CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9372	5	71-9372	USA/9372/B(U)F-96	1	OF	13

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)*

Framatome, Inc.
2101 Horn Rapids Rd.
Richland, WA 99354

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Framatome TN-B1 Safety Analysis Report, FS1-0014159, Revision No. 11, dated January 31, 2023, as supplemented.

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.: TN-B1

(2) Description

The Model No. TN-B1 package is a rectangular box, 742 mm (29.21 in) high by 720 mm (28.35 in) wide by 5,068 mm (199.53 in) long, designed for the transport of unirradiated fuel assemblies or individual fuel rods with an enrichment up to 8.0 weight percent U-235. The package carries a maximum of (i) two Boiling Water Reactor (BWR) fuel assemblies or individual rods, containing enriched commercial grade uranium or of uranium with a trace amount of materials as defined in Table 2 of this CoC, or (ii) uranium oxide generic pressurized water reactor (PWR), or uranium carbide loose fuel rods in a 5 inch diameter stainless steel pipe.

The package is comprised of one inner container and one outer container both made of stainless steel. The inner container has a double-wall stainless steel sheet structure with an alumina silicate thermal insulator, filling the gap between the two walls, to reduce the flow of the heat into the contents in the event of a fire. Foam polyethylene or rubber cushioning material is placed on the inside of the inner container for protection of the fuel assembly.

The outer container is comprised of a stainless-steel angular framework covered with stainless steel plates. Inner container clamps are installed inside the outer container with damping devices to minimize vibrations during transport. Wood and honeycomb resin-impregnated kraft paper act as shock absorbers. The fuel rod clad and ceramic nature of the fuel pellets provide primary containment of the radioactive material.

5.(a)(2) continued

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

Maximum gross shipping weight	1,614 kg (3,558 lbs)
Maximum weight of inner container	308 kg (679 lbs)
Maximum weight of outer container	622 kg (1,371 lbs)
Maximum weight of packaging	930 kg (2,050 lbs)
Dimensions of inner container	
Length	4,686 mm (184.49 in)
Width	459 mm (18.07 in)
Height	286 mm (11.26 in)
Dimensions of outer container	
Length	5,068 mm (199.53 in)
Width	720 mm (28.35 in)
Height	742 mm (29.21 in)

(3) Drawings

This packaging is constructed in accordance with the TN-B1 Drawing Nos.:

Outer Container Drawings
105E3737, Rev. 6
FS1-0042698, Rev. 1
FS1-0042699, Rev. 1
FS1-0042700, Rev. 2
105E3741, Rev. 1
105E3742, Rev. 3
FS1-0042703, Rev. 1
02-9162717, Rev. 1

Inner Container Drawings
FS1-0042705, Rev. 2
105E3746, Rev. 1
FS1-0042707, Rev. 1
FS1-0042708, Rev. 1
02-9162722, Rev. 1

Contents Containers
105E3773, Rev. 1
0028B98, Rev. 1

5.(b) Contents

(1) Type and form of material

(i) ≤ 5.0 weight percent U-235

Enriched commercial grade uranium, or slightly contaminated uranium with trace quantities limits, as specified in Table 2 below. Uranium oxide or uranium carbide fuel rods enriched to no more than 5.0 weight percent U-235, with weight limits specified in Table 1 below.

(ii) > 5.0 to ≤ 8.0 weight percent U-235

Enriched commercial grade uranium, or slightly contaminated uranium with trace quantities limits, as specified in Table 3 below. Uranium oxide or uranium carbide fuel rods enriched to no more than 8.0 weight percent U-235, with weight limits specified in Table 1 below.

Table 1: Maximum weight of uranium dioxide pellets per fuel assembly

Type 8x8 fuel assembly	**Type 9x9 fuel assembly**	**Type 10x10 fuel assembly**	**Type 11x11 fuel assembly**
235 kg	240 kg	275 kg	281 kg

5.(b)(1) continued

Table 2: Maximum Concentrations of Authorized Contamination Material
for ≤ 5.0 weight percent U-235

Isotope	Maximum content
U-232	2.00×10^{-9} g/gU
U-234	2.00×10^{-3} g/gU
U-235	5.00×10^{-2} g/gU
U-236	2.50×10^{-2} g/gU
U-238	Balance of Uranium
Np-237	1.66×10^{-6} g/gU
Pu-238	6.20×10^{-11} g/gU
Pu-239	3.04×10^{-9} g/gU
Pu-240	3.04×10^{-9} g/gU
Gamma Emitters	5.18×10^{5} MeV - Bq/kgU

Maximum content of U-238 is 9.23×10^{-1}g/gU for a maximum U-235 concentration of 5%. Since, for concentrations less than 5%, the U238 value will be higher, it is shown as “Balance of Uranium” in Table 2.

5.(b)(1) continued

Table 3: Maximum Concentrations of Authorized Contamination Material
for > 5.0 to ≤ 8.0 weight percent U-235

Isotope	Maximum content
U-232	1.10×10^{-7} g/gU
U-234	7.65×10^{-3} g/gU
U-235	8.00×10^{-2} g/gU
U-236	2.50×10^{-2} g/gU
U-238	Balance of Uranium
Np-237	1.66×10^{-6} g/gU
Pu-238	6.20×10^{-11} g/gU
Pu-239	3.04×10^{-9} g/gU
Pu-240	3.04×10^{-9} g/gU
Gamma Emitters	5.18×10^{5} MeV - Bq/kgU

Maximum content of U-238 is 8.87×10^{-1}g/gU for a maximum U-235 concentration of 8%. Since, for concentrations less than 8%, the U238 value will be higher, it is shown as “Balance of Uranium” in Table 3.

5.(b)(1) continued

(i) 8 x 8 fuel assemblies comprised of 60 to 64 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod and poison rod specifications are in accordance with Table 4 below.

(ii) 9 x 9 fuel assemblies comprised of 72 to 81 rods in a square array with a maximum active fuel rod length of 381 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod and poison rod specifications are in accordance with Table 4 below.

(iii) 10 x 10 fuel assemblies comprised of 91 to 100 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod and poison rod specifications are in accordance with Table 4 below.

(iv) 11 x 11 fuel assemblies comprised of 112 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod and poison rod specifications are in accordance with Table 5 below for ≤ 5.0 weight percent U-235.

(v) 11 x 11 fuel assemblies comprised of 112 rods in a square array with a maximum active fuel rod length of 385 cm. The maximum pellet diameter, minimum clad thickness, rod pitch, water rod and poison rod specifications are in accordance with Table 6 below for > 5.0 to ≤ 8.0 weight percent U-235.

(vi) Uranium oxide fuel rods configured loose, in a 5-inch diameter schedule 40 stainless steel pipe/protective case or strapped together. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 7 below for ≤ 5.0 weight percent U-235.

(vii) Uranium carbide or generic PWR uranium oxide fuel rods configured loose, in a 5-inch diameter schedule 40 stainless steel pipe/protective case. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 7 below for ≤ 5.0 weight percent U-235.

(viii) Uranium oxide fuel rods or 17x17 PWR uranium oxide fuel rods configured loose, in a 5-inch diameter schedule 40 stainless steel pipe. When fuel rods are placed in polyethylene sleeves, each polyethylene sleeve shall not exceed 0.0152 cm in thickness. The maximum pellet diameter, minimum clad thickness, and rod specifications are in accordance with Table 8 below for > 5.0 to ≤ 8.0 weight percent U-235.

Fuel rods, assembled into fuel assemblies, contain sintered pellets of uranium oxides and/or sintered pellets of uranium oxides mixed with various additives such as chromia, gadolinia, and silica.

5.(b)(1) continued

Table 4: Fuel Assembly Parameters (8x8, 9x9, 10x10) for ≤ 5.0 weight percent U-235.

Parameter	**Units**	**Type**	**Type**	**Type**	**Type**
Fuel Assembly Type	Rods	8x8	9x9	FANP 10x10	GNF 10x10
UO_2 Density	%	≤ 98% Theoretical	≤ 98% Theoretical	≤ 98% Theoretical	≤ 98% Theoretical
Number of water rods[a]	#	0, 2x2	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3	0, 2-2x2 off-center diagonal, 3x3
Number of fuel rods	#	60 - 64	72 - 81	91 - 100	91 - 100
Fuel Rod OD	cm	≥ 1.176	≥ 1.093	≥ 1.000	≥ 1.010
Fuel Pellet OD	cm	≤ 1.05	≤ 0.96	≤ 0.895	≤ 0.895
Cladding Type		Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Cladding ID	cm	≤ 1.10	≤ 1.02	≤ 0.933	≤ 0.934
Cladding Thickness	cm	≥ 0.038	≥ 0.036	≥ 0.033	≥ 0.038
Active fuel length	cm	≤ 381	≤ 381	≤ 385	≤ 385
Nominal Fuel Rod Pitch	cm	1.63	≤ 1.45	≤ 1.30	1.30
U-235 Pellet Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Maximum Lattice Average Enrichment	wt%	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
Channel Thickness[b]	cm	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048	0.17 – 0.3048
Partial Length Fuel Rods (1/3 through 2/3 normal length)	Max #	None	12	14	14
Gadolinia Requirements Lattice Average Enrichment[c]	# @ wt% Gd_2O_3				
≤ 5.0 wt% U-235		7 @ 2 wt%	10 @ 2 wt%	12 @ 2 wt%	12 @ 2 wt%
≤ 4.7 wt% U-235		6 @ 2 wt%	8 @ 2 wt%	12 @ 2 wt%	12 @ 2 wt%
≤ 4.6 wt% U-235		6 @ 2 wt%	8 @ 2 wt%	10 @ 2 wt%	10 @ 2 wt%
≤ 4.3 wt% U-235		6 @ 2 wt%	8 @ 2 wt%	9 @ 2 wt%	9 @ 2 wt%
≤ 4.2 wt% U-235		6 @ 2 wt%	6 @ 2 wt%	8 @ 2 wt%	8 @ 2 wt%
≤ 4.1 wt% U-235		4 @ 2 wt%	6 @ 2 wt%	8 @ 2 wt%	8 @ 2 wt%
≤ 3.9 wt% U-235		4 @ 2 wt%	6 @ 2 wt%	6 @ 2 wt%	6 @ 2 wt%
≤ 3.8 wt% U-235		4 @ 2 wt%	4 @ 2 wt%	6 @ 2 wt%	6 @ 2 wt%
≤ 3.7 wt% U-235		2 @ 2 wt%	4 @ 2 wt%	6 @ 2 wt%	6 @ 2 wt%
≤ 3.6 wt% U-235		2 @ 2 wt%	4 @ 2 wt%	4 @ 2 wt%	4 @ 2 wt%
≤ 3.5 wt% U-235		2 @ 2 wt%	2 @ 2 wt%	4 @ 2 wt%	4 @ 2 wt%
≤ 3.3 wt% U-235		2 @ 2 wt%	2 @ 2 wt%	2 @ 2 wt%	2 @ 2 wt%
≤ 3.1 wt% U-235		None	2 @ 2 wt%	2 @ 2 wt%	2 @ 2 wt%
≤ 3.0 wt% U-235		None	None	2 @ 2 wt%	2 @ 2 wt%
≤ 2.9 wt% U-235		None	None	None	None
Polyethylene Equivalent Mass[d] (Maximum per Assembly)	kg	11	11	10.2	10.2

a. For 8 x 8 fuel assembly designs, there can be either 0 or 1 water rod; the water rod location occupies a space equivalent to 2 x 2 fuel rods. This is designated as 0, 2 x 2 in the table. For 9 x 9 and 10 x 10 fuel assembly designs, there can be either 0, 1, or 2 water rods in the assembly; the water rod location occupies a space equivalent to (i) two 2 x 2 fuel rod equivalent spaces on a diagonal at the center of the assembly, or (ii) one 3 x 3 fuel rod equivalent space (9 fuel rods space) in the center of the assembly. These configurations are designated as 0, 2 - 2x2 off-center diagonal, 3x3 in the table.

b. Transport with or without channels is acceptable

c. Required gadolinia rods must be distributed symmetrically along the major diagonal.

d. Polyethylene equivalent mass calculation per Section 6.3.2.2 of the application.

5.(b)(1) continued

Table 5: Fuel Assembly Parameters (11x11) for ≤ 5.0 weight percent U-235.

Parameter	Units	Value
Fuel Assembly Type	Rods	11x11
UO_2 Density[a]	g/cm^3	≤10.763
Number of water rods	#	3x3 center
Number of fuel rods	#	112
Fuel rod OD	cm	≥0.930
Fuel Pellet OD	cm	≤0.820
Cladding Type		Zirconium Alloy
Cladding ID	cm	≤0.840
Cladding Thickness	cm	≥0.045
Equivalent Nominal Fuel Rod Pitch	cm	≤1.195
U-235 Pellet Enrichment	wt%	≤5.0
Maximum Lattice Average Enrichment	wt%	≤5.0
Fuel Channel Side Thickness[b]	cm	≤0.320
Full Length Fuel Rods Quantity Active Length	 # cm	 92 ≤385
Short Part Length Fuel Rods Quantity Active Length	 # cm	 12 ≤155.1
Long Part Length Fuel Rods Quantity Active Length	 # cm	 8 ≤236.8
Gadolinia Requirements Lattice Average Enrichment[c] ≤ 5.0 wt% U-235 ≤ 4.8 wt% U-235 ≤ 4.6 wt% U-235 ≤ 4.4 wt% U-235 ≤ 4.2 wt% U-235 ≤ 4.1 wt% U-235 ≤ 3.9 wt% U-235 ≤ 3.8 wt% U-235 ≤ 3.6 wt% U-235 ≤ 3.5 wt% U-235 ≤ 3.3 wt% U-235 ≤ 3.2 wt% U-235 ≤ 2.9 wt% U-235	# @ wt% Gd_2O_3	 13 @ 2 wt% 12 @ 2 wt% 11 @ 2 wt% 10 @ 2 wt% 9 @ 2 wt% 8 @ 2 wt% 7 @ 2 wt% 6 @ 2 wt% 5 @ 2 wt% 4 @ 2 wt% 3 @ 2 wt% 2 @ 2 wt% None
Polyethylene Equivalent Mass (Maximum per Assembly)[d]	kg	10.2

a. Density based on a pellet modeled as a right circular cylinder

b. Transport with or without channels is acceptable.

c. Required gadolinia rods must be distributed symmetrically along the major diagonal and shall not be placed on the periphery.

d. Refer to Section 6.3.2.2 of the application.

5.(b)(1) continued

Table 6: Fuel Assembly Parameters (11x11) for > 5.0 to ≤ 8.0 weight percent U-235.

Parameter	Units	Value
Fuel Assembly Type	Rods	11x11
UO_2 Density[a]	g/cm^3	≤10.763
Number of water rods	#	3x3 center
Number of fuel rods	#	112
Fuel rod OD	cm	≥0.930
Fuel Pellet OD	cm	≤0.820
Cladding Type		Zirconium Alloy
Cladding ID	cm	≤0.840
Cladding Thickness	cm	≥0.045
Equivalent Nominal Fuel Rod Pitch	cm	≤1.195
U-235 Pellet Enrichment	wt%	≤8.0
Maximum Lattice Average Enrichment	wt%	≤8.0
Fuel Channel Side Thickness[b]	cm	≤0.320
Full Length Fuel Rods		
Quantity	#	92
Active Length	cm	≤385
Short Part Length Fuel Rods		
Quantity	#	12
Active Length	cm	≤155.1
Long Part Length Fuel Rods		
Quantity	#	8
Active Length	cm	≤236.8
Gadolinia Requirements Lattice Average Enrichment[c]		
≤ 8.0 wt% U-235 ≤ 7.5 wt% U-235 ≤ 7.0 wt% U-235 ≤ 6.5 wt% U-235 ≤ 6.1 wt% U-235 ≤ 5.8 wt% U-235	# @ wt% Gd_2O_3	21 @ 4 wt% 19 @ 4 wt% 17 @ 4 wt% 15 @ 4 wt% 13 @ 4 wt% 13 @ 2 wt%
Polyethylene Equivalent Mass (Maximum per Assembly)[d]	kg	10.2

a. Density based on a pellet modeled as a right circular cylinder
b. Transport with or without channels is acceptable.
c. Required gadolinia rods must be distributed symmetrically along the major diagonal and shall not be placed on the periphery.
d. Refer to Section 6.3.2.2 of the application.

5.(b)(1) continued

Table 7: Fuel Rod Parameters for ≤ 5.0 weight percent U-235

Parameter	Units	Type						
Fuel Assembly Type		8x8 (UO_2)	9x9 (UO_2)	10x10 (UO_2)	11x11 (UO_2)	CANDU-14 (UC)	CANDU-25 (UC)	Generic PWR (UO_2)
UO_2 or UC Fuel Density[a]	g/cm^3	≤10.74	≤10.74	≤10.74	≤10.763	≤13.36	≤13.36	≤10.74
Fuel rod OD	cm	≥1.10	≥1.02	≥1.00	≥0.930	≥1.340	≥0.996	≥1.118
Fuel Pellet OD	cm	≤1.05	≤0.96	≤0.90	≤0.820	≤1.254	≤0.950	≤0.98
Cladding Type		Zirc. Alloy	Zirc. Alloy	Zirc. Alloy	Zirc. Alloy	Zirc. Alloy or SS	Zirc. Alloy or SS	Zirc. Alloy or SS
Cladding ID	cm	≤1.10	≤1.02	≤1.00	≤0.930	≤1.267	≤0.951	≤1.004
Cladding Thickness	cm	≥0.038	≥0.036	≥0.038	≥0.045	≥0.033	≥0.033	≥0.033
Active fuel Length	cm	≤381	≤381	≤385	≤385	≤47.752	≤40.013	≤450
Maximum U-235 Pellet Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Maximum Average fuel rod Enrichment	wt.%	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0	≤5.0
Loose Rod Configuration								
Freely Loose or Strapped Together	#	≤25	≤25	≤25	≤25	N/A	N/A	N/A
Packed in 5" SS Pipe or protective Case, i.e., SS Box with Lid	#	≤22	≤26	≤30	≤30	≤74[b]	≤130[b]	≤105[b]

a. Density based on a pellet modeled as a right circular cylinder.
b. Including partial rods –using dense packing of congruent rods- in the 5" SS pipe

5.(b)(1) continued

Table 8: Fuel Rod Parameters for > 5.0 to ≤ 8.0 weight percent U-235.

Parameter	Units	Type	
Fuel Assembly Type		11x11 (UO_2)	17x17 PWR (UO_2)
UO_2 or UC Fuel Density[a]	g/cm^3	≤10.763	≤10.763
Fuel rod OD	cm	≥0.930	≥0.945
Fuel Pellet OD	cm	≤0.820	≤0.827
Cladding Type		Zirc. Alloy	Zirc. Alloy or SS
Cladding ID	cm	≤0.930	≤0.841
Cladding Thickness	cm	≥0.045	≥0.033
Active fuel Length	cm	≤385	≤381
Maximum U-235 Pellet Enrichment	wt.%	≤8.0	≤8.0
Maximum Average fuel rod Enrichment	wt.%	≤8.0	≤8.0
Loose Rod Configuration			
Freely Loose or Strapped Together	#	≤25	≤25
Packed in 5" SS Pipe or protective Case, i.e., SS Box with Lid	#	≤30	≤30

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

Total weight of payload contents (fuel assemblies, or fuel rods, and rod shipping containers) not to exceed 684 kg (1508 pounds).

(i) For the contents described in 5(b)(1)(i), 5(b)(1)(ii), 5(b)(1)(iii), 5(b)(1)(iv), and 5(b)(1)(v): two fuel assemblies.

(ii) For the contents described in 5(b)(1)(vi), 5(b)(1)(vii), and 5(b)(1)(viii): allowable number of fuel rods per compartment (2 compartments per package).

5.(c) Criticality Safety Index, for each content, as described in Table 9 below:

Table 9: Criticality Safety Index

Contents	Type and Limits	Enrichment wt.% U-235	CSI
5(b)(1)(i) 5(b)(1)(ii) 5(b)(1)(iii) 5(b)(1)(vi)	8x8, 9x9, 10x10 Fuel Assemblies 5(b)(2)(i), or BWR Rods 5(b)(2)(ii)	≤5.0	1.0
5(b)(1)(iv)	11x11 Fuel Assemblies 5(b)(2)(i)	≤5.0	1.5
5(b)(1)(v)	11x11 Fuel Assemblies 5(b)(2)(i)	≤8.0	3.2
5(b)(1)(vii)	Loose or Bundled or Rods in a Protective Carrier, 5(b)(2)(ii)	≤5.0	2.1
5(b)(1)(viii)	Loose or Bundled Rods, 11x11 or 17x17 PWR, 5(b)(2)(ii)	≤8.0	1.0
5(b)(1)(viii)	Rods in Protective Carrier, 11x11, 5(b)(2)(ii)	≤8.0	2.3
5(b)(1)(viii)	Rods in Protective Carrier, 17x17 PWR, 5(b)(2)(ii)	≤8.0	2.5

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

   (a) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

   (b) The packaging must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

   (c) Prior to each shipment, the fuel rods shall conform to the leak tests and specific inspection techniques used for qualification and in-process inspections as defined in Chapter 8. Stainless steel components of the packaging must be visually inspected. Packages in which stainless steel components show pitting corrosion, cracking, or pinholes, are not authorized for transport.

   (d) If wrapping is used on the unirradiated fuel assemblies, their ends must be assured to be open during transport.

7. All fuel to be shipped must meet the maximum P(r/t) criterion–product of the pre-pressure and of the maximum Inside Radius/Thickness- of 10.18653 MPa. Shipment of 11x11 fuel designs manufactured by other suppliers than Framatome is not authorized.

8. Cluster separators are optional and may be comprised of polyethylene or other plastics with mass limits determined in accordance with Section 6.3.2.2 (Material Specifications) of the application.

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

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

11. Expiration date: June 30, 2029.

<u>REFERENCES</u>

Framatome TN-B1 Safety Analysis Report, FS1-0014159, Revision 11, dated January 31, 2023.

Supplements dated October 9, 2023; March 22, 2024.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Yoira K. Diaz Sanabria

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

Date: June 6, 2024

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

SAFETY EVALUATION REPORT
Docket No. 71-9372
Model No. TN-B1
Certificate of Compliance No. 9372
Revision No. 5

SUMMARY

By letter dated January 31, 2023 (Agencywide Documents Access and Management System [ADAMS] Accession No: ML23031A213), Framatome Inc. (the applicant), requested an amendment to Certificate of Compliance (CoC) No. 9372, for the Model No. TN-B1 transportation package to support Framatome's advancement to enrichments greater than 5 weight percent (wt.%) ^{235}U.

On March 22, 2024, the applicant responded to staff's request for additional information dated November 1, 2023 (ML23321A179). The applicant provided updated calculations including the additional benchmarks establishing the upper safety limit (USL) as requested by the staff. The calculation determines the USL conservatively moves from 0.9318 to 0.9325. As a result, there are no changes to the gadolinia requirements or criticality safety index values previously presented in the application.

The certificate has also been renewed in accordance with the timely request for renewal letter dated December 14, 2023.

The NRC staff reviewed the applicant's request and found that the package meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

EVALUATION

The application, in its revision No. 11, includes the following changes:

(i) identification of two enrichment ranges (<5% and <8% wt.% ^{235}U) for the package contents and discussion of the new 17x17 Type 3 pressurized-water reactor (PWR) fuel rod criteria,
(ii) revision of drawings to include varying sizes and quantities of the vibro-isolator rubbers for an enhanced vibration performance,
(iii) updated leak rate information corresponding to the higher enrichment contents.

The fuel assemblies in this packaging, with two enrichment ranges up to a maximum of 8.0 wt.% ^{235}U, include both Type A and Type B materials. If the contents consist of Type B quantities of material, the main nuclides are shown in the Table below.

Type B Quantity of Radioactive Material (Type A and B) for ≤ 8.0 wt.% ^{235}U

Isotope	Maximum content[4]	Maximum mass, g	Specific Activity[5], TBq/g	Total Activity, TBq	Total Activity, Ci
U-232	$1.10\ 10^{-7}$g/gU	$1.10\ 10^{-7}$	0.83	$4.53\ 10^{-2}$	1.20
U-234	$7.65\ 10^{-3}$g/gU	$7.65\ 10^{-3}$	$2.3\ 10^{-4}$	$8.73\ 10^{-1}$	23.5
U-235	$8.00\ 10^{-2}$ g/gU	$8.00\ 10^{-2}$	$8.00\ 10^{-8}$	$3.17\ 10^{-3}$	$8.73\ 10^{-2}$
U-236	$2.50\ 10^{-2}$ g/gU	$2.50\ 10^{-2}$	$2.40\ 10^{-6}$	$2.98\ 10^{-2}$	$8.06\ 10^{-1}$
U-238	$8.87\ 10^{-1}$ g/gU	$8.87\ 10^{-1}$	$1.20\ 10^{-8}$	$5.28\ 10^{-3}$	$1.50\ 10^{-1}$
NP-237	$1.66\ 10^{-6}$ g/gU	$1.66\ 10^{-6}$	$2.60\ 10^{-5}$	$2.14\ 10^{-5}$	$5.85\ 10^{-4}$
PU-238	$6.20\ 10^{-11}$ g/gU	$6.20\ 10^{-11}$	$6.30\ 10^{-1}$	$1.94\ 10^{-5}$	$5.23\ 10^{-4}$
PU-239	$3.04\ 10^{-9}$ g/gU	$3.04\ 10^{-9}$	$2.30\ 10^{-3}$	$3.47\ 10^{-6}$	$9.35\ 10^{-5}$
PU-240	$3.04\ 10^{-9}$ g/gU	$3.04\ 10^{-9}$	$8.40\ 10^{-3}$	$1.27\ 10^{-5}$	$3.47\ 10^{-4}$
Gamma Emitters	$5.18\ 10^{+5}$ MeV-Bq/kgU	N/A	N/A	$2.57\ 10^{-2}$	$6.94\ 10^{-1}$
			Total	$2.82\ 10^{-1}$	26.5

Drawing No. 105E3739 Rev. 4 was replaced with Drawing No. FS1-0042699 Rev.1 to allow multiple rubber options with the goal of optimizing vibration reduction characteristics of the rubber material during transport. Drawing No. FS1-0042700 Rev. 2 was revised to include a new Note 4 regarding the optional vibro-isolator size and quantity to optimize vibration absorption options.

The full list of licensing drawings for this package is below:

Drawing number	Number of Sheets	Revision #	Name
105E3737	1	6	Outer/Inner Container Assembly Licensing Drawings
FS1-0042698	4	1	TN-B1 Outer Container Main Body Assembly Licensing Drawings
FS1-0042699	2	1	Outer Container Fixture Assembly Licensing Drawings
FS1-0042700	1	2	TN-B1 Outer Container Fixture Assembly Installation Licensing Drawings
105E3741	1	1	Outer Container Shock Absorber Assembly Licensing Drawings
105E3742	1	3	Outer Container Bolster Assembly Licensing Drawings

FS1-0042703	1	1	TN-B1 Outer Container Lid Assembly Licensing Drawings
02-9162717	1	1	Outer Container Marking Licensing Drawings

Inner Container Drawings

Drawing number	Number of Sheets	Revision #	Name
FS1-0042705	4	1	TN-B1 Inner Container Main Body Assembly Licensing Drawings
105E3746	1	1	Inner Container Parts Assembly Licensing Drawings
FS1-0042707	2	1	TN-B1 Inner Container Lid Assembly Licensing Drawings
FS1-0042708	1	1	TN-B1 Inner Container End Lid Assembly Licensing Drawings
02-9162722	1	1	Inner Container Marking Licensing Drawings

Contents Container Drawings

Drawing number	Number of Sheets	Revision #	Name
105E3773	1	1	Protective Case
0028B98	1	1	Shipping Container Loose Fuel Rods

The applicant added clarification wording to Section 6.0 and 6.1 of the application, along with adding new gadolinium oxide (Gd_2O_3) requirements for the added enrichment percentages between 5.0 and 8.0 weight percent (wt.%) uranium-235 (^{235}U) in Table 6-1. The applicant also added the new technical specifications of the PWR 17x17 Type 3 fuel assembly type and the loose rod limits packed in a 5-inch SS pipe or protective case within Table 6-2.

Table 6-3, the Critical Evaluation Summary, was amended by the applicant to include the normal conditions of transport (NCT) and hypothetical accident conditions (HAC) k-eff values, along with the number of packages used within the evaluation for the ATRIUM 11 assemblies, ATRIUM 11 rods, ATRIUM 10 rods, the PWR 17x17, type 3 rods, and the criticality safety index (CSI) values for the 11x11 assemblies, the 11x11 fuel rods contained in a 5-inch steel pipe, and the PWR 17x17 type 3 fuel rods all containing ≤ 8.0 wt.% ^{235}U enrichment.

The applicant revised the application to include a new Appendix 6.13, which evaluates the criticality safety for the package containing Atrium 11 fuel assemblies, ATRIUM 11 fuel rods, and individual PWR 17x17, type 3 fuel rods all with a maximum enrichment of 8.0 wt.% ^{235}U.

Also, the applicant included additional limits for transport of Atrium 10XM fuel rods with a maximum enrichment of 5.0 wt. % ^{235}U. The models' geometry was unchanged from previous versions of the application.

Previously the Gd_2O_3 wt.% minimum was 2.0 wt.%, and the scoping studies performed by the applicant supported this minimum Gd_2O_3 content up until a fuel enrichment of 5.0 wt.% ^{235}U. With the amendment request to increase fuel enrichment to 8.0 wt.% ^{235}U, the Gd_2O_3 wt % minimum was increased to 4.0 wt.% to maintain a maximum CSI of 3.3 for ATRIUM 11 assemblies.

A study was conducted by the applicant to determine the Gd Rod loading pattern which has the greatest reactivity, the results of which are shown in Figure 6-87 of the application. The applicant's loading study used the design requirements of the orientation such that the Gd rods must be placed symmetrically with respect to the assembly major diagonal and the Gd rods could not be placed on the periphery of the assembly. For enrichments above 5 wt.% ^{235}U, each face of the enriched lattices must contain a minimum of one equivalent Gd rod in the second and tenth row/column locations.

Using the Gd rod pattern, the most reactive fuel assembly orientation within the TN-B1 was determined by the applicant as shown in Figure 6-88 of the application. Studies for fuel channel position, polyethylene pad loss, preferential flooding of the package, and flooding between packages were performed by the applicant. A maximum of 10.2 kg of polyethylene per assembly can be contained within the TN-B1. When modeling, SCALE limitations dictate that the polyethylene be homogenized either with the clad or the water to perform resonance self-shielded multi-group cross-section calculations. The applicant determined that for the ATRIUM 11 fuel assemblies in an HAC array, the most reactive model is the polyethylene layered on clad and homogenized with the clad for determining cross sections.

Material compositions within the model were determined by the applicant using a previously approved process based on 98.2% of the UO_2 maximum theoretical density (MTD). For fuel rods containing Gd, the overall density of the fuel was calculated as a linear combination of the MTD UO_2 and Gd_2O_3 applying the 98.2% factor to UO_2. The applicant credits 75% of the Gd wt.%, therefore the models contain 1.5 wt.% or 3 wt.% of Gd_2O_3. The polyethylene layer thickness for the ATRIUM 11 fuel assembly model was determined by the applicant using the outer radius of the fuel rod clad, polyethylene density, and total mass of polyethylene. The applicant modeled the homogenized clad and polyethylene along with the homogenized water and polyethylene by determining volume fractions of each component and weighting the homogenized mixture by the volume fractions. A SCALE user notice [REF] identified an under-prediction of k-eff values in systems containing h1-poly. The applicant adjusted their k-eff values by adding a poly bias to consider the possible underprediction.

There are three transport cases for this package:

(i) containing ATRIUM 11 fuel assemblies,

(ii) containing loose fuel rods, and

(iii) containing loose fuel rods within a stainless-steel pipe.

For an ATRIUM 11 assembly single package, the applicant evaluated the package under NCT with water in-leakage between the TN-B1 inner and outer containers. The assembly contained the highest enrichment, 8 wt.% ^{235}U, and the lowest wt.% and number of Gd rods, 2 wt.% Gd_2O_3 and 13 rods. The most reactive orientation of the fuel channel and the location of the polyethylene was used. The resulting k-eff under NCT was 0.7514, under the applicant's calculated Upper Subcritical Limit (USL) of 0.9325.

For an ATRIUM 11 assembly single package, the applicant evaluated the package under HAC with full density water between the inner and outer container, the cushioning pads replaced with water, and the fuel channel location chosen for maximum reactivity. The assembly contained the highest enrichment, 8 wt.% ^{235}U, and the lowest wt. % and number of Gd rods, 2 wt.% Gd_2O_3 and 13 rods. The resulting k-eff under HAC was 0.85127, below the USL of 0.9325.

The staff reviewed the applicant's single package evaluations and finds that the applicant has demonstrated that a single package with water in-leakage is subcritical per 10 CFR 71.55(b), and that a single package is subcritical under NCT and HAC per 10 CFR 71.55(d) and (e), respectively.

For package arrays under HAC, the applicant modeled a finite 4x1x8 array of packages containing ATRIUM 11 fuel assemblies. The applicant found and used the most reactive water density and the most reactive preferential flooding pattern between each package. The preferential flooding pattern includes the internal compartment flooded with full density water and the volume between the inner and outer compartments empty. Instead of adjusting array size until the USL is approached, the applicant had a CSI goal of 3.2 and found the Gd rod limit for each enrichment between 5 and 8 wt.%. Each k-eff value includes bias and uncertainties. The results are summarized in Table 1 below.

Table 1. Enrichment and Minimum Gd rods for ATRIUM 11 HAC Array

wt.% ^{235}U	Minimum Gd Rods	Max k-eff	USL
6.3	13 @ 2 wt.%	0.9300	0.9325
6.7	15 @ 2 wt.%	0.9301	0.9325
6.6	13 @ 4 wt.%	0.9304	0.9325
7	15 @ 4 wt.%	0.9289	0.9325
7.6	17 @ 4 wt.%	0.9307	0.9325
8	19 @ 4 wt.%	0.9255	0.9325

The k-eff value for each combination of enrichment and the minimum number and absorber concentration of Gd rods is below the USL of 0.9325.

For package arrays under NCT, the applicant modeled a finite 9x1x9 array of packages containing ATRIUM 11 fuel assemblies. The model still used the same preferential flooding pattern within the packages, but differed by keeping the polyethylene foam pads intact, maintaining a larger outer container height, and reducing the fuel rod spacing to the undamaged value. The results are summarized in Table 2 below.

Table 2. Enrichment and Minimum Gd rods for ATRIUM 11 NCT Array

Wt.% ^{235}U	Minimum Gd Rods	Max k-eff	USL
5.8	13 @ 2 wt.%	0.9295	0.9325
6.1	13 @ 4 wt.%	0.9311	0.9325
6.5	15 @ 4 wt.%	0.9308	0.9325
7	17 @ 4 wt.%	0.9299	0.9325
7.5	19 @ 4 wt.%	0.9292	0.9325
8	21 @ 4 wt.%	0.9252	0.9325

The k-eff value for each combination of enrichment and the minimum number and absorber concentration of Gd rods is below the USL of 0.9325.

The second case the applicant modeled was for loose fuel rods in the TN-B1 package, including ATRIUM 11, ATRIUM 10XM, and PWR 17x17 type 3 fuel rods. Based on sensitivity studies done before, the most reactive configuration contained:

- TN-B1 central storage volumes flooded with full density water
- Volume between outer and inner TN-B1 containers empty
- Full thickness of polyethylene pads
- 25 full-length fuel rods in a 5x5 square-pitched array
- Fuel rod gap and clad modeled as full density water
- Evaluated with and without polyethylene sleeve to determine h-poly bias
- No credit for Gd_2O_3

The single package results are summarized in Table 3 below.

Table 3. Single Package Results for Loose Rods

Fuel Rod Design	HAC Max Keff	NCT Max Keff
ATRIUM 11	0.6832	0.6647
ATRIUM 10XM	0.6328	0.6181
PWR 17x17	0.6838	0.6656

The k-eff values for each single package fuel rod design under HAC and NCT are well below the USL of0.9325.

The staff reviewed the applicant’s single package evaluations for loose rods and finds that the applicant has demonstrated that a single package with water in-leakage is subcritical per 10 CFR 71.55(b), and that a single package is subcritical under normal conditions of transport and hypothetical accident conditions per 10 CFR 71.55(d) and (e), respectively.

Similarly to the ATRIUM 11 evaluation, the NCT array for loose fuel rods differed from the HAC array by increasing the height of the outer container according to the results of the HAC structural tests. For package arrays under NCT, the applicant modeled a finite 16x1x16 array, and for package arrays under HAC, the applicant modeled a finite 10x1x10 array. The maximum k-eff for each fuel rod type is summarized below.

Table 4. K-eff Values for Loose Rods NCT

Fuel Rod Design	Max Keff	USL
ATRIUM 11	0.8904	0.9325
ATRIUM 10XM	0.8255	0.9325
PWR 17x17	0.8937	0.9325

Table 5. K-eff Values for Loose Rods HAC

Fuel Rod Design	Max Keff	USL
ATRIUM 11	0.8652	0.9325
ATRIUM 10XM	0.8024	0.9325
PWR 17x17	0.8681	0.9325

For the fuel rod designs above, the maximum k-eff values are below the USL of each system. The applicant determined the CSI for each of the above contents of the TN-B1 package according to the requirements for package arrays in 10 CFR 71.59. For NCT, the applicant showed that an array of 256 packages is subcritical. For HAC, the applicant showed that an array of 100 packages is subcritical. Using the most limiting case, the applicant determined the CSI to be 1.0.

The staff finds that the applicant has appropriately determined the package CSI in accordance with the requirements of 10 CFR 71.59.

The third case the applicant modeled was for ATRIUM 11 rods (≤ 8.0 wt.% ^{235}U), ATRIUM 10XM rods (≤ 5.0 wt.% ^{235}U), and PWR 17x17, type 3 rods (≤ 8.0 wt.% ^{235}U) contained in either a 5-inch schedule 40 stainless steel pipe or a protective carrier. Both the pipe and the protective carrier have been approved in previous amendments. The applicant conducted sensitivity studies to determine the most reactive configuration, which includes:

- Optimum triangular pitch for the fuel rods within the pipe.
- Optimum water density for water that is inside the inner package and outside the pipe.
- Optimal model for polyethylene sleeves.

The applicant's HAC model determined the optimal peripheral pad thickness and determined the optimal pipe locations within the internal transport volume. The most reactive pin pitch was determined to be quite large, and because the protective case's cross-sectional area is much smaller than the stainless-steel pipe's cross-sectional area, the transport of loose rods is bounded by the stainless-steel pipe geometry. Similar to the loose fuel rods case, the NCT model differed by increasing the height of the outer container and keeping full thickness of the peripheral cushioning pads.

The results of the applicant's HAC and NCT single package evaluations are summarized below. The polyethylene error was added when applicable, along with the calculation's biases and uncertainties.

Table 6. Loose Rods in Pipe, HAC and NCT Single Package Results

Fuel Rod Design	HAC Single Package Keff	NCT Single Package Keff
ATRIUM 11	0.7630	0.7014
ATRIUM 10XM	0.7114	0.6531
PWR 17x17	0.7655	0.7043

All maximum k-eff values are well below the USL of 0. 9325.The staff reviewed the applicant's single package evaluations for loose rods and finds that the applicant has demonstrated that a single package with water in-leakage is subcritical per 10 CFR 71.55(b), and that a single package is subcritical under normal conditions of transport and hypothetical accident conditions per 10 CFR 71.55(d) and (e), respectively.

To determine CSI values, the applicant modeled arrays of the packages under HAC and NCT. The models and their subsequent k-eff values and CSI values are summarized below.

Table 7. NCT and HAC CSI Calculations for Loose Rods in Pipes

Fuel Rod Design	HAC Array	Keff	CSI_{HAC}		NCT Array	Keff	CSI_{NCT}
ATRIUM 11	9x1x9	0.9240	1.3		10x1x11	0.9285	2.3
ATRIUM 10XM	10x1x10	0.8690	1.0		11x2x12	0.8978	1.0
PWR 17x17	9x1x9	0.9266	1.3		10x1x10	0.9241	2.5

Taking the most limiting CSI of each case, the applicant determined that for the TN-B1 loaded with loose fuel rods in a pipe, the CSIs for ATRIUM 11, ATRIUM 10XM, and the PWR 17x17 fuel rods are 2.3, 1.0, and 2.5 respectively. The staff finds that the applicant has appropriately determined the package CSI in accordance with the requirements of 10 CFR 71.59.

The staff reviewed the configurations modeled by the applicant for the single package and array analyses. The staff finds with reasonable assurance that the applicant has identified the most reactive credible condition of the single package and arrays of packages, consistent with the condition of the package under NCT and HAC, and the chemical and physical form of the fissile and moderating contents.

The applicant selected applicable benchmark experiments to validate the ATRIUM 11 contents enriched up to 8.0 wt.% and loose PWR rod contents enriched up to 8.0 wt.% using sensitivity/uncertainty analysis methods (S/U).

The applicant used the TSUNAMI-3D sequence included in the SCALE 6.2.4 code package to calculate the sensitivity of the k-eff value from the TN-B1 package bounding array case (known as the application model) for both the ATRIUM 11 and loose PWR rods to variations of the nuclear data used in the k-eff calculation. The TSUNAMI-3D sequence generates sensitivity data files (SDFs) containing sensitivities of k-eff to variations in cross section data for each reaction type. The applicant then used TSUNAMI-IP sequence to compare the application SDF against potential critical benchmark experiment SDFs. The TSUNAMI-IP sequence generates correlation coefficients (c_k values) that indicate similarity between the application model and the critical benchmark experiment. The applicant only used this S/U method to select applicable benchmarks for validation.

When using TSUNAMI-3D it is important to ensure that the model is divided into an appropriate grid size to determine the sensitivities, and determining the right grid size can be difficult. The need for finer grids for fissile material regions must be balanced against the large computer memory requirement and longer computation time for a smaller grid size.

The applicant calculated sensitivities for both ^{235}U and ^{1}H using direct perturbations and compared them to the sensitives generated using TSUNAMI-3D.

> For the package containing the ATRIUM 11 assembly, sensitivity coefficients using direct perturbation were 0.201 (^{235}U) and 0.054 (^{1}H) and using TSUNAMI-3D were 0.182 (^{235}U) and 0.062 (^{1}H).
>
> For the package containing loose PWR rods, sensitivity coefficients using direct perturbation were 0.220 (^{235}U) and 0.258 (^{1}H) and using TSUNAMI-3D were 0.209 (^{235}U) and 0.207 (^{1}H).

The staff reviewed the applicant's use of S/U methods to ensure they were applied appropriately.

The experiments selected by the applicant for validation can be found in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (ICSBEP Handbook). For the package containing the ATRIUM 11 assembly, the applicant selected a total of 254 experiments with a minimum c_k value of 0.80, of which 27 had c_k values > 0.90.

For the package containing loose PWR rods, the applicant selected a total of 241 experiments with a minimum c_k value of 0.80 of which 96 had c_k values > 0.90. Table 6-155 of Appendix 6.13.10.1 of the application includes comparisons between the critical benchmark experiments and the TN-B1 package with the ATRIUM 11 assemblies, demonstrating the applicability of the included benchmarks. Table 6-156 of Appendix 6.13.10.1 of the application includes comparisons between the critical benchmarks experiments and the TN-B1 package with loose PWR rods demonstrating the applicability of the included benchmarks. Many methods used to calculate bias and bias uncertainty (used to determine the USL) rely on the assumption that the population of critical experiments constitutes a normal distribution. Since the data set does not follow a normal distribution, the applicant applied a non-parametric technique that uses an analysis of ranks to determine the USL.

> For the ATRIUM 11 assembly sample population size, the rank index for a one-sided distribution-free tolerance limit with 95% confidence that 95% of the population is covered is 7, meaning the seventh lowest calculated k-eff value, which for the population of experiments selected by the applicant is 0.98644. Including an administrative margin of 0.05 yields a USL of 0.9325.
>
> For the loose PWR rod sample population size, the rank index for a one-sided distribution-free tolerance limit with 95% confidence that 95% of the population is

covered is also 7, The seventh lowest calculated k-eff value is also 0.98644, obtaining the same USL of 0. 9325.

The staff finds that the applicant determined an appropriate USL using S/U methods. Staff S/U analysis and USL determination confirms that the applicant's USL is conservative.

The staff performed confirmatory calculations using the SCALE 6.2.4 Monte Carlo radiation transport code, with the CSAS6 criticality sequence and the continuous energy ENDF/B-VII neutron cross section library. The staff's confirmatory analyses focused on the most reactive configuration of single packages and arrays of packages. Using modeling assumptions similar to the applicant's, the staff's evaluation resulted in keff values that were similar to, or bounded by, the applicant's results.

The staff contracted with Oak Ridge National Laboratory (ORNL) to perform confirmatory benchmarking calculations for validation purposes. ORNL used TSUNAMI-3D to generate SDFs from both the ICSBEP Handbook and from the ORNL SCALE Verified, Archived Library of Inputs and Data (VALID), and TSUNAMI-IP to generate c_k values between the application model and critical experiments. From the VALID library, ORNL found 69 experiments with c_k values > 0.8 for the ATRIUM 11 assembly model, and from ICSBEP Handbook, 187 experiments for the ATRIUM 11 assembly and 154 for loose PWR rods with c_k values > 0.8. Neither experiment set had a normalized distribution and a non-parametric technique as described by the applicant was used by ORNL to determine the USL.

For the ATRIUM 11 assembly, ORNL determined USLs of 0.94340 (VALID) and 0.93242 (ICSBEP), both including the administrative margin of 0.05.

For the loose PWR rods, ORNL determined USLs of 0.94340 (VALID) and 0.93460 (ICSBEP), both including the administrative margin of 0.05.

The USLs generated by ORNL are similar to or greater than the applicant's determined USLs of 0.9325 (ATRIUM 11 and loose PWR rods) which demonstrates that the applicant's USL is appropriate.

Based on the discussion above, the staff found the changes to the CoC would not affect the ability of the TN-B1 package to meet the criticality safety requirements of 10 CFR Part 71.

CONDITIONS

The following changes have been made to the CoC:

Item No. 3.b. has been updated to include application FS1-0014159, Revision 11, as supplemented on March 22, 2024.

Condition No. 5(a)(2) was revised to include the maximum enrichment of 8 wt.%.

Condition No. 5(a)(3) was revised due to drawings being replaced. A new drawing FS1-0042699-1.0 allows multiple rubber options and drawing FS1-0042700-2.0 was revised.

Condition No. 5(b(1)(ii) was added to include contents > 5.0 to ≤ 8.0 wt. % U-235. A new Table 3 shows the maximum concentrations of authorized contamination material for > 5.0 to ≤ 8.0 weight percent U-235.

Conditions No. 5(b)(1)(v) and (viii) were added for 11x11 fuel assemblies and fuel rods respectively. Tables 6 and 8 are new and give the fuel assemblies and fuel rods parameters, respectively, for > 5.0 to ≤ 8.0 weight percent U-235.

Condition No. 5(b)(2) has been revised.

Condition No. 5(c) has been entirely revised.

Condition No. 11 has been revised to extend the validity of the certificate to June 30, 2029.

The references section has been updated to include the March 22, 2024, supplement to the application.

CONCLUSION

Based on the statements contained in the application, and the conditions listed above, the staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9372, Revision No. 5.