnions and bolts beyond the respective material yield and ultimate strengths. The applicant evaluated the structural capacity of key parts of the trunnions, the cask body, and the welded and bolted trunnion-to-cask joints. The cask body is shown to be structurally more capable in resisting loads than the trunnions and the corresponding welded and bolted joints. This demonstrates that, under an excessive load, failure of the lifting devices would not impair the ability of the package to meet other requirements in accordance with 10 CFR 71.45(a).

The cask lid, which weighs 8,869 lbs., is lifted with four 1 inch-diameter swivel hoist rings, each load-rated at 10,000 lbs. and equally located on a 66.86-inch bolt circle.

Considering a cask lid bolt engagement length of 1.54 inches, the applicant calculated a safety factor greater than 3 against yield strength for the bolt hole thread shear stress, which governs the design and is acceptable. Since failure of bolt hole threads is local, under an excessive load, it will not impair the ability of the package to meet other requirements. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.45(a) requirements for an excessive load.

2.5.2 Tie-Down Devices

The tie-down devices, which are structural part of the package, consist of the two trunnion rotation pockets near the cask bottom, two primary trunnions, and two shear rings welded to the top forging of the cask. The package is assumed to be supported horizontally on a railcar, and is analyzed to resist minimum static force components at 2, 10, and 5 times the package weight along the respective vertical, horizontal axial, and horizontal transverse directions, in accordance with 10 CFR 71.45(b)(1). The SAR applied standard structural analysis techniques to calculate stresses in the rotation pocket and shear ring, including weld stresses in the ring-to-cask and pocket-to-cask joints. These stresses are less than the corresponding stress allowables and are acceptable.

The SAR compared shear capacities of the rotation pockets and shear ring welds to that of the cask body to demonstrate that the welds would fail before the cask body. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.45(b)(3) and that a tie-down device failure, under an excessive load, will not impair the ability of the package to meet other regulatory requirements.

2.6 General Considerations for Structural Evaluation of Package

The SAR evaluated the structural performance of the package by analysis and testing. For the evaluation by analysis, both the finite element analysis and closed-form solution were used. For the evaluation by testing, scale model drop tests were used to confirm the adequacy of the design of the impact limiters and to validate the finite element analysis approaches for calculating bounding cask drop deceleration g-loads. In evaluating the application, the staff reviewed initial temperatures and pressures and drop orientations for resulting in the most limiting conditions for the package. The staff also reviewed analysis assumptions to ensure that analysis methods had been used appropriately and results interpreted properly. In the following section the staff discussed the general considerations for the package with respect to evaluation by analysis and by testing, including analytical modeling approaches to calculating bounding drop test decelerations.

2.6.1 Evaluation by Analysis

2.6.1.1 Finite Element Analysis Codes

As described in Section 2.10.1 of the SAR, two general purpose commercially available finite element analysis codes ANSYS and LS-DYNA are used to perform the structural analysis of the package. The applicant used ANSYS to perform stress analysis of the UMS system components, including the UTC, TSC, fuel basket, fuel tubes, and GTCC

waste basket. However, for the package subject to free drop tests, the applicant used the explicit dynamic analysis code, LS-DYNA, to calculate impact responses, which correlate and supplement the scale model drop test results, to establish bounding rigid-body deceleration g-loads.

2.6.1.2 Finite Element Analysis Models

The SAR provides descriptions of the finite element models of key packaging components, including the cask, PWR and BWR TSCs, basket support disks, fuel tubes, and the GTCC basket. For ANSYS element selection, typically used is SOLID45 for the shell and lid/closure plates of the UMS and TSC, PLANE42 for the basket support disks, SHELL63 for the basket top and bottom weldments, SHELL43 for potential plastic deformation of the fuel tubes, BEAM4 for closure bolts preload, and CONTAC52 and COMBIN40 for gap opening and closing between individual structural entities. As loading and component configurations permitted, appropriate half-symmetry model configurations were used.

For the structural performance evaluation, the staff reviewed the finite element analyses to ensure that they are implemented appropriately and results properly interpreted. This includes considerations of temperature-dependent material properties, force and displacement boundary conditions, and load combinations of temperatures, pressures, and drop orientations resulting in the most limiting conditions. As used in the stress evaluation for individual components, the SAR defined design margin or margin of safety as the ratio of the stress allowable and the calculated stress minus one, for which the at temperature stress allowables are considered.

2.6.2 Evaluation by Test

The applicant performed 30-ft drop tests of two 1/4-scale models of the packaging to confirm that: (1) the impact limiter crush depths are limited to prevent the cask body from direct contact with the impact surface, (2) the cask rigid-body decelerations are bounded by those used in the package design analysis, and (3) the impact limiters remain attached to the cask body and in position after the drop. Additionally, the drop tests served to benchmark a finite element analysis approach from which crush depths of the impact limiters and rigid-body decelerations of the cask body can be conservatively calculated for the UMS system. Section 2.10.3 of the SAR provides details of the confirmatory testing program.

2.6.2.1 Quarter-Scale Cask Model Drop Tests

The 1/4-scale model 30-ft. drop test program consisted of three tests: top end drop and top corner drop tests at the Oak Ridge National Laboratories (ORNL) and the side drop using a different 1/4-scale model at the Sandia National Laboratories (SNL). NAC Drawings 790-300 and 790-301 provide cask specimen details and packaging assembly of the model, for the top end and top corner drop tests respectively, and Drawings 790-308 and 790-309 provide similar information for the side drop test model. The total weight of the scale model cask body and impact limiters is approximately 4,060 lbs., which corresponds to the full-scale 260,000 lbs. design limit of the UMS system. Since the impact limiters, including the attachment and interface with the cask body, are essentially a true-scale representation and the cask is properly simulated with its mass

and stiffness properties, the staff agrees with the applicant that the 1/4-scale models are appropriate for the intended tests.

Sections 2.10.3.3 and 2.10.3.4 of the SAR describe the scale model tests performed. Included in the description are the instrumentation and data reduction programs for recording cask impact response time histories, measuring impact limiter crush depths, and low-pass filtering raw data to obtain rigid-body decelerations of the cask body. The staff reviewed program implementations and finds that the test results, such as peak rigid-body decelerations and corresponding response pulse durations and shapes, are adequately reduced. The tests confirmed the capabilities of the impact limiters. They also provided needed data for benchmarking a finite element analysis approach to calculating cask impact response as discussed below.

2.6.2.2 Correlation of Tested and Calculated Results

The applicant used the LS-DYNA code to develop a finite element analysis approach to calculating drop test responses of the 1/4-scale UMS equipped with impact limiters. Section 2.10.3.7 of the SAR presents details of the finite element model which was constructed primarily of 8-node bricks and 4-node shells. Sections 2.10.1.2 of the SAR provides a description of the material models, including the material options to model the redwood and balsa wood, and the use of strain-rate dependent properties.

Recognizing the primary objective of calculating rigid-body responses of the cask body, the applicant used a single-wall representation, but with adjusted elastic modulus, to simulate the cross sectional properties of the cask body of the multi-wall construction. The applicant performed dynamic crush tests of wood specimens to obtain stress-strain curves at varied strain rates to model the redwood and balsa properties with the LS-DYNA Modified_Crushable_Foam material option. Other standard LS-DYNA options used included the Piecewise_Linear_Plasticity material type for large deformations of the steel shell and gussets of the impact limiter, Surface_to_Surface contact for interfaces between the cask body and impact limiters, and the Rigidwall_Geometric_Flat options to represent unyielding impact surface. An initial velocity of 527.4 in/sec was applied to the entire model to represent the 30-ft drop test.

Section 2.10.3.3.4 of the SAR presents two static crush tests for the end drop orientation, using a 45^{o} section of a model impact limiter. No static crush test was performed for the corner and side drop orientations. Figures 2.10.3.15 through 2.10.3.17 of the SAR present the force-deflection curves of the 1/4-scale impact limiter based on the end, corner, and side drop tests, respectively. Since these curves are characteristic of static crush tests and explicit load-deflection curves are not part of the LS-DYNA finite element modeling of the impact limiter, the staff found it acceptable that impact limiter static crush tests need not be performed as a basis for the safety evaluation.

Figures 2.10.3-8 and 2.10.3-9 of the SAR compare the calculated values with the tested rigid-body deceleration time histories, for the upper and lower accelerometer locations, respectively. The calculated response pulse durations and shapes correlate well with those obtained by the tests. Similar deceleration time history plots with more conservatively calculated results are displayed in Figure 2.10.3-12 of the SAR for the top end drop and Figure 2.10.3-13 of the SAR for the top corner drop. Since the calculated peak decelerations all bound the corresponding test results and the calculated and measured impact limiter crush depths correlate reasonably well, as shown in Tables 2.1

and 2.2 below, the staff concludes that the LS-DYNA finite element analysis approach is adequately benchmarked for application to the UMS system.

Table 2.1, Calculated and Tested Peak Decelerations - 1/4-Scale Model, 30-Ft Drop				
Drop Orientation	Drop Test (g)		Calculated (g)	
	Top	Bottom	Top	Bottom
Top Corner	121	---	143	---
Top End	220	---	226	---
Side	190	198	193	210

Table 2.2, Calculated and Measured Crush Depths - 1/4-Scale Impact Limiter, 30-Ft Drop		
Drop Orientation/Location	Measured Crush (inches)	Calculated Crush (inches)
Top Corner	2.95	3.36
Top End	2.04	2.21
Side/Under the Trunnion	2.87	2.36
Side/Bottom Impact Limiter	2.75	2.71

2.6.3 LS-DYNA Analysis of the UMS System

Following the analytical approach discussed above, the applicant constructed a finite element model of the UMS system to calculate impact limiter crush depths and cask rigid-body decelerations. Section 2.6.7.5 of the SAR presents the LS-DYNA model for the packaging. The Modified_Crushable_Foam material type was used to model the impact limiters to allow for the input of the strain rate dependent stress-strain curves. To account for crush strength fabrication tolerances, stress values of the stress-strain curves for the hot (200^{o}F) and cold (-40^{o}F) conditions were further adjusted by the factors of 0.90 and 1.10, respectively.

The applicant considered 30-ft. HAC drop orientations for which the maximum package damage is expected. Section 2.6.7.5.8 of the SAR used NAC-STC cask (Docket 71-9235) data to perform a sensitivity analysis, with the cask axis angle down to 5^{o} with respect to the horizontal surface, to evaluate effects of shallow-angle drops on package performance. The analysis established that the side drop results are bounding for the UMS, for which the ratio of the length to radius of gyration (L/r) of the package is less than 2. Figures 2.6.7.5-5 through 2.6.7.5-7 of the SAR present calculated rigid-body responses of the cask for the 30-ft drop tests for the top-end, CG.-over-corner, and side drops, respectively. Table 2.6.7.5-6 of the SAR lists the calculated impact limiter crush depths and cask rigid-body decelerations for the hot and cold temperature conditions. For the 1-ft NCT drop test, the calculated peak deceleration is 17 g, which is associated with the cold, side drop condition. As shown in Table 2.3 below, since all calculated decelerations for the cold, governing, condition bound the corresponding test equivalent values obtained from the 1/4-scale model drop tests, the LS-DYNA analysis is conservative.

Table 2.3, Calculated and Test Equivalent Peak Decelerations - Prototype UMS, 30-Ft Drop				
Rigid-Body Decelerations (g)				
Drop Orientation	Calculated Hot (200°F)	Calculated Cold (-40°F)	1/4 scale Model Test Equivalent	Design Basis
Side	49.6	52.0	49.5 (= 198 x 1/4)	60
Top/Bottom End	40.7	57.8	51.8 (= 207 x 1/4)	60
C.G.-Over-Corner	35.3	36.5	30.3 (= 121 x 1/4)	60

On the basis of the evaluation above, the staff agrees with the applicant's conclusion that the design basis decelerations of 20 g and 60 g bound the peak decelerations associated with the 1-ft NCT and 30-ft HAC free drop tests, respectively.

2.7 NCT

The SAR presents a variety of analyses to demonstrate that, under the tests specified in 10 CFR 71.71 for NCT, there would be no substantial reduction in the effectiveness of the packaging.

2.7.1 Heat

Section 2.6.1 of the SAR presents an ANSYS stress analysis of the UMS by considering thermal heat, internal pressure, bolt preload, gravity, and combined loading conditions. Table 3.4.1 of the SAR lists maximum cask component temperatures under the normal condition of transport maximum decay heat and maximum ambient temperature of 100° F. The applicant performed radial and axial thermal expansion evaluations of the TSC and the UTC and determined that they would not bind. For the stress analysis, an internal pressure of 150 psig was applied to the cask, which envelops the maximum normal operating pressure (MNOP) of 6.91 psig for the cask. The analysis shows that the combined stresses in the cask are all below the allowables. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(1) for the heat condition.

2.7.2 Cold

Section 2.6.2 of the SAR presents an ANSYS stress analysis of the UMS by considering an ambient temperature of -40° F, an internal pressure of 150 psig, no decay heat load, no solar insolation, in still air and shade. Tables 2.6.2.3-1 through 2.6.2.3-5 of the SAR document stress results with stress margins all shown positive. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(2) for the cold condition.

2.7.3 Reduced External Pressure

The drop in atmospheric pressure to 3.5 psia effectively results in an additional cask internal pressure of 11.2 psig. This pressure plus the MNOP of 6.91 psig for the cask is less than the 150 psig considered in the heat case evaluated in SER Section 2.7.1 above. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that all stress margins will be positive and that the package meets the requirements of 10 CFR 71.71(c)(3).

2.7.4 Increased External Pressure

An increased external pressure of 20 psia produces negligible effect on the UMS. The pressure will result in a hoop stress of 3,685 psi in the 0.25 inch-thick shell encasing the neutron shield. This stress is low compared to the material strength and will have no adverse effect on the performance of the cask. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(4).

2.7.5 Vibration

Section 2.6.5 of the SAR determined that resonant response of the package is insignificant, considering the periodic load effect as the two closest rail car wheels pass over a rail junction. To perform structural capability evaluations, the applicant used conservatively a 2-g cyclic load amplitude to compute stress margins, using the allowable alternating stress intensity of 23,000 psi, which corresponds to 10^{11} cycles of load applications for austenitic stainless steel. This results in the positive stress margins of 0.77 and 3.2 for the cask body and rotation pocket, respectively. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(5).

2.7.6 Water Spray

Water causes negligible corrosion of the exterior stainless steel surfaces of the package. The cask surface temperature specified during the water spray is between -20^o F and 100^o F. As a result, the induced thermal stress in the cask components is less than the thermal stresses that occur during the extreme temperature conditions for NCT. Therefore, the water spray test has no adverse effect on the package. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(6).

2.7.7 1-ft. Free Drop

The applicant performed finite element analyses and closed-form design calculations of the UMS components subject to a 20 g design basis deceleration plus other environment loading effects, such as temperature, pressure, and bolt pre-load. As discussed in SER Section 2.6.3 above, the design deceleration conservatively bounds the calculated peak deceleration of 17 g for which maximum damage to the package is expected for a 1-ft free drop test. For the c.g.-over-corner drop, axial and transverse components of 20g each were concurrently applied to the cask. As discussed below, the analyses considered stress and other structural failure modes, as appropriate, of the structural

components, including the UTC body, closure bolts, neutron shield, and TSC shell, bottom plate, lid, and basket components.

2.7.7.1 Cask Body

Section 2.10.2 of the SAR provides a description of the three-dimensional finite element analysis model, including the cask specific features, such as the lower neutron shield, gamma shield, ligaments, and bottom rings, used for structural analysis of the cask body. Also provided are load application criteria, boundary condition assumptions, and cask body section locations for post-processing stress results. Section 2.6.7.1 through 2.6.7.3 of the SAR showed the structural analysis that was performed for the cask for the design basis end-, side-, and corner-drop load of 20 g, respectively. As listed in the summary tables for the most critically stressed locations, the minimum stress margin of 0.08 for the cask corresponds to the primary membrane stress intensity at the ligament region that separates the gamma shield and the bottom ring.

Section 2.6.7.6 of the SAR presented an analysis of the cask closure assembly as per NUREG/CR6007 to demonstrate structural adequacy for the 1-ft. drop tests. The forces considered in the analysis include an installation torque of 3,900±100 ft-lb, an internal pressure on the inner lid of 80 psi, the O-ring compression force, and the inertia weight of the lid, canister, basket, and fuel during the corner drop. The 2 inch-diameter SB-637, Grade N07718 closure bolts were evaluated for two failure modes. The tensile plus shear stress analysis was evaluated using the interaction equation (i.e., the sum of the squares of the stress ratios). The calculated value of 0.33 is less than 1 and is acceptable. For the combined state of stress that includes tensile, shear, and bending against the stress intensity limit of 1.35 S_m, the stress margin is 0.61, and is acceptable.

Considering the thermal load and the preload corresponding to the maximum installation torque of 4,000 ft-lbs, Section 2.6.7.6.1 of the SAR calculated a bolt fatigue life of 944 cycles on the basis to the ASME Code Section III, Appendix I criteria.

Section 2.6.7.7 of the SAR evaluated the UMS neutron shield shell for two distributed-load conditions: a 1-ft. end drop and a 1-ft. side drop. By converting the weight of NS-4-FR into the pressure equivalent inertia load and by assuming conservative force and displacement boundary conditions, the applicant used well established formulas to perform stress analyses for various structural components. Considering the ASME Code Section III, stress limits, as appropriate, the applicant computed stress margins for the neutron shield shell, end plate, and radial heat transfer fins with acceptable results.

Section 2.6.7.8 of the SAR evaluated stress performance at the intersection of the upper ring and the outer shell of the cask being lifted at the lifting trunnions. Using a closed-form ring solution, the applicant computed conservatively the moment and torque on the equivalent ring. The resulting maximum stress intensity of about 28,300 psi corresponds to the stress margin of 0.06 (30,000/28,300 -1 = 0.06), which is positive and acceptable.

2.7.7.2 TSC

Section 2.6.12 of the SAR presented a finite element structural analysis of the bounding PWR canister class for a design basis drop static equivalent load of 20 g. The analysis covered cases with and without thermal stresses. It also considered combined effects of a canister drop with a canister internal pressure of 25 psig. Figure 2.6.12.3-1 identifies

16 canister section locations for which stresses of various categories were calculated and tabulated for individual as well as combined loading conditions of interest. As listed in the summary tables in the SAR for the most critically stressed locations, the combined internal pressure and either the side- or the corner-drop condition governs the stress performance of the canister. This includes a minimum primary membrane-plus-bending positive stress margin of 0.02, which occurs in the shield lid-to-shell weld for the pressure-plus-side drop condition. Considering the stress results associated with the 20-g end drop, the applicant also evaluated the canister buckling strength, in accordance with the ASME Code Case N-284-1 criteria, with acceptable results.

Similar to those for the PWR canister evaluations reviewed above, Section 2.6.14 of the SAR presented a structural analysis for the BWR canisters with acceptable results.

2.7.7.3 Fuel Basket

Section 2.6.13.1 of the SAR provides a description of the finite element models, including loading and boundary condition considerations, for stress analysis of the PWR fuel basket support disk: for the end drop and for the side drop. The models, which accommodate the thermal stress analysis, are conservatively subject to the 20-g design basis inertia force, after considering the dynamic load factor effect evaluated in SAR Section 2.10.4. For the side drop analysis, four support disk drop orientations, at 0^o, 18.22^o, 26.28^o, and 45^o with respect to the cask axis, were considered to ensure that the structural performance for the most damaging drop orientations was evaluated for the support disk. Considering the off-angle between the cask axis and horizontal surface, stress results of the end and side drop tests are used to compute those for the oblique drop. The analysis demonstrates that all stress margins are positive and a minimum stress margin of 0.19 is associated with the primary membrane-plus-bending stress category.

Considering the stress results associated with all end drops and off-angle drops, Section 2.6.13.14 of the SAR evaluated buckling capacities of the support disk ligaments, in accordance with the NUREG/CR-6322 criteria, with acceptable margins.

The applicant also evaluated structural performance of the other PWR fuel basket components such as tie rods, spacers, and the top and bottom weldments. All stresses, including the bearing stress at the interface between the basket and the canister shell, are demonstrated to be within the allowable limits.

Following essentially the same approaches to those for the PWR fuel basket discussed above, Section 2.6.15 of the SAR evaluated the structural performance of the BWR fuel basket. The only major exceptions for the BWR fuel basket evaluation are: (1) the oblique drop was not considered, because the g-loads produced by the end drop and side drop bound the g-loads produced by the corner drops, and (2) the support disk was analyzed for five different orientations, at 0^o, 31.82^o, 41.46^o, 77.92^o, and 90^o with respect to the cask axis. The analysis results demonstrate acceptable structural performance of the basket components.

2.7.7.4 Canister Spacers

Each canister containing Class 1 or Class 2 PWR fuel is located by one spacer to maintain its center of gravity at the required distance from the bottom inner surface of the

cask. The spacers at 18.25 inch-high and 11.25 inch-high for the Class 1 and Class 2 PWR fuel applications, respectively, are weldments made of 3/8 inch-thick stainless steel. Each spacer has a 67 inch-diameter circular base with six cylindric shells welded to it. Section 2.6.16.1 of the SAR presented a structural evaluation of the spacers for a 20-g design basis drop. The results demonstrate acceptable structural performance of the spacer for maintaining the canister in its analyzed configuration inside the cask cavity.

The circular spacer for the Class 3 and Class 4 BWR fuel applications is made of the 1.5 inch-thick ASTM B209 6061-T651 aluminum alloy plate. The Class 4 fuel application requires one spacer while Class 5 requires 4 stacked spacers with an overall height of six inches to properly position the canister within the cask. Section 2.6.16.2 of the SAR presents a structural analysis of the spacer with acceptable results.

2.7.7.5 MYFC

Section 2.11.1.1.1 of the SAR considered the 20-g design basis end- and side-drop inertia forces to perform structural analyses of the Maine Yankee fuel can. The stress margins for the fuel can components and buckling capability of the fuel can tube body are acceptable.

2.7.7.6 Greater-Than-Class-C Waste Basket

The GTCC waste canister design and the center of gravity of the transport cask containing the loaded GTCC waste canister are essentially identical to those for the PWR Class 1 fuel application. Therefore, the staff agrees that no additional evaluation is required of the GTCC waste canister.

Section 2.11.2.1 of the SAR evaluated the basket assembly in accordance with the requirements of ASME Code, Section III, Subsection NF. Figure 2.11.2.1-1 models a typical section of the basket weldment, including contact interfaces between the rib disks and the canister shell and between the canister shell and the cask inner shell. This model and two others: one for the basket support wall and one for the basket separator plate, were used to calculate stresses in the basket assembly subject to the 20-g design basis end drop and side drop loads. The resulting stresses were used to evaluate various failure modes of the basket components and showed acceptable structural performance.

The staff reviewed the applicant’s evaluation and agreed with the applicant’s conclusion that the structural performance of the UMS system satisfies the requirements of 10 CFR 71.71(c)(7).

2.7.8 Corner Drop

The corner drop test, per 10 CFR 71.71(c)(8), does not apply because the package weighs more than 50 kg (110 lbs.) and is not of fiberboard or wood construction.

2.7.9 Compression

The compression test, per 10 CFR 71.71(c)(9), does not apply because the weight of the package exceeds 5,000 kg (11,000 lbs.).

2.7.10 Penetration

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(10) because the package has no unprotected valves or rupture disks that could be affected by the NCT.

2.8 HAC

The applicant performed a variety of analyses to demonstrate that the package has adequate structural integrity to satisfy the containment, shielding, and subcriticality requirements of 10 CFR Part 71.

2.8.1 Thirty-Foot Free Drop

Except for the loading conditions and acceptance criteria, the applicant used the same structural analysis approaches and finite element models for the 30-ft. drop test as it did for the 1-ft. free drop evaluation. In the following sections, the structural performance of the UMS components, including the cask body, closure bolts, neutron shield, and the TSC shell, bottom plate, lid, and basket components are discussed.

2.8.1.1 Cask Body

Section 2.7.1.1 through 2.7.1.3 of the SAR presented a structural analysis of the cask for the design basis end-, side-, and corner-drop load of 60 g, respectively. Section 2.7.1.4 of the SAR presented the analysis of the 30-ft top- and bottom-oblique drops, assuming the cask strikes an unyielding surface at a 75^{o} angle between the cask axis and the vertical. As listed in the summary tables in the SAR for the most critically stressed locations, the minimum stress margin of 0.04 corresponds to the primary membrane-plus-bending stress at the interface between the cask outer shell and the bottom plate.

Section 2.7.1.5 of the SAR assumed that the lead gamma shielding could slump and fill the annular gap created by the cooling of the lead after fabrication. The applicant calculated a maximum gap of 3.05 inches at the top of the lead annulus during a cask end drop and a maximum radial slump of 0.91 inches during a side drop.

Section 2.7.1.7 of the SAR analyzed the cask closure assembly to demonstrate structural adequacy for two load cases: the design basis end drop of 60 g and the puncture load applied at the center of the cask lid. For the end drop, and the failure mode associated with bolt tensile plus shear stresses, the interaction equation was used (the sum of the squares of the stress ratios). The calculated value was 0.22, which is less than 1.0 and is acceptable. For the puncture load and the same interaction equation, the resulting stress ratio is 0.40. The applicant used finite element analyses to demonstrate structural performance of the cask lid and its protective flange. Since the calculated gap closures for the side drop and top end drop are less than the nominal radial gap between the lid and the flange, the closure bolts will not be subject to impact shear forces.

Section 2.6.7.7 of the SAR evaluated the UMS neutron shield shell for the 60-g design basis side drop. Considering the ASME Code Section III, stress limits, as appropriate, the applicant computed acceptable stress margins for the neutron shield shell and the radial heat transfer fins.

2.8.1.2 TSC

Section 2.7.7 of the SAR presents a finite element structural analysis of the PWR canister for a design basis 30-ft. drop test static equivalent load of 60 g. The analysis considered combined effects of canister drop with and without a canister internal pressure of 25 psig. Table 2.7.7.2-11 of the SAR summarizes the minimum stress margins for all drop orientations. This includes a minimum primary membrane-plus-bending stress margin of 0.02, which occurs in the canister shell next to the structural lid-to-shell joint during the side drop. Considering the stress results associated with the 60-g end drop, the applicant also evaluated the canister buckling strength, in accordance with the ASME Code Case N-284-1 criteria, with acceptable results.

Similar to those for the PWR canister evaluations reviewed above, Section 2.7.9 of the SAR presented an acceptable structural analysis for the BWR canisters.

2.8.1.3 Fuel Basket

Sections 2.7.8 and 2.7.10 of the SAR presented a structural analysis of the PWR and BWR fuel basket assemblies, respectively, for the 30-ft. drop tests. The applicant continued to use the same analysis approaches and finite element models as were used in the 1-ft NCT cask drop evaluation, except that the 60-g design basis drop load and the HAC structural performance criteria were considered. The resulting stress margins are all positive, which demonstrate structural adequacy of the fuel baskets.

Section 2.11.1.1 of the SAR presents a load combination parametric study of five fuel loading positions for the PWR support disk subject to the 60-g design basis side drop. The results demonstrate structural adequacy of the support disk for transporting a consolidated fuel lattice, weighing up to 2,100 lbs., held in a MYFC, and placed in one of the four corner basket positions.

Sections 2.7.8.3 and 2.7.10.3 of the SAR evaluated buckling capacities of the PWR and BWR support disk ligaments. The applicant also evaluated structural performance of other fuel basket components such as tie rods, spacers, and the top and bottom weldments. The results presented are acceptable.

Sections 2.7.8.4 and 2.7.10.4 of the SAR evaluated stresses and deformations in the PWR and BWR fuel tubes, respectively. Two load conditions were considered. The first simulates the fuel assembly load as a distributed pressure on the inside surface of the fuel tube. The second postulates that the fuel assembly grid is located at the center of the span between the support disks, which imposes a displacement constraint to produce a localized distributed load. Using a yield strength of 17.3 ksi for the Type 304 stainless steel at 750^{o} F, the applicant calculated maximum total strains of 0.11 and 0.10 for the PWR and BWR fuel tubes, respectively. These strains are far less than the material failure strain of 0.40. This ensures that the fuel tube will maintain its required structural performance during the 30-ft HAC drop test.

2.8.1.4 Canister Spacers

Section 2.6.16 of the SAR demonstrates acceptable structural performance of the PWR and BWR spacers for maintaining the canister in its analyzed configuration inside the cavity of the cask subject to a design basis end drop of 60 g.

2.8.1.5 MYFC

Section 2.11.1.1.1 of the SAR presented the analysis of the MYFC. The applicant considered the 60-g design basis end- and side-drop inertia forces to perform structural analyses of the Maine Yankee fuel can. The stress margins for the fuel can components and buckling capability of the fuel can tube body are acceptable.

2.8.1.6 Greater-Than-Class-C Waste Basket

Section 2.11.2.1 of the SAR presented an evaluation of the basket assembly in accordance with the requirements of ASME Code, Section III, Subsection NF. The applicant calculated stresses of the basket assembly subject to the 60-g design basis end drop and side drop conditions. The resulting stresses were used to evaluate various failure modes of the basket components with acceptable results.

The staff reviewed the applicant’s evaluation and agreed with the applicant’s conclusion that the package meets the requirements of 10 CFR 71.73(c)(1).

2.8.2 Crush

Because the package weighs more than 500 kg (1,100 lbs.) and an overall density is greater than water, this test, per 10 CFR 71.73(c)(2), is not applicable.

2.8.3 40-inch Puncture Test

The applicant evaluated the effects of a 40-inch free drop of the cask onto an upright 6-inch-diameter mild steel punch pin for four puncture locations: (1) cask side–midpoint, (2) center of the cask lid, (3) center of the cask bottom, and (4) cask port covers.

Using a total force corresponding to the material yielding, the applicant calculated the puncture pin reaction force equivalent to a cask deceleration force of about 5.1 g. This deceleration is bounded by those associated with the 30-ft. free drops. As a result, the effects of the 40-inch puncture test on the cask are bounded by the structural performance of the package under the 30-ft. free drops, which were discussed in Section 2.8.1 of the SER.

The localized puncture effects were evaluated using a general approach of applying a uniformly distributed pressure of 47,000 psi, the material dynamic yield stress over a 6-inch-diameter region at the puncture pin location, for the stress analyses. Sections 2.7.2.1 through 2.7.2.3 of the SAR presented an ANSYS finite element analysis of the cask side-midpoint, lid center, and bottom plate center, respectively, with acceptable stress margins. Using a simplified approach to calculate shear stresses at the punch cross sections, the applicant also demonstrated with an alternative method that puncture at those locations would not occur.

The valves for the vent and drain ports are recessed within the 2-inch diameter openings in the cask lid and bottom forging, respectively. Although they are inaccessible to the 6-inch-diameter puncture pin during a 40-inch drop, the applicant performed a closed-form design analysis of a standard port cover and demonstrated an acceptable stress margin.

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(3).

2.8.4 Thermal

Tables 2.7.3.1-1 and 2.7.3.1-2 of the SAR summarize the maximum canister and cask cavity pressures, respectively, for the cask assumed to be subject to a 1,475^{o} F thermal test for 30 minutes. The calculated maximum canister internal pressure of 74.3 psig and cask cavity internal pressure of 69.3 psig are enveloped by the design basis pressures of 80 psig and 150 psig, respectively for structural evaluation of the UMS system.

2.8.5 Immersion - Fissile Material

The package is subject to a head of water of 0.9 m, which is equivalent to an external pressure of 1.3 psig. This pressure is negligibly small compared to the external pressure of 290 psig for which the package is determined to be acceptable in SER Section 2.9. The applicant assumed water in leakage for the criticality evaluation. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(5).

2.8.6 Immersion - All Packages

The immersion test requirements are met because the effect of an external pressure of 21.7 psig caused by immersion under 50 feet of water is of negligible consequence, compared to that associated with an external pressure of 290 psig as evaluated in SER Section 2.9 below. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(6).

2.9 Special Requirements for Irradiated Nuclear Fuel Shipments

Section 2.7.6 of the SAR evaluated the package to withstand an external pressure of 290 psig by performing closed-form analyses of cask components, such as the outer shell, bottom plate, lid, and port cover plate. The results demonstrate that the containment boundary is capable of withstanding the pressure without collapse, buckling, or water in leakage, in meeting the deep immersion test. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.61.

2.10 Internal Pressure Test

Section 8.1.2.3 of the SAR states that the cask containment boundary is hydrostatically pressure tested to 125% of the design pressure in accordance with ASME Code, Section III, Paragraph NB-6220. Since the cask containment is tested to 85 psig internal pressure, which is greater than 150% of the maximum normal operating pressure (MNOP) of 6.9 psig, the internal pressure test requirements are satisfied. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.85(b).

2.11 Fuel Rods

Section 2.9 of the SAR analyzed buckling capacities of a few representative classes of PWR and BWR fuel assemblies. The analysis followed the buckling evaluation approach for the approved NAC-STC (Docket 71-9235) in that the frequency difference between the axial and lateral modes of vibration were considered, as appropriate, in calculating the effective, static equivalent axial g-load in a fuel rod. The applicant also included pellets weight and derated material properties of irradiated fuel in calculating clad section properties, per NRC Interim Staff Guidance (ISG) 11, Rev. 1. The results show that the representative fuel rods will not buckle when subjected to the 60g design basis end-drop load. This satisfies Section 2.9 of Regulatory Guide 7.9, which states that analysis or test results be provided to show that the cladding will maintain sufficient mechanical integrity to provide the degree of containment claimed.

2.12 Review Findings

The staff reviewed the statements and representations in the application by considering the regulations, appropriate Regulatory Guides, applicable codes and standards, and acceptable engineering practices. The staff concludes that the structural design has been adequately described and evaluated and the structural performance of the package meets the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The objective of this review is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of the Thermal Design

3.1.1 Thermal Packaging Design Features

The thermal design features of the package include the following:

A. Helium backfill gas for heat conduction in the transport cask and in the canister and also to provide an inert atmosphere to prevent fuel cladding oxidation and degradation;

B. Radial disks, called heat transfer disks and structural disks, are utilized to dissipate heat from the fuel assemblies to the canister shell. These disks are stacked axially averaging less than 3 inches center to center spacing. The heat transfer disks are made of aluminum and the structural disks are either made of stainless steel for the PWR baskets or carbon steel for the BWR baskets;

C. Contact only between the canister inner shell and the structural/heat transfer disks, and; between the canister outer shell and the transportation cask inner wall, to conservatively account for transportation in the horizontal orientation since contact between the fuel assemblies and the

fuel tubes and contact between the fuel tube and structural disks has been conservatively omitted; and

D. 24 longitudinal fins comprised of explosively bonded copper and stainless steel, connect the outer stainless steel shell of the transportation cask to the neutron shield side plates, to increase the radial heat dissipation through the neutron shield material (i.e., NS-4-FR).

3.1.2 Codes and Standard

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code or ASTM Standards are referenced by the applicant.

3.1.3 Content Heat Load Specification

The applicant analyzed the transportation cask for PWR and BWR fuel assembly types. The package was analyzed based on a maximum decay heat of 0.83 kW per fuel assembly for up to 24 PWR fuel assemblies and 0.29 kW per fuel assembly for up to 56 BWR fuel assemblies. This corresponds to a maximum decay heat of 20 kW for the PWR fuel or 16 kW for the BWR fuel. For PWR fuel, the total package heat load is further limited to values less than 20kW (i.e. to a lower limit of 16.5 kW) depending upon cooling time and burn-up, to maintain the maximum cladding temperatures below acceptable limits. This was not necessary for the BWR fuel since the maximum clad temperature at 16 kW was lower than the lowest allowable BWR cladding temperature limit (refer to SAR Table 3.4-16). The maximum assembly average burn up is 45,000 MWD/MTU for general spent fuel contents and 50,000 MWD/MTU for Maine Yankee site-specific fuel contents. For general spent fuel contents, the minimum cool times vary with enrichment, burn up, and fuel type and are shown in Tables 1.2-6 and 1.2-7 of the SAR. For site-specific fuel contents the minimum cooling time varies with enrichment, burn up, and content as shown in Tables 1.3.1-2, 1.3.1-3, 1.3.1-4, 1.3.1-5 and 1.3.1-6 of the SAR. Site-specific contents, in addition to spent PWR fuel, include: control element assemblies, in-core instrumentation thimbles, activated stainless steel replacement rods, and damaged fuel. Also, for the site specific fuel contents, GTCC waste is included but is bounded by the PWR thermal analysis and is not thermally evaluated.

In order for a package to be transported, it needs to meet all three of the following thermal criteria:

(1) Each assembly must have a heat load rating equal to or less than 0.83 kW for PWR fuel or 0.29 kW for BWR fuel,

(2) The cooling time limits must be met. Tables 1.2-6 and 1.2-7 of the SAR are applicable for general spent fuel content; or, Tables 1.3.1-2, 1.3.1-3, 1.3.1-4, 1.3.1-5 and 1.3.1-6 of the SAR are applicable for site-specific fuel content, and,

(3) The total package heat load must be in conformance with SAR Table 3.4-16.

The applicant used the SAS2H/ORIGEN-S code to determine the assembly decay heat load using burn-up, enrichment, and cooling time of the fuel. The staff verified the various fuel cooling times in the aforementioned tables by calculating the associated

assembly heat load using ORIGEN ARP and determined that the method the applicant used to determine heat load was adequate.

3.1.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, contained in Tables 3.4-1, 3.5-1 and 3.5-2 of the SAR for NCT and HAC, were verified to include the important components such as containment vessel, seals, gamma and neutron shielding, and fuel cladding. The temperatures were consistent with those presented throughout the SAR for both the NCT and HAC. The staff also confirmed that the summary tables contained the design temperature limits for each of the critical components of the package. For HAC, the applicant accounted for the pre-fire, fire, and post-fire component temperatures. With the exception of the impact limiters and the neutron shield, which are not critical components needed to function after the hypothetical accident condition fire test, all components remain below their material property temperature limits.

3.1.5 Summary Tables of Pressures in the Containment System

The pressure calculation for the containment system under the NCT and HAC were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Thermal Evaluation sections of the SAR. The Maximum Normal Operating Pressure (MNOP) was reported along with the hypothetical accident condition pressure and the hydrostatic test pressure, and follows the approach in the NRC Standard Review Plan (NUREG-1617).

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The SAR provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. Conservative thermal emissivities were used to model the radiative heat transfer to and away from the package. The thermal properties used for the analysis were appropriate for the materials specified. Additionally, the fluid properties of the surrounding air were provided in the evaluation of thermal convection parameters. The staff questioned the applicants' use of a lower conductivity value of helium than was used in the storage SAR, and was satisfied that it is conservative for NCT. For the HAC, the basket was not explicitly modeled but its internal temperatures were derived by adding the difference between the transportation cask inside surface temperature during HAC and the canister outside wall at NCT to the internal basket component temperatures for NCT. This is conservative since the NCT temperatures were determined by use of the lower helium conductivity which would result in slightly higher temperatures. Overall, these properties were appropriate for the conditions of the cask required by 10 CFR Part 71.

3.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant. All components were shown to satisfactorily perform under NCT with an ambient temperature of -40^{0}F.

3.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality was specified.

For the maximum allowable cladding temperature the applicant utilized values ranging from 322 ^{o}C to 389 ^{o}C, as shown in Table 3.4-15 of the SAR, depending on cooling time and burn-up. These maximum allowable cladding temperatures were based on the methodologies of PNL-6189 and PNL-6364 which the staff has previously accepted. However, during the review of this application, the staff issued ISG-11, Revision 2, entitled "Cladding Considerations for the Transportation and Storage of Spent Fuel" on July 30, 2002. This ISG changed the cladding temperature limit to 400 ^{o}C (752 ^{o}F) for normal and off-normal (applicable to storage only) conditions, which is less restrictive than the previously issued guidance contained in PNL-6189. Also, ISG-11, Revision 2 changed the cladding temperature limit to 400 ^{o}C for "short term operations including cask drying" but that criteria was subsequently amended via a staff analysis for low burn up fuel (i.e., < 45 Gwd/MTU). The results of the staff analysis were presented by NRC staff in a public meeting on September 6, 2002, to allow loading of spent fuel with burn-ups less than 45 Gwd/MTU and hoop stresses less than 90 MPa up to the corresponding temperature, possibly as high as 570 ^{o}C (1058 ^{o}F) for short term conditions. For high burn-up fuel (> 45 Gwd/MTU), the cladding temperature limit was established at 400 ^{o}C and thermal cycling limited to temperature differences less than 65 ^{o}C. The cladding temperature limit for accident conditions remains at 570 ^{o}C.

The new staff guidance (ISG-11, Revision 2) raises the cladding temperature limit to 400^{0}C for NCT. Therefore, the cladding temperature limits utilized for this application are conservative and acceptable.

3.3 Thermal Evaluation under NCT

3.3.1 Heat

The UMS transportation cask is designed to transport either one of three classes of PWR fuel or one of two classes of BWR fuel. To minimize the number of analyses the applicant performed one bounding analysis for the PWR fuel and one for the BWR fuel. In general, the bounding cases are composed of the shortest fuel assemblies with the lowest effective thermal conductivity, the shortest baskets and shortest canisters. The shortest fuel basket contains the least number of support disks and heat transfer disks and the longest space in the bottom of the cask, resulting in a higher heat concentration and reduced heat dissipation, consequently maximizing calculated temperatures. Typically, solar insolation, natural convection and thermal radiation boundary conditions are applied to the outside surface of the transportation cask (except for the area of the cask that is covered by the impact limiters which are modeled as adiabatic). To determine the maximum temperatures in the cask, a steady-state analysis was performed utilizing an ambient temperature of 100^{0}F with solar insolation and maximum decay heat.

For each PWR or BWR fuel type the applicant performed the thermal analyses using three models and the ANSYS computer code, described as follows:

A) A three-dimensional half-symmetry finite element model is used to represent the loaded package (refer to SAR Figure 3.4-1 for a diagram of the PWR finite element model). The model includes the basket with fuel tubes and fuel assembly, canister, spacer (located between canister bottom and cask shell bottom forging), cask body, and helium gas between components. Since the cask is analyzed in its shipping horizontal position, contact was only simulated between: the cask inner shell and the outside surface of the canister, and; between the fuel basket and the inside surface of the canister. A two-degree contact surface region is assumed for all contact load-bearing surfaces (only the heat transfer disks assume a line contact). Contact between the fuel assemblies and fuel tube, and contact between the fuel tube and structural disks was conservatively ignored. Effective conductivities are used in this model to account for the fuel region inside the tubes; the fuel tube including the BORAL plates and gaps on both sides; and the neutron shield material with internal copper/steel fins. A volumetric heat generation rate is applied to the active fuel region on the basis of the basket's rated power (i.e., 20 kW for PWR or 16 kW for BWR) and an axial power distribution in accordance with Figures 3.4-2 and 3.4-6 of the SAR for PWR and BWR fuel, respectively. This model is utilized to determine the maximum temperature for the basket, canister, cask shells, radial shielding, and surface conditions under NCT.

B) A two-dimensional quarter-symmetry finite element model is used to determine the fuel assembly with the lowest effective thermal conductivity (refer to SAR Figure 3.4-3 for a diagram of the PWR finite element model). The model includes the fuel pellets, cladding, helium gas between the fuel rods, and helium gas occupying the gap between the fuel pellets and cladding. A volumetric heat generation rate that is representative of the heat load and peaking factor associated with the type of fuel, is applied to the fuel pellets. The BWR model developed by the applicant is similar.

C) A two-dimensional full-size finite element model is used to determine the effective conductivity of the fuel tube composite wall, as shown in SAR Figure 3.4-4 for the PWR fuel. The model starts from the inside surface of the fuel tube where a heat flux is applied that is representative of the heat load and peaking factor associated with the type of fuel. The model includes the BORAL plate with its aluminum cladding, gaps on both sides of the aluminum cladding prior to the stainless steel fuel tube, and ends with a gap before the surface of the support/heat transfer disk. The BWR model is similar except that BORAL is not used on all sides of the fuel tube.

For NCT, the applicant performed a steady-state evaluation of the entire cask. This analysis produced a maximum cladding temperature of 673°F for PWR fuel which remains below the limit of 752°F (400°C) as established by ISG-11, Rev.2. The maximum seal temperature under normal conditions is 266°F, which is substantially below the limit of 300°F. The neutron shield material was close to its allowable temperature (within a few degrees) and as a consequence, the staff requested that an additional sensitivity study be performed to demonstrate that its limit was not exceeded,

and is discussed below in Section 3.3.6, Effects of Uncertainties. The PWR fuel analysis bounds the results of the BWR fuel analysis. The PWR analysis also bounds the Maine Yankee site specific contents including GTCC.

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(1).

3.3.2 Cold

With no decay heat and an ambient temperature of -40^o F, the entire package will maintain a steady-state temperature of -40^o F. Cask components, including the containment system seals, would not be adversely affected by this temperature.

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(2).

3.3.3 Maximum Normal Operating Pressure (MNOP)

The applicant calculated the maximum normal cask and canister pressures by assuming that 3% of the fuel rods fail during NCT and that 30% of the gaseous fission products are available for release. In addition, gases from burnable poison rod assemblies and initial backfill of rods and cask/canister were also included. The gaseous fission products were calculated based on a maximum fuel burn-up of 60 GWD/MTU. As a result, the MNOP was calculated to be 6.91 psig. The staff also evaluated the MNOP assuming that 100% of the fuel rods fail and that 30% of the gaseous fission products are available for release under NCT. Based on these assumptions, the staff evaluated an MNOP of 49 psig. In Section 8.1.2.3 of the SAR, the applicant defines the MNOP of the transportation cask to be 7.3 psig and the hydrostatic test pressure to be 85 psig (i.e., 1.25 times the design pressure per ASME Code). The staff reviewed the applicant's calculations and agreed with their conclusion that transportation requirements have been met.

3.3.4 Evaluation of Accessible Surface Temperature

Under NCT, the package is enclosed by a protective screen (a personnel barrier) to ensure that the accessible surface remains well below a temperature of 185 ^{0}F. No solar insolation was applied to the package in making this determination**.** Therefore, the package must be shipped by exclusive use within a closed conveyance.

The staff reviewed the applicant's evaluation which demonstrates that the maximum accessible surface temperature is less than approximately 150 ^{0}F and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(g) for exclusive use shipments.

3.3.5 Maximum Thermal Stresses

The temperatures calculated by the applicant for NCT are consistent with those used in structural Sections 2.6.1 and 2.6.2 of the SAR in their evaluation of thermal stresses.

3.3.6 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and in analytical methods. For the three-dimensional cask model the applicant explicitly addressed the accuracy of the model by identifying the inherent conservatism and by performing a sensitivity study. The conservatism in the model included ignoring the contact of the fuel assemblies with the fuel tube, ignoring the contact of the fuel tube with the support disks, ignoring the contact of the lead with the cask inner shell and ignoring internal convection heat transfer. The sensitivity study evaluated the following eight effects:

A. 10% reduction of emissivity of stainless steel and aluminum
B. 10% reduction of heat transfer coefficient at cask outer surface
C. Reduced contact area from 2 to 1% between the disks and canister shell
D. 8% reduction of heat transfer disk thickness based on plate thickness tolerance
E. Increased gap between disks and canister shell based on tolerances
F. Increased gap between canister shell and cask inner shell based on tolerances
G. 6% reduction of the neutron shield copper fin thickness based on tolerances
I. 10% reduction in lead emissivity

The results of the sensitivity indicate about a 25 ^{o}F increase in the maximum temperatures for fuel cladding, support disks, and heat transfer disks assuming that all eight effects occur simultaneously. Also applying the applicable effects to the remainder of the components (except for the neutron shield), all components remain below their allowable temperatures.

A separate sensitivity was performed for the neutron shield since the evaluation discussed above did not maximize its temperature. The neutron shield sensitivity study included the following effects:

A. 10% increase of emissivity of stainless steel and aluminum
B. No change to heat transfer coefficient at cask outer surface
C. Increase contact area from 2 to 3% between the disks and canister shell
D. 8% increase in heat transfer disk thickness based on plate thickness tolerance
E. Decrease gap between disks and canister shell based on tolerances
F. Decrease gap between canister shell and cask inner shell based on tolerances
G. 6% reduction of the neutron shield copper fin thickness based on tolerances
H. 10% increase in lead emissivity
I. Contact was modeled between the fuel assembly and fuel tube, and between the fuel tube and structural disks.

No reduction was made to the heat transfer coefficient effect for this neutron shield case since the applicant presented test data from the NAC-LWT (71-9225) cask, that the staff reviewed, which demonstrated that the free convection correlation utilized for this application is conservative. The results from this study indicate that the maximum neutron shield temperature is 293 ^{O}F, as designed.

3.3.7 Summary

A summary of calculated temperatures versus allowables for NCT can be found in Table 3.1. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71.

3.4 Thermal Evaluation under HAC

3.4.1 30-Minute Thermal Test

The applicant performed a 30-minute transient thermal analysis using the ANSYS computer code to evaluate the package under HAC. A two-dimensional axis-symmetric finite element ANSYS model was utilized to analyze the postulated regulatory 30-minute fire at 1475 ^{o}F.

The model, as shown in Figures 3.5.1 through 3.5.3 in the SAR, comprises three concentric shells representing an inner cask (ss), a gamma shield (lead), and the outer cask (ss). The outer shell also includes the neutron shield and its stainless steel cover. A heat flux representing the decay heat load with axial distribution is applied as a boundary condition on the inside surface of the inner cask shell. Portions of the lead shield which extend above and below the neutron shield are protected from external heating via a thin layer of firebrick. All heat transfer from the inner surface of the cask to the neutron shied shell is modeled as conduction. No gaps are modeled between the different material layers which conservatively increases the heat input from the fire. The effective conductivity of the neutron shield material with its internal copper/stainless steel fins is modeled the same as was done for the NCT. The analysis is conducted in three phases:

A. Steady state initial condition with maximum decay heat, solar insolation, and an ambient temperature of 100^{o}F,

B. Transient 30 minute fire with maximum decay heat, no solar insolation, and a fire temperature of 1475^{o}F (including convection and radiation), and

C. Transient post fire cool-down with maximum decay heat, solar insolation, and an ambient temperature of 100^{o}F.

At the end of the fire the neutron shield material is replaced with air and a new effective thermal conductivity is determined for the air and copper/stainless steel fins. This is done to compensate for the neutron shield material being destroyed during the fire (a temperature limit of only 300^{o}F and temperatures in the area being in excess of 1000^{0}F) such that heat input is maximized during the fire and heat is conservatively restricted from leaving after the fire. During all phases of the fire the impact limiters are modeled as an adiabatic surface, because they have been demonstrated to remain connected to the package after the 30-ft. drop and 40-inch puncture tests. The wood in the impact limiter is presumed not to combust because its stainless steel shell restricts oxygen from supporting combustion which is a position the staff has previously supported. The emissivity of the stainless steel outer shell of the neutron shield material was raised to 0.9 during the fire, but during post fire conditions was lowered to its original pre-fire value of 0.36, which conservatively restricts heat from leaving the package after the fire.

Many of the peak temperatures are not realized until the package reaches post-fire steady-state conditions. Therefore, some of the HAC temperatures shown in the table below reflect the peak temperature of a specified component in the period following the 30-minute thermal test. The post-fire transient was evaluated for a period of approximately 18 hours to observe the cooling of the package to post-fire steady-state temperatures. The peak basket component temperatures and peak cladding temperatures for the fire were determined by adding the difference between the cask inner shell temperatures from NCT and HAC, to the NCT temperature from the component of interest.

For HAC, the analyses produced a maximum cladding temperature of 808°F, which is below the limit of 1058°F. Under these conditions, the maximum O-ring temperature was shown to be 320°F which is below its maximum temperature limit of 375°F. It was noted that the heat transfer and structural disks exceeded their HAC allowable temperatures, but were determined to be acceptable because their intended safety functions of removing heat and maintaining fuel configuration, respectively, were not compromised.

3.4.2 Maximum Pressure

The applicant calculated maximum pressure under HAC as 74 psig, based on the average cavity gas temperature of 588°F. This pressure is well below the NCT 85 psig hydro test pressure of the package. The analysis for HAC assumes failure of all rods and the canister.

3.4.3 Maximum Thermal Stresses

The staff agrees with the applicant's position that thermal stresses are not required by the ASME Code to be evaluated as part of HAC (Level D service limits). Staff has made a recommendation to change NUREG-1617 "SRP for Transportation Packages for Spent Fuel", to delete Section 3.5.6.4 which suggests that this be done.

3.4.4 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and in analytical methods. Because of the significant safety margins between the temperatures for the cladding, seals, lead shielding, cask and canister shells, and their associated allowable temperature, staff felt that a specific sensitivity study was not needed. It should be noted that the neutron shield was assumed to fail during the fire and was replaced with air in the thermal model after the fire duration to maximize post-fire component temperatures. Also, the heat transfer and structural disks exceeded their accident allowable temperatures, but are able to perform their intended safety functions of removing heat and maintaining fuel configuration. Therefore, the staff feels that the thermal model for the UMS transportation package reasonably evaluates the consequences of the HAC thermal test.

3.4.5 Misloading

The NRC has determined that the overall risk associated with a misloading event does not warrant special consideration within the thermal analysis arena because there is no adverse safety impact and the low likelihood of it happening. A misloading, in a worst

case scenario, would cause some rods to overheat and possibly result in fuel cladding rupture. However, the fuel would remain in its analyzed configuration since the rupture openings would be small for the ductile cladding and the containment boundary of the cask and canister would remain intact. Possible thermal damage to the neutron shield could occur but it would likely be detected prior to transport and corrective action taken. Since risk is defined as frequency times consequence, the overall risk associated with a misloading event would be acceptable since there are no adverse consequences even though some misloading events have occurred and could possibly occur again.

3.4.6 Summary

A table of calculated temperatures versus allowables can be found in Table 3.1. The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(4).

3.5 Site Specific Contents

The section addresses the Maine Yankee site specific fuel which is limited to the PWR design basis heat load for transportation of 20 kW per cask and 0.83 kW per assembly and utilizes a standard 14 X 14 fuel assembly. The applicant specifically evaluated the following Maine Yankee contents:

A. Two consolidated fuel rod latices, each 17 X 17. One contains 283 fuel rods with 2 vacancies and the other contains 172 fuel rods with the remaining locations either empty or containing stainless steel dummy rods;

B. Standard fuel assemblies with a control element assembly (CEA) inserted in each one;

C. Standard fuel assemblies that have had damaged fuel rods removed and replaced with stainless steel dummy rods, solid zirconium rods, or 1.95% enriched fuel rods;

D. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes;

E. Standard fuel assemblies that have in-core instrument thimble assemblies stored in the center guide tube;

F. Standard fuel assemblies that have variable radial enrichment and axial blankets;

G. Standard fuel assemblies that have fuel rods removed; and

H. Fuel assemblies with damaged fuel tubes.

The applicant analyzed all of the above configurations for variances in heat load and/or effective thermal conductivity, and for the damaged fuel assumed a 50% compaction of the damaged fuel assembly located at the center of the active fuel region. The results for the first seven cases indicate that they are bounded by the thermal analysis of the standard fuel assemblies. For damaged fuel the applicant demonstrated, to the

satisfaction of the staff, that the resulting impact on the component temperatures remained below their allowables.

3.6 Evaluation Findings

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

Table 3.1, Summary of Maximum Temperatures For The UMS[1]

Component	NCT[2]		HAC[3]		
	Maximum Temperature,°F	Allowable,°F	Maximum Temperature, °F	Time, hours	Allowable, °F
Fuel Cladding	673	752[4]	808	NA	1058
Cask O-Rings	266	300	320	0.8	375
Radial Neutron Shield	293	300	Limit Exceeded- Not modeled post-fire	NA	NA
Lead Gamma Shield	306	600	473	2.9	600
Aluminum Heat Transfer Disk	605	700	739	NA	Note 5
Stainless Steel Support Disk	608	650	743	NA	Note 5
Cask Inner Shell	344	650	479	NA	650
Canister Shell	408	650	Not Specified[6]	NA	650
Canister Gas	453	NA	588	NA	NA

Notes:
1. Only PWR values are given since they generally are bounding.
2. Data taken from SAR Tables 3.4-1 & 3.4-3 for NCT.
3. Data taken from SAR Table 3.5-1 for HAC.
4. Cladding temperature limit from ISG-11, Rev.2.
5. These components remain well below their respective melting points during the fire and are not load bearing at this time after the accident.
6. Even though this value was not specified in SAR it is bounded by the cask inner shell temperature and the canister gas temperature and is well below it's allowable by approximately 100 °F.

4.0 CONTAINMENT

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC.

4.1 Description of the Containment System

4.1.1 Containment Boundary

The containment boundary of the UMS consists of the following components: (1) the inner shell, (2) the bottom forging, (3) the top forging, (4) the cask lid, (5) the lid inner o-ring, (6) the vent port cover plate, (7) the vent port cover plate inner o-ring, (8) the drain port cover plate, and (9) the drain port cover plate inner o-ring. Table 4-1 lists all containment boundary components and their materials of construction.

Table 4-1: NAC-UMS Universal Transport Cask Containment System Components

COMPONENT	MATERIAL	Drawing No.
Inner Shell	ASME SA-240, Type 304	790-502, Rev. 4
Bottom Forging	ASME SA-336, Type 304	790-502, Rev. 4
Top Forging	ASME SA-336, Type 304	790-502, Rev. 4
Cask Lid	ASME SA-336, Type 304	790-503, Rev. 1
Cask Lid Inner O-Ring	Ethylene Propylene Rubber	790-503, Rev. 1
Vent Port Cover plate	ASME SA-240 or SA-479, Type 304	790-504, Rev. 1
Vent Port Cover plate Inner O-Ring	Ethylene Propylene Rubber	790-504, Rev. 1
Drain Port Cover plate	ASME SA-240 or SA-479, Type 304	790-504, Rev. 1
Drain Port Cover plate Inner O-Ring	Ethylene Propylene Rubber	790-504, Rev. 1

All containment seals are ethylene propylene rubber (EPDM) seals. The top closure, drain and vent port cover plates are equipped with dual O-ring seals. The inner O-rings are the containment seals for the UTC. The EPDM seals are generally rated for service at temperatures between -65 and 300°F and are chemically compatible with their stainless steel mating surfaces. The temperatures that these seals experience are well below 300°F. Further, EPDM has good resistance to degradation in steam and water, and is adequately resistant to the radiation levels that the seals are likely to encounter in the cask lid and cover plate regions. All containment seals are leak tested in accordance with ANSI N14.5-1997.

The cask lid is fastened to the cask body with 48 2-inch diameter ASME SB-637 nickel alloy bolts. The vent and drain port cover plates are each fastened to the cask lid with 4

1/2-inch diameter ASME SA-193, Grade B6, Type 410 stainless steel bolts. Bolt torques for these bolts are specified in Table 7-1 of the SAR.

4.1.2 Codes and Standards

All containment welds of the UTC are full-penetration bevel or groove welds and are radiographically inspected in accordance with ASME Code Section III, Division 1, Subsection NB-5320.

The staff has reviewed the description of the containment system, as shown in Chapters 1 and 4 of the SAR. The staff findings include: (1) the SAR described the containment system in sufficient detail to provide an adequate basis for its evaluation; (2) the SAR identifies established codes and standards for the containment system; (3) the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package; and (4) the containment system is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction.

4.1.3 Special Requirements for Damaged Spent Nuclear Fuel

The applicant has designated the TSC as the separate inner container (i.e., secondary containment system) to meet the requirements of 10 CFR 71.63 when shipping Maine Yankee damaged fuel and fuel debris. The secondary containment boundary includes the TSC shell, bottom plate, shield lid, vent and drain port covers and their associated welds, and the structural lid.

Since the TSC is completely welded closed and leak tight in accordance with the requirements of ANSI N14.5-1997, it will adequately contain the Maine Yankee damaged PWR fuel with associated debris for the following reasons. The applicant demonstrated that the TSC maintains its structural integrity during NCT and HAC as indicated in SAR Chapter 2. Further, the peak containment boundary component temperatures and pressures are within the design-basis limits, as indicated in SAR Chapter 3. Additionally, the integrity of the TSC closure welds are assured through; (1) nondestructive examinations; (2) leakage testing; and (3) pneumatic testing. The staff agrees with the applicants conclusion that the TSC is an acceptable separate inner container for transportation of Maine Yankee damaged fuel and fuel debris and met the requirements of 10 CFR 71.63(b).

4.2 Containment Under NCT

4.2.1 Pressurization of Containment Vessel

Within SAR Chapter 3, the applicant demonstrated, and the staff confirmed, that the maximum normal operating pressure (MNOP) does not exceed the package design pressure. The applicant used a fuel rod failure of 3% in their MNOP calculation.

4.2.2 Containment Criteria

The applicant calculated the releaseable radiological source term and the maximum allowable leakage rate. The applicant performed a detailed containment analysis to demonstrate that the calculated leakage rates of the UMS, based on 10 CFR 71.51

criteria, will not be exceeded during NCT and HAC. The applicants' analysis was performed in accordance with NUREG/CR-6487, ISG-15, and ANSI N14.5-1997.

The contents of the UMS include:

A. Up to 24 PWR spent fuel assemblies with burn up, initial enrichment and cooling time combinations described in SAR Chapter 1;

B. Up to 56 BWR spent fuel assemblies with burn up, initial enrichment and cooling time combinations described in SAR Chapter 1;

C. Maine Yankee Greater Than Class C (GTCC) waste described in SAR Chapter 1; and

D. Maine Yankee intact fuel, damaged fuel, and fuel debris within a TSC with burn up, initial enrichment and cooling time combinations described in SAR Chapter 1.

The staff reviewed the analytic assumptions used by the applicant for its containment analyses. The source terms developed for the containment analysis were conservative. These assumptions were consistent with the guidance provided in NUREG/CR-6487 and ANSI N14.5-1997. Table 4-2 summarizes the assumptions used in the analysis.

Table 4-2, Containment Analysis Assumptions

Assumptions	Normal	Accident
Crud Spallation Factor, f_c	0.15	1.0
Crud surface activity, S_c (Ci/cm^2) (Assumed to be Cobalt-60)	PWR- 140×10^{-6} BWR- 1254×10^{-6}	PWR- 140×10^{-6} BWR- 1254×10^{-6}
Fraction of rods that develop cladding breach, f_B	0.03*	1.0
Fraction of fines released, f_f	3×10^{-5}	3×10^{-5}
Fraction of gases released, f_g	0.3	0.3
Fraction of volatiles released, f_v	2×10^{-4}	2×10^{-4}

*Based on ISG-15, the applicant used a 0.20 value to evaluate the failure of high burn up fuel rods during NCT.

The applicant identified the releaseable source terms, which comprises fission product gases, volatiles, fines and crud, and is presented in SAR Section 4.2 and Table 4.2-2.

In the containment analysis, the applicant calculated the effective A_2 of the releaseable source term using the relative release fraction for each constituent (as shown above in Table 4-2). As presented in SAR Section 4.2, the maximum permissible release rate and the maximum permissible leakage rate were calculated in accordance with the guidance of NUREG-6487 and ANSI N14.5-1997. The applicant, then, converted the maximum allowable leakage rate under normal transport conditions to a reference air leakage rate under standard leakage test conditions as presented in SAR Table 4.2-4.

To summarize, the following leakage rates and test sensitivities, as indicated in SAR Chapter 4 were utilized:

	Primary Containment Boundary (NAC-UMS Transport Cask)	**Secondary Containment Boundary (TSC)**
Intact Fuel (56 BWR, 24 PWR, or Intact Maine Yankee Fuel), and Maine Yankee GTCC Waste		
"As Tested" Leakage Test Sensitivity	$\leq$ 4.2 x 10^{-6} ref-cm^3/sec $\leq$ 2.1 x 10^{-6} ref-cm^3/sec	not required not required
Maine Yankee Damaged Fuel or Fuel Debris		
"As Tested" Leakage Test Sensitivity	$\leq$ 4.2 x 10^{-6} ref-cm^3/sec $\leq$ 2.1 x 10^{-6} ref-cm^3/sec	$\leq$ 1 x 10^{-7} ref-cm^3/sec $\leq$ 5 x 10^{-8} ref-cm^3/sec

In addition, in accordance with ANSI N14.5, fabrication verification, periodic verification, and assembly verification leak tests will be performed to verify the containment capability of the package. The staff reviewed the applicant's analyses and performed confirmatory calculations and calculated similar reference air leakage rates as that of the applicant.

The staff reviewed the applicant's containment evaluation and performed independent confirmatory analysis. The staff's results were in agreement with the applicant's results.

4.2.3 Compliance with Containment Criteria

Results of the applicant's structural and thermal analyses (SAR Chapters 2 and 3, respectively) show that, under NCT, the integrity of the containment boundary is not compromised when subjected to the conditions and tests specified in 10 CFR 71.71. Therefore, the staff agrees with the applicants conclusion that the loss or dispersal of radioactive material from the cask will be less than 10^{-6} A_2 per hour, as required by 10 CFR 71.51(a)(1).

4.3 Containment Under HAC

4.3.1 Pressurization of Containment Vessel

Within the thermal evaluation, presented in Chapter 3, the applicant demonstrated, and the staff confirmed that the pressure under HAC would not exceed the package design pressure.

4.3.2 Containment Criteria

The containment system is designed to a leakage rate of 4.2 x 10^{-6} ref-cm^3/sec or less. The applicant calculated the releaseable radiological source term and the maximum allowable leak rate under HAC. The applicant demonstrated that the NCT leakage rate bounds the permissible accident leakage rate.

4.3.3 Compliance with Containment Criteria

Results of the thermal analysis show that the seal temperatures will remain below the seal material temperature limits during and after the 30-minute fire. Results of the structural analysis show that the cask inner shell will not buckle under HAC.

Overall, results of the structural and thermal analyses also showed that the containment system remained intact under the tests specified in 10 CFR 71.73. The applicant demonstrated that the integrity of the containment boundaries will be maintained under HAC. Therefore, krypton gas releases are not expected to exceed 10 A_2 in a week, and the release of other radioactive materials are not expected to exceed A_2 in 1 week, as required by 10 CFR 71.51(a)(2).

The staff has reviewed the applicants' evaluation of the containment system under HAC and agrees with the conclusion that the package satisfies the containment requirements of 10 CFR 71.51(a)(2).

4.4 Evaluation Findings

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

5.0 SHIELDING

The objective of this review is to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under NCT and HAC.

5.1 Shielding Design Description

The UMS transport cask is constructed in a multi-walled shielding configuration in both the radial and axial directions. Type 304 stainless steel and lead provide the gamma shielding in the radial direction. There is a 2.0 inch stainless steel inner shell and a 2.75 inch outer shell, with 2.75 inches of chemical copper-grade lead filling the annulus between the two shells. Neutron shield material (4.5 inches thick) is also placed in an annulus that surrounds the cask outer shell along the length of the cask cavity and is enclosed by a stainless steel shell with top and bottom plates. The neutron shield material is a solid synthetic polymer (NS-4-FR). Twenty-four bonded copper and Type 304 stainless steel fins are located in the radial neutron shield to enhance the heat rejection capability of the cask and to support the neutron shield shell and end plates. The neutron shield is covered by 0.25 inches of stainless steel.

The bottom of the cask consists of the bottom forging, neutron shield, and bottom plate in a steel/NS-4-FR/steel configuration. The bottom forging is 4.3 inches of Type 304 stainless steel and the bottom plate is 5 inches of Type 304 stainless steel. In between the forging and the plate is 1.0 inches of NS-4-FR.

The cask lid when bolted to the top forging provides the sealed closure. The cask lid is 6.5 inches of SA-336, Type 304 stainless steel. The top forging is a ring that forms the upper end of the cask and is welded to the inner and outer shells. A vent port is

recessed into the lid and is protected by a cover plate made of SA-240, Type 304 stainless steel.

The staff reviewed the shielding design features and criteria and found them to be described acceptably.

5.2 Radiation Source

The fuel to be transported by the UMS is divided into five classes based upon length. There are three PWR and two BWR classes. The TSC will store up to 24 PWR or up to 56 BWR spent fuel assemblies. Based upon the evaluation performed by the applicant, the design basis PWR fuel is the Westinghouse 17 x 17 standard assembly with a burn up of 40,000 MWD/MTU, an initial enrichment of 3.7 wt.%^{235}U, and a 5-year cooling time. The design basis BWR fuel has been determined to be the GE 9 x 9 assembly with a burn up of 40,000 MWD/MTU, an initial enrichment of 3.25 wt.%^{235}U, and a 5-year cooling time.

SAR Table 1.2-4, PWR Fuel Assembly Characteristics and Table 1.2-5, BWR Fuel Assembly Characteristics identify the types and characteristics of the fuel to be transported.

The applicant used the SAS2H code of the SCALE4.3 computer code package to generate the gamma and neutron source terms. SAS2H includes the XSDRNPM neutronics and ORIGEN-S fuel depletion modules. Source terms were determined for the active fuel, plenum zone, top-fitting zone, and bottom-fitting zone of the spent nuclear fuel assemblies. Appropriate axial burn up profile parameters were applied for the design basis fuel in the source term modeling.

The gamma source term contains contributions from both fission products and actinides from the fuel, and activation products from the non-fuel hardware. The non-fuel hardware gamma source term is due primarily to ^{60}Co and is from the activation of the Type 304 stainless steel with a ^{59}Co impurity level of 1.2 g/kg. Minor contributions to the gamma non-fuel hardware source term are from ^{59}Ni and ^{58}Fe.

The neutron source term results from actinide spontaneous fission and the alpha-neutron reactions. The neutron spectra was determined using the 27-group neutron cross section library.

The applicant's methods for calculating the radiation source terms were reviewed. The staff used the SCALE-4.4 SAS2H and ORIGEN-S computer modules to perform confirmatory analyses and found acceptable agreement with the applicant's reported values.

5.3 Shielding Model

The applicant demonstrated compliance with the external radiation requirements specified in 10 CFR Parts 71.47 and 71.51 by using the three-dimensional SAS4 module of the SCALE computer code. The applicant used a one-dimensional analysis to identify the limiting PWR and BWR fuel. The dose rate results are presented in Section 5 of the SAR. All of the applicant's reported dose rate results are below the applicable 10 CFR Part 71 regulatory limits for exclusive-use transport in a closed transport vehicle.

The applicant calculated the dose rates for NCT and HAC for PWR and BWR fuel. UMS features such as trunnion recesses, ventilation port openings, and heat transfer fins were explicitly modeled. The hypothetical accident condition is modeled omitting the neutron shield, neutron shield shell, and impact limiters, and also assumed a simultaneous radial and axial slumping of the lead gamma shield. Table 5-1 and 5-2 show the maximum calculated doses for PWR and BWR fuel for both NCT and HAC.

Table 5-1, Calculated doses for the UMS package for NCT

Normal Conditions	Package Surface (mrem/h)		Personnel Barrier (mrem/h)		2 meters from Vehicle (mrem/h)	
Radiation	**PWR Fuel**	**BWR Fuel**	**PWR Fuel**	**BWR Fuel**	**PWR Fuel**	**BWR Fuel**
Gamma	66.2	50.4	23.5	20.7	6.4	4.0
Neutron	101.1	34.5	23.2	19.8	3.2	3.4
Total	167.2	84.8	46.6	40.4	9.6	7.4
Limit	1000	1000	200	200	10	10

Table 5-2, Calculated Doses for the UMS package for HAC

HAC	**1 meter from package surface (mrem/h)**	
Radiation	PWR Fuel	BWR Fuel
Gamma	44.8	25.2
Neutron	416.7	472.7
Total	461.5	497.9
Limit*	1000	1000

*Limits are for Exclusive Use shipments in a closed conveyance

5.4 Maine Yankee Site Specific Contents

The standard fuel assembly used at Maine Yankee was the CE 14 x 14 assembly. Fuel of the same design was also supplied by Westinghouse and by Exxon. The total number of fuel assemblies in inventory is approximately 1,434 assemblies. There are 90 assemblies with a burn up between 45,000 and 50,000 MWd/MTU, with the remaining assemblies having a burn up less than or equal to 45,000 MWd/MTU.

The standard source for Maine Yankee fuel assemblies with no additional non-fuel material was determined using the SAS2H model. Then, one-dimensional shielding calculations were performed for the fuel region sources at various combinations of initial enrichment, burn up, and cool time. The resulting dose rate and source term information was used to determine the cool time required for each combination of enrichment and burn up to decay below the design basis limiting values of dose rate and heat generation rates. The standard Maine Yankee 14 x 14 fuel assembly with no additional hardware

and burn up less than or equal to 45,000 MWd/MTU, is bounded by the original source term evaluation.

Some of the Maine Yankee fuel assemblies have additional hardware inserted into the assemblies. There are 168 assemblies with a CEA inserted and 138 assemblies with an ICI thimble inserted. Fuel assemblies containing CEA or ICI thimble hardware were specifically evaluated to establish the limiting values for storage.

The Maine Yankee CEA consists of five control rods that are inserted in the fuel assembly guide tubes. The rods are made of Inconel 625 or stainless steel and encapsulate B_4C as the primary neutron poison material. The ICI thimble is inserted in the center guide tube of the fuel assembly. The ICI thimble is made primarily of Zircaloy. Prior to being placed in the TSC, the detector material and lead wire are removed from the thimble assembly. The CEA and ICI thimble are non-fuel bearing components, which due to activation of the materials of composition, add to the gamma component of the source term of the fuel assembly.

The Maine Yankee spent fuel inventory also contains damaged fuel rods and fuel debris. Damaged fuel rods and fuel debris will be placed in a MYFC can prior to loading into the basket. Fuel debris will be placed into a rod structure prior to loading into the MYFC. For the damaged fuel and fuel debris placed in a rod structure, the shielding evaluation for an intact fuel assembly bound that of the damaged fuel rods. The effect of collapsed fuel in the fuel can was evaluated, and it was determined that no significant increases in personnel exposure would be expected as a result of the collapsed fuel.

The PWR canister can also be loaded with a basket containing GTCC. The GTCC waste basket is an enclosed cylinder, has the same dimensions as the Class 1 canister, and is constructed of Type 304 stainless steel. There is no generic design basis GTCC waste, since the type of waste and radioactive constituents vary from site to site. Therefore, a site specific evaluation was be done to evaluate the radiological impact from transporting GTCC waste.

For the Maine Yankee site, the GTCC waste consists of activated and contaminated steel. The major component of the GTCC waste is the stainless steel core baffle structure. The core baffle structure is located adjacent to the reactor vessel in a high neutron flux area resulting in the activation of the metal. The core baffle structure will be cut underwater and then loaded into a GTCC waste basket. The radionuclides that are the primary contributors to the gamma source term are ^{54}Mn, ^{55}Fe, ^{60}Co, and ^{63}Ni. There is no neutron source term associated with GTCC waste. The total Curie inventory to be stored in a GTCC waste basket is 5.8E+05 Curies, based on the Maine Yankee design basis GTCC waste.

Staff has reviewed the Maine Yankee site specific source term and shielding evaluations and has reasonable assurance that the source term and dose rates generated for the Maine Yankee fuel are bounded by the design basis PWR fuel.

5.5 Shielding Evaluation

Using the information provided in the SAR, the staff performed confirmatory analyses using the SAS2H/ORIGEN-S and SAS4 modules from the SCALE 4.4 computer code. These analyses were performed to confirm that the applicant's source term and dose rate calculation methodologies were satisfactory. The staff's dose rate results were found to be in agreement with those reported by the applicant. The dose rate results calculated by the staff and applicant were found to be below the applicable regulatory limits specified in 10 CFR 71.47 and 71.51. Based upon the information provided by the applicant and the confirmatory calculations, the staff believes that the calculated dose rates are representative of the dose rates which would be detected on the outside of the transport cask loaded with design basis PWR, BWR fuel, and Maine Yankee site specific contents.

5.6 Evaluation Findings

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

6.0 CRITICALITY

The objective of the review is to ensure that the UMS package design satisfies the criticality safety requirements of 10 CFR Part 71.

6.1 Description of Criticality Design

The package design criterion for criticality safety is that the effective neutron multiplication factor, k_{eff}, including statistical biases and uncertainties, shall not exceed 0.95 for all postulated arrangements of fuel within the cask under NCT and HAC.

6.1.1 Packaging Design Features

The packaging design features relied upon to prevent criticality are the basket geometry and the fixed neutron poisons in the basket. The criticality evaluation determines the most reactive mechanical perturbations and geometric tolerances for both the PWR and BWR baskets. The applicant assumed that 75% of the minimum specified Boron-10 (^{10}B) areal density is present for the fixed neutron absorber plates in each basket. The minimum ^{10}B areal density is 0.025 g/cm^2 for the Boral in the PWR basket and 0.011 g/cm^2 in the BWR basket. Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71.71 and 71.73.

6.1.2 Codes and Standards

The criticality evaluation is consistent with the appropriate codes and standards for nuclear criticality safety. The criticality evaluation is also consistent with the recommendations provided in NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages."

6.1.3 Summary Table of Criticality Evaluations

Table 6.1-1 of the SAR contains a summary of the final analysis results for the most reactive PWR and BWR assemblies, under both NCT and HAC. These results are for a single package and for arrays of damaged and undamaged packages, as required by 10 CFR 71.55 and 71.59. The results illustrate that the package meets the criticality safety criteria of 10 CFR Part 71 and that the package would remain subcritical under NCT and HAC.

The maximum k_{eff} for each condition, as calculated by the applicant, is summarized in Table 6-1 below. The results are below the Upper Subcritical Limit (USL), calculated in Section 6.5 of the SAR, which is 0.9361 for calculations using SCALE 4.3 with KENO V.a, and 0.9426 for calculations using MONK 8A. The results of the staff's confirmatory calculations are in close agreement with the applicant's results.

Table 6-1, Maximum k_{eff} ,
(SCALE 4.3 USL = 0.9361, MONK 8A USL = 0.9426)

Condition	Code Used	$k_{eff} + 2\sigma$	
		PWR[1]	BWR
Single Package, Flooded 10 CFR 71.55(b), (d), and (e)	SCALE	0.9265	0.9071
Infinite Array of Undamaged Packages, Dry 10 CFR 71.59(a)(1)	SCALE	0.4002	0.4021
Infinite Array of Damaged Packages, Flooded 10 CFR 71.59(a)(2)[2]	MONK	0.9351	0.9373

1. Bounds the results of the Maine Yankee site-specific criticality analysis.
2. Includes the results of the top end impact evaluation in Section 6.4.5.

6.1.4 Transport Index for Criticality Safety

The criticality analysis shows that an infinite rectangular array of undamaged or damaged packages will remain subcritical with close full-water reflection and optimum interspersed hydrogenous moderation. Therefore, per 10 CFR 71.59(b), the criticality transport index for the UMS Universal Transport Package is 0.0.

6.2 Spent Nuclear Fuel Contents

The UMS is designed to transport a maximum of 24 PWR or 56 BWR intact spent fuel assemblies. The characteristics of PWR and BWR assemblies to be transported in the UMS are listed in Tables 6.2-1 and 6.2-2 of the SAR. The number of fuel rods, pitch, uranium mass, initial enrichment, rod diameter, clad thickness, pellet diameter, guide tube thickness, and active length are controlled to maintain subcriticality. Maximum initial enrichment for PWR and BWR assemblies is 4.2 and 4.0 wt.% ^{235}U, respectively. The enrichment limit for BWR assemblies refers to the peak planar-average enrichment, which is the average enrichment of the pins across the assembly at any axial plane. The assembly parameters important to criticality safety are listed in Section 1.2.3 of the SAR.

Specifications for standard BWR and PWR fuel condition are also included such that fuel with cladding defects greater than pinhole leaks and hairline cracks may not be loaded into the package. Assemblies with missing pins are not allowed unless: 1) the missing pin is replaced by a solid Zircaloy or 304 stainless steel filler rod that displaces an equivalent or greater volume, or 2) the assembly is Maine Yankee site-specific and is loaded in one of the four corner positions of the PWR basket.

The UMS is also designed to transport up to 24 Maine Yankee site-specific intact 14 x 14 spent fuel assemblies with initial enrichments up to 4.2 wt.% ^{235}U and 176 fuel rods or solid filler rods. Fuel assemblies with variable pin enrichments, annular axial end blankets, or with the lattice dimensional variations discussed in Section 6.6.1.1.2 of the SAR are also allowed for transport. Additionally, the four corner guide tubes of the Maine Yankee site-specific canister can be used for the placement of non-standard fuel types, including:

- A. assemblies with less than 176 fuel rods or solid filler rods,
- B. assemblies with hollow rods,
- C. assemblies with fuel rods in the control rod guide tubes and a maximum of 176 total rods,
- D. 17 x 17 consolidated fuel lattices,
- E. assemblies with start-up sources or other non-fuel components inserted in a control rod guide tube, or
- F. Damaged fuel or fuel debris in a rod type structure, inside of a MYFC.

The staff reviewed the description of the spent nuclear fuel contents and agreed that all relevant specifications have been provided. The staff also verified that the specifications used in the criticality safety analysis are consistent with or bound those given in SAR Tables 1.2-4 and 1.2-5.

6.3 General Considerations for Evaluations

6.3.1 Model Configuration

The UMS packaging was modeled as infinite in length with external full water reflection. The models are based on the engineering drawings in section 1.3.4 of the SAR and considered various geometric tolerances and mechanical perturbations of the canister, including:

- A. fuel movement in the guide tube,
- B. guide tube movement in the spacer disk opening,
- C. combined fuel and guide tube movement,
- D. variation in guide tube opening size,
- E. variation in spacer disk opening size, and
- F. variation in the position of the disk opening.

The neutron absorber sheets are modeled as 75% of the minimum specified ^{10}B areal density in B_4C mixed with an aluminum matrix. For the PWR class canister, Section 6.4.1.3.1 of the SAR determined that the k_{eff} of the most reactive configuration could be increased by Δk_{eff} = 0.00246 when considering minimum width tolerances and shifting of the poison plates. This Δk_{eff} was added to the results of subsequent PWR class canister

models which did not incorporate poison plate tolerances and shifting. For the BWR class canister, a similar evaluation resulted in a Δk_{eff} that was not statistically significant compared to the Monte Carlo statistics of the code and was therefore not added to the results of BWR canister models.

The active fuel region was modeled explicitly, and was assumed to be infinite in length, thereby eliminating axial neutron leakage. Each canister model consisted of a spacer disk region, a heat transfer disk region, and two water regions in between disks, stacked accordingly inside the canister and cask shells, and then reflected axially to create an infinite height cask. The applicant did not take credit for fuel burn up or burnable absorbers in the fuel.

Under HAC, the applicant considered the effects of the 30-ft. end drops on the spent fuel axial poison plate coverage. For the top end impact, the applicant assumed that the basket and poison plates would remain in contact with the canister bottom, while the fuel assemblies would move up until in contact with the canister shield lid. The model also considered the potential for top end hardware deformation, and the movement of fuel into the fuel rod plenum. The bottom end impact was not considered, since it was shown in Section 6.4.5 of the SAR that the lack of axial poison coverage resulting from the top end impact is bounding.

The Maine Yankee site-specific fuel evaluation for intact fuel is bounded by that for the most reactive intact PWR assembly (W 17 x 17 OFA) under NCT and HAC. The fuel configurations allowed for transport in the MYFC, discussed in Section 6.6 of the SAR, are evaluated using the same model used for PWR assemblies. The results of the various fuel configuration models were compared to the most reactive PWR assembly model results to ensure all the Maine Yankee fuel configurations are bounded.

The staff reviewed the applicant's models and agreed that they are consistent with the description of the cask and contents given in SAR Sections 1 and 6, including engineering drawings. The staff also reviewed the applicant's calculations, and results for determining the worst-case manufacturing tolerance and fuel condition. Based on the information presented in the SAR, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances was incorporated into the calculation models.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety analysis models are provided in Section 6 of the SAR. No credit was taken for burnable absorbers in the fuel. The applicant's calculations take credit for 75% of the minimum required ^{10}B areal density in the neutron absorber plates. The fabrication requirements and acceptance criteria are outlined in SAR Section 8.1.7. Wet chemistry tests are performed on coupon samples taken from 20 randomly selected sheets from each set of 100 sheets. Section 8 of the SAR discusses the qualification and acceptance tests of the poison material.

The basket materials do not degrade such that there is any impact on criticality safety. The neutron absorber is a metal matrix composite material sandwiched between aluminum cladding and steel plates that meet all structural and thermal requirements and can be expected to have no significant erosion or corrosion. A structural analysis was

performed which demonstrates that the basket plates will remain in place during all accident conditions. The neutron flux in the package is very low such that depletion of the ^{10}B is negligible.

The material properties used in the calculation models were reviewed by the staff and determined to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant used the CSAS modules of the SCALE version 4.3 computer codes and the accompanying 27-group cross-section library for the infinite height cask models and associated benchmark calculations. For the finite height, explicit cask models used for the top end impact and MYFC evaluations, the applicant used the MONK 8A computer code with the JEF 2.2-based point energy neutron libraries. Both codes are widely used in the nuclear industry for performing criticality analyses. The staff agrees that the codes and cross-section sets used are appropriate for this particular cask design and contents.

6.3.4 Demonstration of Maximum Reactivity

A number of parametric cases were analyzed to determine the most reactive model under NCT and HAC. First, the most reactive PWR and BWR assemblies were determined, along with the most reactive mechanical perturbations and geometric tolerances for each canister. All PWR assemblies were placed in a single assembly model, consisting of a two-spacer plate slice of PWR canister guide tube, reflected in the x-y direction to make an infinite array, and with an axial periodic boundary condition to make an infinite height model. Each assembly was centered in the guide tube, and the web thickness between adjacent guide tubes was varied to approximate the variation in web thickness present in the PWR canister. The most reactive assembly determined from this analysis, the W 17 x 17 OFA, was then used in the same model while varying assembly location, guide tube and spacer disk opening size, guide tube location, and spacer disk opening location. The most reactive configuration was determined to be with the maximum guide tube and spacer disk opening, the assembly centered in the guide tube, and the guide tube and spacer disk opening shifted radially inward. This configuration was then placed within the full canister model and used to determine the maximum k_{eff} under NCT (infinite array, dry interior) and HAC (infinite array, no neutron shield, wet interior and fuel-clad gap).

To determine the most reactive BWR assembly, each assembly type is evaluated in a full-cask model. This model consisted of a two-spacer plate slice of the BWR canister in the transport cask, loaded with 56 assemblies, each centered in a guide tube. The model was given a periodic boundary condition in the axial direction in order to simulate an infinite height cask. The most reactive assembly determined using this model is the Ex/ANF 9 x 9 with 79 fuel rods. This assembly was then used to determine the most reactive mechanical perturbations and geometric tolerances of the BWR canister, similar to the most reactive configuration analysis performed for the PWR canister. Due to the asymmetric neutron absorber plate pattern in the BWR canister, it was necessary to consider component shifting radial inward, radial outward, left, right, top, bottom, and to the four basket corner locations. The most reactive configuration was that with the maximum tube and spacer disk openings, and with the fuel assembly, guide tube, and spacer disk opening shifted radially inward. This configuration was then used to

determine the maximum k_{eff} under NCT (infinite array, dry interior) and HAC (infinite array, no neutron shield, wet interior and fuel-clad gap).

For the top end 30-ft HAC impact analysis, a structural analysis of BWR assemblies under the 60g top end impact showed that the lifting bail would deform 2.371 inches, and that the top rod plenum spring would compress completely and then rebound 1.729 inches. A structural analysis for PWR assemblies showed no deformation in the top end hardware. The PWR rod plenum spring was assumed to remain fully compressed, since no analysis was performed to determine its response to the top end impact. The distance between the top of the active fuel and the top of the poison plates was calculated for all assembly types. The worst-case lack of axial poison coverage was 4.312 inches for the Ex/ANF 9 x 9 BWR assembly and 4.472 inches for the W 15 x 15 PWR assembly. The BWR and PWR lack of axial poison coverage was conservatively increased to 7.625 inches and 4.52 inches, respectively, and applied to the most reactive assemblies in a finite length, optimally moderated, explicit cask model to determine top end drop effect on maximum reactivity.

All of the Maine Yankee site-specific fuel configurations, including those limited to the four corner positions of the canister, are shown in Section 6.6 of the SAR to be bounded by the W 17 x 17 OFA assembly. For the MYFC damaged fuel analysis, 100% dispersal is considered from 176 damaged fuel rods, or from 289 consolidated fuel rods. All loose fuel is modeled as a homogeneous mixture of fuel and varying densities of water, since the damaged fuel can has screened holes allowing water entry. Analyses are performed placing this mixture in between the rods of the most reactive missing rod array, above and below the active fuel region of the most reactive missing rod array, and replacing the entire contents of the damaged fuel can. Preferential flooding is also modeled inside and outside the damaged fuel can in order to approximate uneven drain-down conditions. The most reactive condition was found to be with full water density inside both the canister and the MYFC.

Based on the applicant's results and the staff's independent confirmatory calculation as discussed below, the staff concludes that the most reactive combination of parameters and dimensional tolerances has been considered.

6.3.5 Confirmatory Analyses

The staff used the CSAS25 sequence of SCALE4.4a, along with the SCALE system's 44-group cross-section library for the confirmatory analysis. The SCALE 4.4a code system is routinely used for performing criticality analyses and is appropriate for this particular application and fuel system.

The staff's independent calculations were based on the information provided in the SAR. Similar to the applicant's model, the staff considered a small slice of the PWR and BWR canisters in the transport cask, reflected in the x-y plane to create an infinite array, and with a periodic boundary condition in the axial direction to create an infinite height cask. The cask slice consisted of one steel spacer disk region, one heat transfer disk region, and two intermediate water regions, alternately stacked in a four-unit array. The staff used the most reactive assemblies identified by the applicant, and modeled them shifted radially inward inside a guide tube with the maximum tolerance opening. This guide tube was also modeled shifted radially inward inside a maximum tolerance disk opening, which was in turn modeled shifted radial inward by the maximum disk location tolerance.

The canister was modeled with full internal moderation and the assembly models included moderator in the fuel-clad gap. Overall, the results of the confirmatory analysis performed by the staff are in close agreement with the applicant's results for both PWR and BWR canisters inside the UMS (including Maine Yankee site specific contents).

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff agrees with the applicants conclusion that the package will remain subcritical, under NCT and HAC.

6.4 Single Package Evaluation

The single package requirements of 10 CFR 71.55 are satisfied by the fully flooded PWR and top end impact BWR models used for the HAC array evaluation. Both of these models show the UMS in the most reactive credible condition with respect to the results of the HAC and with moderation by water to the most reactive credible extent. Modeling the damaged package in an array is expected to bound the single package, since the array model consists of a single package with a reflective boundary condition in the x-y plane, creating an infinite array. The results of the applicant's criticality analysis show that k_{eff} of an infinite array of UMS packages will remain below the design criterion of 0.95 for all allowed fuel loadings.

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.55(b), (d), and (e).

6.5 Evaluation of Package Arrays Under NCT

The evaluation of package arrays under NCT was performed by modeling an infinite array of infinite height packages, with the worst case mechanical perturbations and geometric tolerances, a dry interior and fuel-clad gap, and optimum interspersed water moderation. This evaluation resulted in a maximum PWR and BWR k_{eff} well below the design criterion of 0.95, as reported in Table 6.1-1 of the SAR. Since an infinite array of undamaged packages is determined to be subcritical, a transport index of zero for criticality control is acceptable under NCT.

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.59 for arrays of undamaged packages.

6.6 Evaluation of Package Arrays Under HAC

The evaluation of package arrays under HAC considered the worst case mechanical perturbations and geometric tolerances determined in Section 6.4.1.3 of the SAR, as well as optimum internal and interspersed moderation for an infinite array of infinite height packages. The applicant also considered the potential for active fuel to lose axial neutron poison coverage in a top end 30-ft. drop, as discussed in Section 6.4.5 of the SAR. Optimum moderation occurs with the packages fully flooded with 100% density water. The interspersed moderation has no significant impact on the reactivity because the thick walled gamma shield precludes neutron coupling between packages. As reported in Table 6.1-1, the k_{eff} of an infinite array of damaged packages, including biases and uncertainties, is below 0.95 for all fuel types considered. Since an infinite

array of packages is subcritical under both NCT and HAC, a transport index of 0 is acceptable, as determined according to 10 CFR 71.59(b)

The staff reviewed the applicant's evaluation and agreed with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.59(a)(1), 71.59(a)(2), and 71.59(b).

6.7 Benchmark Evaluations

6.7.1 Experiments and Applicability

The applicant performed benchmark comparisons on selected critical experiments that were chosen to bound the variables in the UMS cask design. The benchmark parameters bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, ^{10}B areal density in the poison plates, hydrogen to U-235 ratio, flux trap gap thickness, and average neutron group causing fission.

The staff reviewed the benchmark comparisons in the SAR and agreed with that applicants' conclusions that the CSAS module of the SCALE computer code and the MONK 8A code used for the analysis were adequately benchmarked to represent critical experiments relevant to the cask design and contents specified.

6.7.2 Bias Determination

In order to determine the bias 63 critical experiments were selected for SCALE 4.3 and 123 for MONK 8A to bound the range of parameters in the UMS design. USLs of 0.9361 and 0.9426 were calculated by the applicant for SCALE 4.3 and MONK 8A, respectively. The USL incorporates 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, which is the design criterion.

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations. For the MONK 8A benchmark calculations, the benchmark models were created by the code developer instead of the UMS analyst. This could potentially introduce bias uncertainty due to individual modeling technique and code input option differences between the benchmark models and the UMS models. The applicant provided an analysis showing that all the primary modeling and input options used in the UMS analysis were represented in the code developer's benchmark models. The most significant code feature affecting evaluation results, choice of neutron cross-section library, was the same (JEF 2.2) in all of the benchmark and UMS models. The few modeling and input options that are not represented in the benchmark models are not expected to significantly affect the uncertainty in the calculation bias.

The staff reviewed the applicant's methods for determining the USLs and found them to be acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.8 Evaluation Findings

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

7.0 OPERATING PROCEDURES

Operating procedures for the package are specified in Chapter 7 of the SAR. The chapter includes sections on package loading, unloading, preparation of an empty package for transport, and preparation of a package that was used in spent fuel storage.

The chapter discusses the procedure to dry the cask cavity using a vacuum drying system. The bolt torque levels are provided in Table 7-1.

The inner lid, inner vent, and cover seals are to be leak tested to a leak rate of not greater than 4.2×10^{-6} ref cm^3/sec (or 6.5×10^{-6} cm^3/sec, helium) in accordance with ANSI N14.5.

The procedures include the steps needed to prepare the UMS, the loading of the TSC, and the loading of the TSC into the UTC. The UMS is dry loaded in the spent fuel building or onsite ISFSI by using a transfer cask and attendant support hardware. The procedures also include the installation of the upper and lower impact limiters.

Surface doses and temperatures are checked to ensure that the regulatory limits are not exceeded.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Acceptance tests and the maintenance program for the package are specified in Chapter 8 of the SAR. The acceptance tests from Section 8.1 of the SAR include visual inspections, structural and pressure tests, leak tests, component tests, shielding tests, thermal acceptance test, and neutron absorber tests. The maintenance program is also specified in Chapter 8.2 of the SAR.

Section 8.2.2 and Table 8.2-1 of the SAR describes the containment periodic verification leak test. The periodic verification leak test will be performed on each cask after the third use, and every 12 months thereafter, and whenever a replaceable containment component is replaced. Leak tests will be performed in accordance with the methodologies and requirements of ANSI N14.5-1997. Personnel performing tests shall be qualified in accordance with Section 8.5 of ANSI N14.5-1997.

The bolts used for the cask structural lid will be reused after each shipment. Periodic inspections of these bolts will be performed to ensure there is no excessive wear, fatigue cracks, or other adverse conditions.

Impact limiters will be visually inspected prior to each use. Additionally, the impact limiters will be weighed once every five years to verify that the impact limiter is within +/- 3% of the original weight.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure that the package will be acceptance tested and maintained in a manner consistent with its evaluation for approval.

9.0 CONDITIONS

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

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, “Operating Procedures,” of the application, as supplemented.

(b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, “Acceptance Tests and Maintenance Program,” of the application, as supplemented.

10.0 CONCLUSION

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

Issued with Certificate of Compliance No. 9270, Revision No. 0
on October 31, 2002.