NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9270	7	71-9270	USA/9270/B(U)F-96	1	OF	19

2. PREAMBLE

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

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

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

a. ISSUED TO *(Name and Address)*

NAC International, Inc.
2 Sun Court, Suite 220
Peachtree Corners, GA 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

NAC International, Inc. application dated August 21, 2025.

4. CONDITIONS

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

5.

(a) Packaging

(1) Model No.: UMS Universal Transport Cask Package

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

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 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 zirconium alloy 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.

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

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 PWR canister is positioned by a stainless steel spacer that is 16.75 inches in length. Each Class 2 PWR canister is positioned by a stainless steel spacer that is 7.65 inches in length. No spacers are used with the Class 3 PWR canister. The Class 4 BWR canister is located by four 1.5 inch aluminum spacers and the Class 5 BWR canister is located with a 1.5 inch aluminum spacer.

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 neutron absorber on all four sides of the tubes. Three types of fuel tubes are designed to contain the BWR fuel: (1) tubes containing neutron absorber on two sides of the tubes; (2) tubes containing neutron absorber on one side; and (3) tubes containing no neutron absorber. 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.

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

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 primarily of 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

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

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

5.(a)(3) Drawings

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

790-209, Rev. 1	790-210, Rev. 1	790-500, Rev. 4	790-501, Rev. 3
790-502, Rev. 7	790-503, Rev. 3	790-504, Rev. 2	790-505, Rev. 2
790-508, Rev. 2	790-509, Rev. 3	790-516, Rev. 3	790-519, Rev. 2
790-520, Rev. 2	790-570, Rev. 4	790-571, Rev. 3	790-572, Rev. 4
790-573, Rev. 7	790-574, Rev. 3	790-575, Rev. 10	790-581, Rev. 9
790-582, Rev. 12	790-583, Rev. 8	790-584, Rev. 19	790-585, Rev. 19
790-587, Rev. 1	790-591, Rev. 6	790-592, Rev. 8	790-593, Rev. 7
790-594, Rev. 2	790-595, Rev. 10	790-605, Rev. 11	790-611, Rev. 6
790-612, Rev. 9	412-501, Rev. 4	412-502, Rev. 6	

5.(b) Contents

(1) Type and Form of Material

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

i. 24 intact irradiated PWR fuel assemblies within a TSC;
ii. 56 intact irradiated BWR fuel assemblies within a TSC;
iii. 24 Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee within a TSC; or
iv. GTCC waste from Maine Yankee within a TSC.

5.(b) Contents (Continued)

Each type of package contents is described in detail below.

(i) Intact PWR assemblies

The package is designed to transport 24 irradiated intact PWR fuel assemblies 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 zirconium alloy or Type 304 stainless steel, which displaces an equal or greater volume than that occupied by a fuel rod.

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimensions and specifications of Table 5.(b)(1)(i)-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 5.(b)(1)(i)-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 5.(b)(1)(i)-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 zirconium alloy type clad.

2 Vendor ID indicates the source of assembly base parameters. Loading of assemblies meeting dimensional limits is 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 5.(b)(1)(i)-2, Loading Table for Intact PWR 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)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	6	8	8	7	8	8	10	11	9	10
2.1 ≤ E < 2.3	6	7	8	6	7	7	10	10	8	10
2.3 ≤ E < 2.5	6	7	7	6	7	7	9	10	8	9
2.5 ≤ E < 2.7	6	7	7	6	7	7	9	9	7	8
2.7 ≤ E < 2.9	6	7	7	6	7	6	8	9	7	8
2.9 ≤ E < 3.1	5	7	7	6	6	6	8	8	7	8
3.1≤ E < 3.3	5	6	7	6	6	6	8	8	7	7
3.3 ≤ E < 3.5	5	6	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	5	6	6	6	6	6	7	7	6	7
3.7 ≤ E ≤ 4.2	5	6	6	6	6	6	7	7	6	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)				
	CE 14x14	14x14	15x15	16x16	17x17	CE 14x14	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	11	15	15	13	15	18	20	21	20	20
2.1 ≤ E < 2.3	10	13	14	12	13	15	19	19	18	19
2.3 ≤ E < 2.5	9	12	13	11	12	14	17	19	17	17
2.5 ≤ E < 2.7	9	12	12	10	11	12	16	18	15	17
2.7 ≤ E < 2.9	8	11	11	9	11	11	15	18	14	17
2.9 ≤ E < 3.1	8	10	10	9	10	10	14	18	13	15
3.1 ≤ E < 3.3	7	10	10	9	10	10	13	17	13	15
3.3 ≤ E < 3.5	7	9	10	8	9	9	12	17	13	15
3.5 ≤ E < 3.7	7	9	10	8	9	8	11	17	12	15
3.7 ≤ E ≤ 4.2	7	8	10	8	8	8	11	15	12	14

5.(b)(1)(ii) Intact BWR assemblies

The 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 planar-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.

5.(b)(1)(ii) Intact BWR assemblies (continued)

The fuel assemblies consist of uranium dioxide pellets with zirconium alloy type cladding. Prior to irradiation, the fuel assemblies must be within the dimension and specifications of Table 5.(b)(1)(ii)-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 5.(b)(1)(ii)-2.

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 zirconium alloy or Type 304 stainless steel.

Table 5.(b)(1)(ii)-1, Intact BWR Fuel Assembly Characteristics

Canister Class [1,5]	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[5]	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 zirconium alloy type 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 5.(b)(1)(ii)-2, 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

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee

The 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 intact PWR CE 14x14 (see section 5.(b)(1)(i)).

5.(b)(1)(iii) Intact and Damaged PWR assemblies, and Fuel Debris from Maine Yankee (continued)

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:

5.(b)(1)(iii)(A) Maine Yankee's site specific contents not requiring preferential loading patterns:

(1) Standard Irradiated CE 14 x 14 intact PWR fuel assemblies meeting the PWR fuel assembly characteristics in Table 5.(b)(1)(i)-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 5.(b)(1)(iii)(A)-1.

(2) 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 5.(b)(1)(iii)(A)-1 except for those assemblies containing ICI thimbles which must meet the specifications of Table 5.(b)(1)(iii)(A)-2.

(3) PWR intact fuel assemblies with fuel rods replaced with stainless steel or zirconium alloy 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 5.(b)(1)(iii)(A)-3.

5.(b)(1)(iii)(A) (Continued)

(4) 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 have 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 5.(b)(1)(iii)(A)-1.

(5) 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 5.(b)(1)(iii)(A)-1.

(6) 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 5.(b)(1)(iii)(A)-1.

(7) 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 5.(b)(1)(iii)(A)-1.

5.(b)(1)(iii)(B) 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 5.(b)(1)(iii)(B)-1.

(1) 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 5.(b)(1)(iii)(A)-

1.
5.(b)(1)(iii)(B) (Continued)

(2) 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 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner loading position in the PWR fuel basket.

(3) PWR intact fuel assemblies with burnable poison rods replaced by hollow zirconium alloy 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 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(4) 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 5.(b)(1)(iii)(A)-1. These assemblies must be placed in a corner PWR fuel basket loading position.

(5) 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 5.(b)(1)(iii)(A)-1. CEA fingertips are not considered as CEAs for determination of minimum cool times.

5.(b)(1)(iii)(C) 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 a 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 MYFC. The contents that must be loaded in the MYFC are:

(1) 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 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(2) 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

5.(b)(1)(iii)(A)-4 for damaged fuel rods.

5.(b)(1)(iii)(C) (Continued)

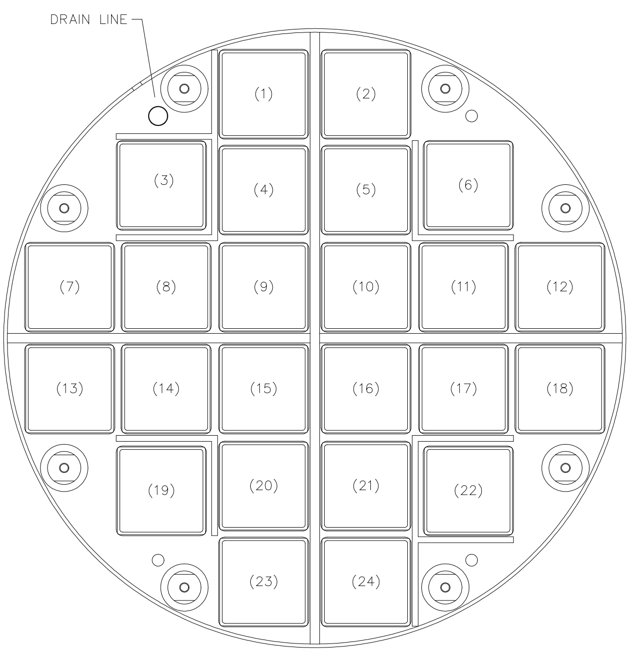
(3) Individual intact or damaged fuel rods in a rod type structure, which may be a guide tube, to maintain configuration control. The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-1 for intact fuel rods and Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(4) 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 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

(5) 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.

(6) High burn up fuel assemblies not meeting the oxide layer thickness criteria previously defined in Section 5.(b)(1)(iii)(B)(1). The combined maximum average burn up, minimum cool time and maximum and minimum initial ^{235}U enrichments must be within the specifications of Table 5.(b)(1)(iii)(A)-4 for damaged fuel rods.

PWR Basket Fuel Loading Position Diagram, Figure 5.(b)(1)(iii)(B)-1



1. Basket corner positions are positions 3, 6, 19, and 22. 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. Periphery positions include the corner

positions.

5.(b)(1)(iii)(C) (Continued)

Table 5.(b)(1)(iii)(A)-1, 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	6	5	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

5.(b)(1)(iii)(C) (Continued)

Table 5.(b)(1)(iii)(A)-1, continued , 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

5.(b)(1)(iii)(C) (Continued)

Table 5.(b)(1)(iii)(A)-1, continued, 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 5.(b)(1)(iii)(A)-2, 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

5.(b)(1)(iii)(C) (Continued)

Table 5.(b)(1)(iii)(A)-3, 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 5.(b)(1)(iii)(A)-4, Cool time (years) for Maine Yankee CE 14x14 damaged fuel

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	7	11	19	28	37
2.1 ≤ E < 2.3	6	9	16	26	34
2.3 ≤ E < 2.5	6	8	14	23	32
2.5 ≤ E < 2.7	6	8	12	21	30
2.7 ≤ E < 2.9	6	7	11	19	27
2.9 ≤ E < 3.1	6	7	10	17	25
3.1 ≤ E < 3.3	5	7	9	15	23
3.3 ≤ E < 3.5	5	6	8	13	21
3.5 ≤ E < 3.7	5	6	8	12	19
3.7 ≤ E ≤ 4.2	5	6	7	11	17

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee

The 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 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 71.15. The maximum curie inventory shall not exceed the values shown in Table 5.(b)(1)(iv)-1.

5.(b)(1)(iv) Greater Than Class C Waste from Maine Yankee (continued)

Table 5.(b)(1)(iv)-1, 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

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

The maximum weight of the contents shall not exceed 77,500 pounds.

(i) For the contents described in 5.(b)(1)(i) and 5.(b)(1)(iii): 24 PWR fuel assemblies, including standard inserts such as burnable poison rods or guides or guide tube thimble plugs, with a maximum weight of 38,500 pounds and a maximum decay heat limit per package not to exceed the values in Table 5.(b)(2)-1. The individual PWR assembly decay heat is limited to 0.83 kW.

Table 5.(b)(2)-1, 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

(ii) For the contents described in 5.(b)(1)(ii): 56 BWR assemblies with a maximum weight of 39,000 pounds and a maximum decay heat limit per package of 16 kW. The individual BWR assembly decay heat is limited to 0.29 kW.

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

(iii) For the contents described in 5.(b)(1)(iv): GTCC waste with a maximum weight per package of 20,000 pounds in total or 10,000 pounds per compartment. The maximum decay heat for the GTCC is 4.5 kW per package.

5.(c) Criticality Safety Index 0.0

6. The package must be transported as exclusive use in a closed transport vehicle as per 10 CFR 71.47(b).

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

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

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented.

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

9. Transport by air of fissile material is not authorized.

10. Revision No. 6 of this certificate may be used until August 31, 2026.

11. Expiration date: October 31, 2027.

REFERENCES

NAC International, Inc., Application dated August 21, 2025.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

YOIRA DIAZ-
SANABRIA

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SANABRIA
Date: 2025.09.09 15:54:02 -04'00'

Yoira K. Diaz Sanabria, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: See digital signature

SAFETY EVALUATION REPORT

Docket No. 71-9270
Model No. UMST
Certificate of Compliance No. 9270
Revision 7

SUMMARY

By application dated December 20, 2024 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML24355A159), as supplemented on June 23, 2025 (ML25174A270), NAC International (NAC or the applicant) applied for an amendment to Certificate of Compliance No. 9270 for the Model No. NAC UMS Universal Transport (UMST) Package. NAC requested an amendment to correct a licensing basis deficiency initially reported to the NRC on March 10, 2023 (ML23069A215). The report identified that a parameter used in the computation of bending stress in the finite element model used to structurally evaluate a fuel rod under the non-mechanistic tip-over accident condition was incorrectly specified resulting in the non-conservative calculation of stresses. Also, by application dated August 21, 2025 (ML25233A147), NAC submitted a consolidated SAR.

During the extent of condition review for the error discussed in the March 10, 2023, report, it was identified that the UMST safety analysis report (SAR) did not include a fuel rod analysis for the 30-foot transport cask side drop Hypothetical Accident Condition (HAC). During the development of corrective actions for NAC's other spent fuel transport packages NAC decided to add this analysis to the UMST SAR for completeness and consistency. SAR Sections 2.9, 2.9.1.1, 2.9.3, 2.9.4, and 2.12 have been revised.

Following staff review of the associated SAR, the staff finds that the changes do not affect the ability of the package to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

EVALUATION

2.0 Structural Evaluation

The objective of the structural evaluation is to verify that the applicant has adequately evaluated the structural performance of the proposed transport package and demonstrated that it satisfies the regulations in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.”

The staff evaluated the proposed changes in the SAR, revision 25A, in accordance with the guidance of NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” USNRC, 2020 (ML20234A651). This section documents the staff’s review, evaluation, and conclusions with respect to the structural safety aspects of the proposed transport package, as well as specific material property definitions.

2.1. Description of Structural Design

In this amendment, the applicant has proposed the addition of the fuel rod evaluation for a 30-foot transport cask side drop event, and submitted the following document in support of this evaluation: EA790-2524, “Fuel Rod Evaluation for the UMST 30-ft Side Drop Accident,” Revision 1.

2.1.1 Fuel Rod Evaluation for 30-foot Side Drop

The applicant documented the analyses of fuel rods for the HAC 30-foot transport cask side drop in calculation EA790-2524 and added SAR sections 2.9.3 and 2.9.4 to describe the pressurized water reactor (PWR) and boiling water reactor (BWR) fuel rod evaluations, respectively. The applicant performed a number of evaluations to represent the effects of this HAC event on 26 PWR (11 fuel types) and 26 BWR (6 fuel types) high burnup fuel assemblies, at a bounding fuel temperature of 400 °C (752 °F), the maximum permitted cladding temperature during normal transport conditions for all cladding types except M5, which was evaluated at 356 °C (673 °F). The applicant employed the finite element program ANSYS for these analyses.

Each evaluation considered a single, empty fuel rod with the cladding weight adjusted for the mass of the missing fuel and applied a cladding flexural rigidity factor of 1.25 to adjust for the absence of the fuel pellet, as recommended by section 2.3.4 of NUREG-2224, *Dry Storage and Transportation of High Burnup Spent Nuclear Fuel,* USNRC, November 2020 (ML21091A321). Cladding thicknesses for both PWR and BWR rods are also reduced to account for oxide layers for the various cladding materials, as recommended by figure 2-5 of NUREG-2224. For the PWR fuel rod analyses, the applicant considers PWR fuel rods with and without grid damage. For the PWR fuel rod without grid damage, the applicant considers the nominal as-built spacing between grids, while for the PWR fuel rod with grid damage, a maximum unsupported length of 60 inches is considered to account for fuel rods with missing, slipped, or damaged grids, which the applicant determines to be a bounding configuration. For the BWR fuel rod analyses, the applicant considers the unsupported lengths to be the nominal, as-built configuration. For both PWR and BWR fuel rods, the applicant limits deflections in the analyses by the space between adjacent fuel rods as well as that between the fuel tube wall. The staff finds the applicant's choice of analysis parameters and method of evaluation to be acceptable as they are either conservative or follow the guidance provided.

For the PWR rod analyses, the applicant considered high burnup conditions and cladding alloys that are typical for PWR fuel. Similarly, the applicant considered high burnup conditions and cladding alloys that are typically used for BWR fuel. The fuel rod and fuel pellet material properties of elastic modulus, density and yield strength, as applicable, are taken from various references.

The staff reviewed the applicant's selection of material properties for the cladding alloys to support the analyses. In addition, the staff reviewed information on cladding material properties not cited by the applicant including:

- Shimskey, R., et al. "FY2014 PNNL Zr Cladding Testing Status Report," PNNL-23594, August 30, 2014.
- Wells, B.E., et al. "Evaluation of Increased Peak Temperatures for Spent Fuel Cladding Performance during Dry Storage," PNNL-30430, Rev. 1, September 2020.

The staff determined that the material properties for the BWR fuel cladding alloys used by the applicant in their side drop evaluation were acceptable because the values were obtained by measurements on irradiated cladding alloys at temperatures consistent with the applicant's analyses. Similarly, the staff determined that the material properties for most of the PWR fuel cladding alloys used by the applicant were obtained by measurements on the irradiated cladding samples at elevated temperatures and were therefore acceptable.

The staff noted that the data used by the applicant for one of the PWR cladding alloys was obtained using tensile tests under non-quasi-static testing conditions. Zirconium-based fuel cladding alloys are known to strain harden as a function of strain rate which, in turn, increases the measured yield strength. Based on the information provided in Shimskey et al. (2014) and Wells et al., (2020), the staff determined that the yield strength of the cladding alloy would be increased by approximately 6 percent under the testing conditions in the reference cited by the applicant. The staff also determined that publicly available data for irradiated cladding material properties at elevated temperatures is limited and alternative references for properties for the alloy under quasi-static testing conditions are not available. After reviewing the available data and the applicant's analysis, the staff determined that the cited material properties for the cladding alloy were acceptable because: (1) the measurements were conducted using irradiated materials over a range of temperatures that bound the applicant's analyzed maximum temperature for the cladding alloy under transportation conditions, (2) the strain rates under drop accident conditions would be greater than quasi-static strain rates typically used to determine material properties, and (3) use of yield strength as an acceptance criteria is conservative because all zirconium-based cladding alloys strain harden above the yield stress and retain measurable ductility after irradiation.

The acceleration values applied to each fuel rod in the analyses were determined through the use of a dynamic load factor based on the modal analysis of each fuel rod, following the general principles presented in section 2.3.5.2 of NUREG-2224. The applicant created a broadened and enveloped response spectra from the acceleration time histories of the LS-DYNA finite element analyses of the 30-foot cask drops. The resulting acceleration values, applied uniformly along the entire length of the fuel rods in the analyses, were 39 g for the PWR rods with grid damage, 59 g for the PWR rods without grid damage, and 61 g for the BWR rods. The staff find this method of acceleration determination to be acceptable as it follows guidance and the resulting applied acceleration values to be acceptable as they are conservative.

For the final analysis, one PWR fuel rod with grid damage and one BWR fuel rod are chosen by the applicant based on the associated cladding section properties and inertia loading that are expected to produce bounding cladding stresses during the 30-foot transport cask side drop event. For both the PWR and BWR rod analyses, the factor of safety for all cladding yield stress versus actual stresses were greater than a value of one for all material types, indicating that the fuel rod cladding material meets the acceptance criteria during the HAC 30-foot cask side drop event. Based on these results, the staff finds the fuel rods to be structurally adequate for the HAC event required by 10 CFR 71.73(c)(1).

2.2. Evaluation Findings

The staff reviewed the amendment package for the additional HAC analyses and concludes that it satisfies the requirements of 10 CFR 71.73(c)(1).

The staff reviewed the structural and material performance of the package under the HAC required by 10 CFR Part 71.73 and concludes that it satisfies the requirements of 10 CFR 71.51(a)(2) for a Type B package and 71.55(e) for a fissile package.

Based on review of the statements and representations in the application amendment request, the NRC staff concludes that the UMST package has been adequately described and evaluated to demonstrate that it satisfies the structural integrity and material performance requirements of 10 CFR Part 71.

CONDITIONS

In addition to small editorial changes, the following changes have been made to the certificate:

- Condition 3(a) has been edited to reflect the applicant's new address.
- Condition No. 10 has been edited to allow use of the previous certificate (revision 6) for up to a year.
- Condition 11 now states the expiration date of the certificate.
- The references section has been updated to reflect the date of the consolidated application.

CONCLUSION

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

Issued with Certificate of Compliance No. 9270, Revision No. 7.