# CERTIFICATE OF COMPLIANCE
## FOR RADIOACTIVE MATERIAL PACKAGES

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9358 | 5 | 71-9358 | USA/9358/B(U)F-96 | 1 | OF | 34 |

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)*

TN Americas LLC
7160 Riverwood Drive, Suite 200
Columbia, MD 21046

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

TN-LC Transportation Package Safety Analysis Report, Revision No. 9.

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-LC

(2) Description

The packaging, designed for transport of irradiated test, research, and commercial reactor fuel in either a closed transport vehicle or an ISO container, consists of a payload basket, a shielded body, a shielded closure lid and top and bottom impact limiters. The packaging body is a right circular cylinder, approximately 197.5 inches long and 30 inches in diameter, composed of top and bottom end flange forgings connected by inner and outer shells. Lead shielding, made of ASTM B29 copper lead, is placed between the two cylindrical shells, in the bottom end assembly, and in the lid. Neutron shielding, composed of a borated resin compound inserted into twenty aluminum shield boxes, is set between the outer shell and a 0.25 inch-thick Type 304 stainless steel outer sheet. Two removable trunnions are bolted to the packaging body using eight 1-8UNC bolts for each trunnion. Two pocket trunnions in the bottom flange, used for rotating the package, may also be used for horizontal package lifting. Impact limiters, with an approximate outside diameter of 66 inches and height of 22.75 inches, consisting of balsa and redwood blocks encased in stainless steel shells, are attached to each end of the packaging during shipment, each with eight 1-8UNC bolts.

Four basket designs are provided for transport of Boiling Water Reactor (BWR), Pressurized Water Reactor (PWR), Mixed Oxide Fuel (MOX), Evolutionary Pressurized Reactor (EPR), National Research Universal Reactor (NRU), National Research Experimental

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

Reactor (NRX), Material Test Reactor (MTR), and Training, Research, and Isotope General Atomics Reactor (TRIGA) fuel assemblies, fuel elements or fuel rods. The packaging may be loaded or unloaded either in a pool or a hot cell environment. The spent fuel payload is shipped dry in a helium atmosphere. The first fabricated packaging, Unit 1, shall only be loaded with the TN-LC-1FA basket.

Nominal weights and dimensions are as follows:

| | |
|---|---|
| - Overall length with impact limiters: | 230 inches |
| - Overall length without impact limiters: | 197.50 inches |
| - Cavity length (minimum): | 182.50 inches |
| | 182.10 inches for Unit 1 |
| - Cavity inner diameter: | 18 inches |
| - Lid thickness: | 7.50 inches |
| - Weight of contents: | 7,100 lbs |
| - Weight of lid: | 1,000 lbs |
| - Weight of impact limiters: | 3,000 lbs |
| - Total loaded weight of the package: | 51,000 lbs |

(3) Drawings

The packaging is constructed and assembled in accordance with the following drawings:

| | |
|---|---|
| 65200-71-01 Revision 9 | TN-LC Cask Assembly (11 sheets) |
| 65200-71-02 Revision 0 | TN-LC Transport Cask Regulatory Plate (1 sheet) |
| 65200-71-20 Revision 5 | TN-LC Impact Limiter Assembly (2 sheets) |
| 5200-71-21 Revision 2 | TN-LC Transport Packaging Transport Configuration (1 sheet) |
| 65200-71-40 Revision 4 | TN-LC-NRUX Basket Basket Assembly (5 sheets) |
| 65200-71-50 Revision 4 | TN-LC-NRUX Basket Basket Tube Assembly (5 sheets) |

| | |
|---|---|
| 65200-71-60 Revision 4 | TN-LC-MTR Basket<br>General Assembly (4 sheets) |
| 65200-71-70 Revision 4 | TN-LC-MTR Basket<br>Fuel Bucket (2 sheets) |
| 65200-71-80 Revision 4 | TN-LC-TRIGA Basket (5 sheets) |
| 65200-71-90 Revision 6 | TN-LC-1FA Basket (5 sheets) |
| 65200-71-96 Revision 5 | TN-LC-1FA BWR<br>Sleeve and Hold-Down Ring (2 sheets) |
| 65200-71-102 Revision 7 | TN-LC-1FA Pin Can Basket (5 sheets) |

5.(b) Contents

(1) Type and Form of Material

(i) Intact or damaged NRU and NRX Mk I fuel assemblies which meet the specifications listed in Table 1 below, respectively, are authorized for transportation in the TN-LC-NRUX basket.

Intact fuel assemblies are fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

Damaged fuel assemblies, with cladding damage in excess of pin hole leaks or hairline cracks, are authorized only if the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod.

5.(b)(1) Type and Form of Materials (continued)

Table 1

NRU and NRX Mk I Fuel Specifications for Transport in the TN-LC-NRUX Basket

| Parameter | NRU | NRX Mk I |
|---|---|---|
| Physical and Material Description | | |
| Number of Assemblies | ≤ 26 | ≤ 26 |
| Number of rods/assembly | ≤ 12 | 7 |
| Assembly length (inch) (1) | ≤ 116 | ≤ 116 |
| Nominal Assembly mass (g) | 4660 | 5780 |
| Fuel form | U-Al | U-Al |
| $^{235}U$ per rod (g) | ≤ 45.4 | ≤ 75.2 |
| Enrichment (wt.% $^{235}U$) | ≤ 93 | ≤ 93 |
| Cladding and Spacer Material | Al | Al |
| Thermal and Radiological Parameters | | |
| Cooling Time (years) (2) | ≥ 10 | ≥ 10 |
| Depletion (wt.% $^{235}U$) (3) | ≤ 80 | ≤ 80 |
| Decay Heat per Assembly (watts) (4) | ≤ 15 | ≤ 15 |

Notes:
1. Maximum length of the fuel assembly (unirradiated) for shipment.
2. The cooling time of the fuel assembly rounded down to 0.5 years.
3. The depletion (or burnup) of the fuel assembly rounded up to 0.5%.
4. The decay heat of the fuel assembly is less than 15 watts at the maximum burnup and minimum cooling time.

(ii) Intact or damaged MTR fuel elements that are enveloped or bounded by the fuel element design characteristics listed in Table 2 below, with an average burnup and minimum cooling time as specified in Table 3 below, and a maximum decay heat of 25 watts per element, are authorized for transportation in the TN-LC-MTR basket.

Intact fuel elements are fuel elements containing fuel plates with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

Damaged fuel elements, with cladding damage in excess of pin hole leaks or hairline cracks, are authorized only if the total surface area of the damaged cladding does not exceed 5% of the total surface area of each element.

The MTR fuel assemblies shall meet all the requirements in Table 3.

5.(b)(1) Type and Form of Materials (continued)

Table 2

MTR Fuel Element Design Characteristics

| Fuel Element Class | M-01 | M-02 | M-03 | M-04 | M-05 | M-06 | M-07 | M-08 (1) |
|---|---|---|---|---|---|---|---|---|
| Number of Fuel Plates (2) | ≤23 | ≤21 | ≤19 | ≤17 | ≤10 | ≤18 | ≤17 | ≤23 |
| $^{235}$U mass per Plate (g) | ≤16 | ≤16.5 | ≤17.5 | ≤19 | ≤22 | ≤20.5 | ≤11.5 | ≤22 |
| Active Fuel Width (cm) | ≤6.7 | ≤6.7 | ≤6.7 | ≤6.7 | ≤6.7 | ≤5.9 | ≤6.7 | ≤6.7 |
| Active Fuel Length (cm) | ≥ 56 | ≥ 56 | ≥ 56 | ≥ 56 | ≥ 56 | ≥ 56 | ≥ 27.5 | ≥ 56 |
| Enrichment (wt.% $^{235}$U) | ≤ 94 | ≤ 94 | ≤ 94 | ≤ 94 | ≤ 94 | ≤ 94 | ≤ 94 | ≤ 94 |
| Fuel Element Depth (cm) | ≥7.5 | ≥7.5 | ≥7.5 | ≥7.5 | ≥7.5 | ≥7.5 | ≥7.5 | ≥7.5 |

Notes:
1. The M-08 Element class requires that the central stack of fuel elements remain empty. Also, the total $^{235}$U mass is limited by the maximum value in Table 3.
2. The plate thickness is greater than 0.12 cm and the clad thickness is greater than 0.02 cm.

Table 3

MTR Fuel Element Qualification

| Enrichment Type | Burnup (MWd/MTU) | Cooling Time (days) |
|---|---|---|
| Type A<br>$^{235}$U Enrichment ≥ 90%<br>$^{235}$U Mass ≤ 380 g | 66,000 | 740 |
| | 165,000 | 1120 |
| | 330,000 | 1440 |
| | 495,000 | 1680 |
| | 660,000 | 1950 |
| Type B<br>$^{235}$U Enrichment ≥ 90%<br>380 g < $^{235}$U Mass ≤ 460 g | 57,750 | 770 |
| | 144,375 | 1150 |
| | 288,750 | 1470 |
| | 433,125 | 1710 |
| | 577,500 | 1950 |
| Type C<br>40% ≤ $^{235}$U Enrichment < 90%<br>$^{235}$U Mass ≤ 380 g | 29,330 | 740 |
| | 73,325 | 1120 |
| | 146,650 | 1440 |
| | 219,975 | 1690 |
| | 293,300 | 1940 |
| Type D<br>19% ≤ $^{235}$U Enrichment < 40%<br>$^{235}$U Mass ≤ 470 g | 13,930 | 830 |
| | 34,825 | 1220 |
| | 69,650 | 1560 |
| | 104,475 | 1850 |
| | 139,300 | 2150 |

5.(b)(1) Type and Form of Materials (continued)

Notes

- Burnup = fuel element average burnup.
- Use burnup (MWd/MTU) and Enrichment Type (A, B, C, or D with limits on $^{235}U$ enrichment and $^{235}U$ mass per element) to look up minimum cooling time in days. Licensee is responsible for ensuring that uncertainties in burnup, enrichment, and mass are applied conservatively.
- Fuel with burnups greater than those listed for each Enrichment Type is not authorized for transport.
- Burnups may be either rounded up to the next higher burnup or linear interpolation may be used to determine the minimum cooling time. However, for conservatism, an additional cooling time of 30 days must be added to any linearly interpolated value.
- Example: An M-06 class element with an enrichment of 45 wt.% $^{235}U$ and a $^{235}U$ mass of 350 grams is classified as enrichment Type C. A burnup of 100,000 MWd/MTU is acceptable for transport after 1440 days cooling time as defined by 146,650 MWd/MTU from the qualification table (when linear interpolation is not employed). When linear interpolation is employed the minimum required cooling time is 1267 days (1237 days based on interpolation + 30 days additional cooling time).

(iii) Intact TRIGA fuel assemblies/elements that are enveloped by the fuel assemblies/element design characteristics listed in Table 4, intact TRIGA fuel follower control rods that are enveloped by the fuel assembly/element design characteristics listed in Table 5, with an average burnup and minimum cooling time meeting the specifications of Table 6 for fuel assemblies/elements or of Table 7 for follower control rods, and a maximum decay heat of 8 watts per assembly/element, are authorized for shipment with the TN-LC-TRIGA basket.

Intact fuel assemblies/elements are fuel assemblies/elements containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks. The design characteristics of the TRIGA fuel assemblies/elements are described in Tables 4 and 5 below.

The fuel qualification Tables 6 and 7 specify the maximum assembly/element average burnup and minimum cooling time. The fuel elements/assemblies shall meet all the requirements of Tables 6 and 7.

The poison plates in TN-LC-TRIGA basket are constructed from either boron aluminum alloy, or metal matrix composite (MMC), or Boral®. The minimum areal density of Boron-10 ($^{10}B$) for either the boron enriched aluminum alloy or the metal matrix composite is 5.56 mg/cm$^2$. The minimum areal density of $^{10}B$ for Boral® is 6.67 mg/cm$^2$.

5.(b)(1) Type and Form of Materials (continued)

Table 4

TRIGA Fuel Assembly/Element Design Characteristics

| Assembly/Element Type | Al Clad | ACPR (1) | Standard | FLIP (2) | FLIP (2) LEU-I (3) | FLIP (2) LEU-II (3) |
|---|---|---|---|---|---|---|
| Element ID | T-01 | T-02 | T-03 | T-04 | T-05 | T-06 |
| Fuel Material | U-ZrH | U-ZrH | U-ZrH | U-ZrH | U-ZrH | U-ZrH |
| Enrichment (wt.% $^{235}$U) | ≤ 20 | ≤ 20 | ≤ 20 | ≤ 70 | ≤ 20 | ≤ 20 |
| $^{235}$U-Mass (g) | ≤ 41 | ≤ 56 | ≤ 41 | ≤ 137 | ≤ 101 | ≤ 169 |
| Active Fuel Length (inch) | ≤ 15 | ≤ 15 | ≤ 15 | ≤ 15 | ≤ 15 | ≤ 15 |
| Pellet Diameter (inch) | ≤ 1.41 | ≤ 1.41 | ≤ 1.44 | ≤ 1.44 | ≤ 1.44 | ≤ 1.44 |
| Clad Material | Al | SS304 | SS304 | SS304 | SS304 | SS304 |
| H/Zr, max. | 1.0 | 1.7 | 1.7 | 1.6 | 1.6 | 1.6 |

Table 5

TRIGA Fuel Follower Control Rods Design Characteristics

| Assembly/Element Type | Standard | FLIP (2) LEU-I (3) | ACPR (1) |
|---|---|---|---|
| Element ID | T-07 | T-08 | T-09 |
| Fuel Material | U-ZrH | U-ZrH | U-ZrH |
| Enrichment (wt. % $^{235}$U) | ≤ 20 | ≤ 20 | ≤ 20 |
| $^{235}$U-Mass (g) | ≤ 38 | ≤ 97 | ≤ 56 |
| Active Fuel Length (inch) | ≤ 15 | ≤ 15 | ≤ 15 |
| Pellet Diameter (inch) | ≤ 1.32 | ≤ 1.32 | ≤ 1.32 |
| Clad Material | SS304 | SS304 | SS304 |
| H/Zr, max. | 1.7 | 1.6 | 1.7 |

Notes:
1. ACPR - Annular Core Pulse Reactor
2. FLIP - Fuel Life Improvement Program
3. LEU - Low Enriched Uranium

5.(b)(1) Type and Form of Materials (continued)

Table 6

TRIGA Fuel Qualification for Fuel Assembly/Elements

| Element ID | Burnup (MWd/MTU) | Cooling Time (days) |
|---|---|---|
| T-01 | 35,750 | 400 |
| | 71,500 | 560 |
| | 107,250 | 640 |
| | 143,000 | 710 |
| T-02 | 35,750 | 650 |
| | 71,500 | 970 |
| | 107,250 | 1310 |
| | 143,000 | 1870 |
| T-03 | 35,750 | 520 |
| | 71,500 | 840 |
| | 107,250 | 1170 |
| | 143,000 | 1730 |
| T-04 | 112,500 | 1000 |
| | 225,000 | 1380 |
| | 337,500 | 1820 |
| | 450,000 | 2520 |
| T-05 | 35,750 | 920 |
| | 71,500 | 1290 |
| | 107,250 | 1710 |
| | 143,000 | 2360 |
| T-06 | 36,500 | 1190 |
| | 73,000 | 1690 |
| | 109,500 | 2320 |
| | 146,000 | 3170 |

5.(b)(1) Type and Form of Materials (continued)

Table 7

TRIGA Fuel Qualification for Fuel Follower Control Rods

| Element ID | Burnup (MWd/MTU) | Cooling Time (days) |
|---|---|---|
| T-07 | 35,750 | 540 |
| | 71,500 | 890 |
| | 107,250 | 1280 |
| | 143,000 | 1960 |
| T-08 | 35,750 | 940 |
| | 71,500 | 1350 |
| | 107,250 | 1840 |
| | 143,000 | 2580 |
| T-09 | 35,750 | 670 |
| | 71,500 | 1020 |
| | 107,250 | 1420 |
| | 143,000 | 2100 |

Notes for Tables 6 and 7:

- Burnup = fuel element / assembly / follower control rod average burnup.
- Use burnup (MWd/MTU) and Element ID to look-up minimum cooling time in days. Licensee is responsible for ensuring that uncertainties in burnup are applied conservatively.
- Fuel with a burnup greater than that listed for each element type in Tables 6 and 7 is unacceptable for transport.
- Burnups may be either rounded up to the next higher burnup or linear interpolation may be used to determine the minimum cooling time. However, for conservatism, an additional cooling time of 30 days must be added to any linearly interpolated value.
- Example: A T-03 element with a burnup of 100,000 MWd/MTU is acceptable for transport after 1170 days cooling time as defined by 107,250 MWd/MTU (Table 6, rounding up) on the qualification table (when linear interpolation is not employed). When linear interpolation is employed the minimum required cooling time is 1133 days (1103 days based on interpolation + 30 days additional cooling time).

(iv) Intact PWR fuel assembly, as specified in Table 8, or intact BWR fuel assembly, as specified in Table 13, or fuel rods in a pin can are authorized for transport with the TN-LC-1FA basket.

Intact fuel assemblies are fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks.

5.(b)(1) Type and Form of Materials (continued)

The fuel rods include irradiated PWR, BWR, MOX, and EPR fuel rods. PWR intact and BWR intact fuel rods may be from any of the fuel assemblies listed in Table 8 or Table 13, respectively.

MOX rods have the same geometry as PWR or BWR rods, as defined in Table 8 and Table 13. The composition of MOX fuel is specified in Table 12.

The EPR fuel rods are specified in Tables 8 and 10.

The poison plates in the TN-LC-1FA basket are constructed from boron aluminum alloy, or metal matrix composite (MMC), or Boral®. The minimum $^{10}B$ areal density of the poison plate is 16.7 mg/cm$^2$ for either the boron aluminum alloy or the MMC. The minimum $^{10}B$ areal density of the poison plate is 20.0 mg/cm$^2$ for Boral®.

In addition to the poison plates provided in the basket, Poison Rod Assemblies (PRAs) are required for transportation of PWR fuel assemblies. The minimum required $B_4C$ content of the absorber rods in the PRA is 40% Theoretical Density (TD). A summary of the number of absorber rods required in the PRA for each PWR fuel class is shown in Table 11. PRA loading configurations are also illustrated in Figure 1 through Figure 5.

The PWR fuel assemblies fuel qualification table (FQT) is provided in Tables 15 and 15a for Unit 1 packaging.

The BWR fuel assemblies FQT is provided in Table 16.

The PWR rod FQTs are shown in Table 17 and Table 18 for the 21 and 9 rod configurations, respectively, and in Table 17a for the Unit 1 packaging.

The BWR rod FQTs are shown in Table 19 and Table 20 for the 21 and 9 rod configurations, respectively.

The MOX rod FQT, provided in Table 21 for both 21 and 9 rods, is applicable to both BWR and PWR MOX rods.

The FQTs for the $UO_2$ Standard EPR rods are governed by the PWR rod FQTs (Tables 17 and 18), while the FQT for the MOX EPR rods is governed by the MOX rod FQT (Table 21).

5.(b)(1) Type and Form of Materials (continued)

Table 8

PWR Fuel Specifications for Transport in the TN-LC-1FA Basket

| | |
|---|---|
| Fuel Class (1) (2) | One intact unconsolidated B&W 17x17, WE 17x17, CE 16x16, B&W 15x15, WE 15x15, CE 15x15, WE 14x14, WE 16x16 or CE 14x14 class PWR assembly (without control components) that are enveloped by the fuel assembly design characteristics listed in Table 9. Reload fuel manufactured by the same or other vendors, but enveloped by the design characteristics listed in Table 9, is also acceptable. |
| Maximum Assembly + PRA Weight | 1850 lbs |
| Fissile Material | $UO_2$ |
| Maximum Initial Uranium Content(4) | 490 kg/assembly |
| Maximum Unirradiated Assembly Length | 178.3 inches |
| Fuel Assembly Average Burnup, Enrichment and Minimum Cooling Time | Per Tables 15 and 15a |
| Maximum Planar Initial Enrichment | 5.0(3) wt.% $^{235}U$ |
| Maximum Decay Heat(5) | 3.0 kW per Assembly |
| Minimum $^{10}B$ areal density in poison plates | • 16.7 mg/cm$^2$ (Natural or Enriched Boron Aluminum Alloy / Metal Matrix Composite (MMC))<br>• 20.0 mg/cm$^2$ (Boral®) |
| Minimum number of absorber rods per PRA as a function of assembly class | Per Table 11 |

Notes:

1. Up to 21 PWR fuel rods, from any of the PWR fuel assemblies listed in Table 9, with a maximum peak burnup of 90 GWd/MTU, may be transported in the TN-LC-1FA basket in a pin can, with a cavity length of 169.55 inches (Option 3), that is placed within the BWR sleeve and hold-down ring and within the TN-LC-1FA basket. The required cooling time, function of a PWR fuel rod burnup and enrichment, is provided in Table 17 and 17a (Unit 1 packaging) for up to 21 rods, and Table 18 for up to 9 rods (not authorized in Unit 1), respectively.
2. Up to 21 EPR fuel rods from any of the fuel classes listed in Table 9, meeting EPR rod parameters specified in Table 10, may be loaded in a pin can with a cavity length of 180.24 inches (Options 1 and 2), which is placed within the BWR sleeve and hold-down ring and within the TN-LC-1FA basket. The maximum peak burnup for the fuel rods is 90 GWd/MTU. The required cooling time, function of an EPR fuel rod burnup and enrichment, is provided in Table 17 for 21 rods and Table 18 for up to 9 rods. EPR rods are not authorized in Unit 1 packaging.
3. For CE 15x15, the maximum planar average initial enrichment is 3.60 wt.% $^{235}U$.
4. The maximum initial uranium content is based on the shielding analysis. The listed value is higher than the actual.
5. The maximum decay heat per rod is 220 watts when loading up to 9 rods. The maximum decay heat per rod is 120 watts when loading 10 or more (up to 21) rods.

5.(b)(1) Type and Form of Materials (continued)

Table 9
PWR Fuel Assembly Design Characteristics for Transportation in the TN-LC-1FA Basket

| Assembly Class | B&W 15x15 | B&W 17x17 | WE 17x17 | CE 15x15 | WE 15x15 | CE 14x14 | WE 14x14 | CE 16x16 | WE 16x16 |
|---|---|---|---|---|---|---|---|---|---|
| Maximum Number of Fuel Rods | 208 | 264 | 264 | 216 | 204 | 176 | 179 | 236 | 235 |
| Maximum Number of Guide/Instrument Tubes | 17 | 25 | 25 | 9 | 21 | 5 | 17 | 5 | 21 |
| Rod Pitch(1) (inch) | ≤ 0.568 | ≤ 0.502 | ≤ 0.496 | ≤ 0.550 | ≤ 0.563 | ≤ 0.580 | ≤ 0.556 | ≤ 0.506 | ≤ 0.496 |
| Pellet Diameter(1) (inch) | ≤ 0.374 | ≤ 0.323 | ≤ 0.323 | ≤ 0.360 | ≤ 0.367 | ≤ 0.382 | ≤ 0.368 | ≤ 0.326 | ≤ 0.323 |
| Clad Outer Diameter(1) (inch) | ≥ 0.416 | ≥ 0.379 | ≥ 0.360 | ≥ 0.417 | ≥ 0.422 | ≥ 0.440 | ≥ 0.400 | ≥ 0.374 | ≥ 0.360 |
| Clad Thickness(1) (inch) | ≥ 0.024 | ≥ 0.024 | ≥ 0.022 | ≥ 0.026 | ≥ 0.024 | ≥ 0.026 | ≥ 0.022 | ≥ 0.022 | ≥ 0.022 |

Note 1. The fuel assembly fabrication documentation may be used to demonstrate compliance with these parameters which are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class or an array type.

Table 10
Irradiated EPR Fuel Rod Parameters

| Parameter | Value |
|---|---|
| Maximum Unirradiated Length | 179.5 inches |
| Cladding Thickness | Nominal 0.022 inch |
| Maximum Initial Uranium Content | 2 kgU/rod |

Table 11
Summary of PRA Requirements for PWR Fuel Assembly Classes

| Assembly Class | Number of Absorber Rods in PRAs and Locations | Diameter of $B_4C$ Absorber (cm) | Minimum $B_4C$ Content (g/cm) |
|---|---|---|---|
| WE 17x17 | 8, Per Figure 4 | 0.88 | 0.613 |
| CE 16x16 | 5, All Guide Tubes | 1.02 | 0.824 |
| BW 15x15 | 8, Per Figure 3 | 0.88 | 0.613 |
| CE 15x15 | 1, Center Guide Tube | 0.76 | 0.475 |
| CE 14x14 | 5, All Guide Tubes | 1.02 | 0.824 |
| WE 14x14 | 8, Per Figure 1 | 0.88 | 0.613 |
| WE 15x15 | 8, per Figure 2 | 0.88 | 0.613 |
| WE 16x16 | 8, Per Figure 5 | 0.88 | 0.613 |
| BW 17x17 | 8, Per Figure 4 | 0.76 | 0.475 |

5.(b)(1) Type and Form of Materials (continued)

Table 12
MOX Fuel Rods Specifications for Transport in the TN-LC-1FA Basket

| | |
|---|---|
| PHYSICAL PARAMETERS: | • Up to 21 PWR MOX fuel rods with physical parameters as those listed in Table 8.<br>• Up to 21 BWR MOX fuel rods with physical parameters as those listed in Table 13.<br>• Up to 21 EPR MOX fuel rods with physical parameters as those listed in Tables 8 and 10. |
| Fissile Material | $UO_2$, $PuO_2$ (Mixed Oxide or MOX) |
| Heavy Metal (HM) Content | ≤ 2.5 kgU/rod |
| CRITICALITY PARAMETERS:<br><br>Initial MOX composition | • $^{235}U$ Content in $UO_2$ : 0.5 ≤ $^{235}U$ ≤ 0.7 wt.%<br>• Plutonium Content: Pu / (U + Pu) ≤ 7.0 wt.%<br>• Initial $^{239}Pu$ Content in $PuO_2$ ≤ 60.0 wt.%<br>• Initial $^{241}Pu$ Content in $PuO_2$ ≤ 7.5 wt.% |
| THERMAL/RADIOLOGICAL PARAMETERS:<br><br>Initial MOX Composition for Fuel Qualification<br><br>Burnup and Minimum cooling time for MOX rods | • $^{238}Pu$ / $^{239}Pu$ ≤ 4.0 wt.%<br>• $^{239}Pu$/ $PuO_2$ ≥ 50 wt.%<br>• $^{241}Am$ / $PuO_2$ ≤ 70 ppm<br>• $^{235}U/U$ ≥ 0.5 wt.%<br><br>Per Table 21. |
| Maximum Decay heat per pin can | • 2.5 kW for the pin can with up to 21 rods<br>• 1.8 kW for the pin can with up to 9 rods |
| Minimum $^{10}B$ areal density in poison plates | • 16.7 mg/cm$^2$ Boron Aluminum Alloy / Metal Matrix Composite (MMC)<br>• 20.0 mg/cm$^2$ (Boral®) |

5.(b)(1) Type and Form of Materials (continued)

Table 13
BWR Fuel Specification for Transport in the TN-LC-1FA Basket

| | |
|---|---|
| PHYSICAL PARAMETERS:<br>Fuel Class(1) | One intact 7x7, 8x8, 9x9, or 10x10 BWR assembly manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table 14. |
| Channels | Fuel may be transported with or without channels, channel fasteners, or finger springs. |
| Fissile Material | $UO_2$ |
| Maximum Assembly Weight with Channels | 790 lbs |
| Maximum Unirradiated Assembly Length | 176.6 inches |
| THERMAL/RADIOLOGICAL PARAMETERS:<br>Maximum Planar Average Initial Enrichment | 5.0 wt.% $^{235}U$ |
| Fuel Assembly Average Burnup, Enrichment and Minimum Cooling Time | Per Table 16. |
| Maximum Decay Heat(2) | 2.0 kW per Assembly |
| Minimum $^{10}B$ aeral density in poison plates | • 16.7 mg/cm$^2$ Boron Aluminum Alloy / Metal Matrix Composite (MMC)<br>• 20.0 mg/cm$^2$ (Boral®) |

Notes:

1. Up to 21 fuel rods from any of the BWR fuel assemblies listed in Table 14 may also be transported in the TN-LC-1FA basket in the pin can. The fuel rods are loaded in a pin can with a cavity length of 169.55 inches (Option 3) which is placed within the BWR sleeve and hold-down ring and within the TN-LC-1FA basket. The required cooling time, as a function of BWR fuel rod burnup and enrichment, is provided in Table 19 for 21 rods and Table 20 for 9 rods, respectively.
2. The maximum decay heat per rod is 220 watts when loading up to 9 rods. The maximum decay heat per rod is 120 watts when loading 10 or more (up to 21) rods.

5.(b)(1) Type and Form of Materials (continued)

Table 14
BWR Fuel Assembly Design Characteristics(1) for Transportation in the TN-LC-1FA Basket
(Part 1 of 3)

| Transnuclear ID | 7x7-49/0 | 8x8-63/1 | 8x8-62/2 | 8x8-60/4 | 8x8-60/1 | 9x9-74/2 |
|---|---|---|---|---|---|---|
| Initial Design or Reload Fuel Designation | GE1 | GE4 | GE-5 | GE8 Type II | GE9 | GE11 |
| | GE2 | | GE-Pres | | GE10 | GE13 |
| | GE3 | | GE-Barrier | | | |
| | | | GE8 Type I | | | |
| | | | FANP 8x8-2 | | | |
| Maximum Number of Fuel Rods | 49 | 63 | 62 | 60 | 60 | 74 |
| Maximum Initial Uranium Content (kg) | 198 | 192 | 192 | 192 | 192 | 192 |
| Rod Pitch(5) (inch) | ≤ 0.738 | ≤ 0.640 | ≤ 0.640 | ≤ 0.640 | ≤ 0.640 | ≤ 0.566 |
| Pellet Diameter(5) (inch) | ≤ 0.487 | ≤ 0.416 | ≤ 0.411 | ≤ 0.411 | ≤ 0.411 | ≤ 0.376 |
| Clad Outer Diameter(5) (inch) | ≥ 0.563 | ≥ 0.493 | ≥ 0.483 | ≥ 0.483 | ≥ 0.483 | ≥ 0.440 |
| Clad Thickness(5) (inch) | ≥ 0.032 | ≥ 0.034 | ≥ 0.032 | ≥ 0.032 | ≥ 0.032 | ≥ 0.028 |

Table 14
BWR Fuel Assembly Design Characteristics(1) for Transportation in the TN-LC-1FA Basket
(Part 2 of 3)

| Transnuclear ID | 10x10-92/2 | 7x7-49/0Z | 7x7-48/1Z | 8x8-60/4Z | FANP 9x9 | Siemens QFA |
|---|---|---|---|---|---|---|
| Initial Design or Reload Fuel Designation | GE12 | ENC-IIIA | ENC-III(2) | ENC Va | FANP9 9x9(3) | 9x9 |
| | GE14 | | | ENC Vb | | |
| Maximum Number of Fuel Rods | 92 | 49 | 48 | 60 | 81 | 72 |
| Maximum Initial Uranium Content (kg) | 192 | 198 | 198 | 192 | 192 | 192 |
| Rod Pitch(5) (inch) | ≤ 0.510 | ≤ 0.738 | ≤ 0.738 | ≤ 0.642 | ≤ 0.572 | ≤ 0.570 |
| Pellet Diameter(5) (inch) | ≤ 0.345 | ≤ 0.488 | ≤ 0.491 | ≤ 0.420 | ≤ 0.357 | ≤ 0.374 |
| Clad Outer Diameter(5) (inch) | ≥ 0.404 | ≥ 0.570 | ≥ 0.570 | ≥ 0.501 | ≥ 0.424 | ≥ 0.433 |
| Clad Thickness(5) (inch) | ≥ 0.026 | ≥ 0.035 | ≥ 0.035 | ≥ 0.036 | ≥ 0.030 | ≥ 0.026 |

| 1. a. CERTIFICATE NUMBER | b. REVISION NUMBER | c. DOCKET NUMBER | d. PACKAGE IDENTIFICATION NUMBER | PAGE | | PAGES |
|---|---|---|---|---|---|---|
| 9358 | 5 | 71-9358 | USA/9358/B(U)F-96 | 16 | OF | 34 |

5.(b)(1) Type and Form of Materials (continued)

Table 14
BWR Fuel Assembly Design Characteristics(1) for Transportation in the TN-LC-1FA Basket

(Part 3 of 3)

| Transnuclear ID | 10x10-91/1 | ABB-8x8 | ABB-10x10 | LaCrosse |
|---|---|---|---|---|
| Initial Design or Reload Fuel Designation | ATRIUM 10 | SVEA-64 | SVEA-100(4) | Allis Chalmers - 10x10 |
| | ATRIUM 10XM | | | Exxon/ANF 10x10 |
| | | | | |
| Maximum Number of Fuel Rods | 91 | 64 | 100 | 100 |
| Maximum Initial Uranium Content (kg) | 192 | 192 | 192 | 125 |
| Rod Pitch(5) (inch) | ≤ 0.510 | ≤ 0.622 | ≤ 0.512 | ≤ 0.565 |
| Pellet Diameter(5) (inch) | ≤ 0.350 | ≤ 0.411 | ≤ 0.346 | ≤ 0.350 |
| Clad Outer Diameter(5) (inch) | ≥ 0.405 | ≥ 0.378 | ≥ 0.378 | ≥ 0.394 |
| Clad Thickness(5) (inch) | ≥ 0.023 | ≥ 0.024 | ≥ 0.022 | ≥ 0.020 |

Notes:
1. Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.
2. Includes ENC-IIIE and ENC-IIIF.
3. Includes FANP 9x9-72, 9x9-79, 9x9-80, and 9x9-81.
4. Includes SVEA-92, SVEA-96, SVEA-96+, SVEA-96 OPTIMA, SVEA-96 OPTIMA2, SVEA-96+/L.
5. The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type).

(2) Maximum quantity of material per package

(i) For the contents described in Item 5(b)(1)(i): 26 intact or damaged either NRU or NRX Mk I fuel assemblies, with an approximate maximum payload of 331 lb.

(ii) For the contents described in Item 5(b)(1)(ii): 54 intact or damaged MTR fuel elements, with an approximate maximum payload of 1,620 lb.

(iii) For the contents described in Item 5(b)(1)(iii): 180 intact TRIGA fuel elements/assemblies with an approximate maximum payload of 2,380 lb.

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

(iv) For the contents described in Item 5(b)(1)(iv): one intact PWR fuel assembly, or one intact BWR fuel assembly, or up to 21 intact PWR (including MOX and EPR) or BWR (including MOX) fuel rods in a pin can. When transporting 9 or fewer fuel rods, the rods shall be placed in the center 3x3 region of the pin can. The approximate maximum payload is 1,650 lb per PWR assembly, 1,850 lb per PWR assembly with PRAs, 710 lb per BWR assembly, 790 lb per BWR assembly with channels, and 16 lb per fuel rod.

(v) For the Unit 1 packaging, the contents described in Item 5(b)(1)(iv) are limited to: one intact PWR fuel assembly, or up to 21 intact PWR (excluding MOX and EPR) fuel rods in a pin can. When transporting 9 or fewer fuel rods, the rods shall be placed in the center 3x3 region of the pin can. The approximate maximum payload is 1,650 lb per PWR assembly, 1,850 lb per PWR assembly with PRAs, and 16 lb per fuel rod.

(3) The maximum decay heat for any payload is 3.0 kW.

5(c) Criticality Safety Index (CSI):

| | |
|---|---|
| For NRU and NRX fuel assemblies described in 5(b)(1)(i) and limited in 5(b)(2)(i) | 100 |
| For MTR fuel elements described in 5(b)(1)(ii) and limited in 5(b)(2)(ii) | 100 |
| For TRIGA fuel assemblies/elements described in 5(b)(1)(iii) and limited in 5(b)(2)(iii) | 0 |
| For intact BWR fuel assemblies described in 5(b)(1)(iv) and limited in 5(b)(2)(iv) | 0 |
| For intact PWR fuel assemblies described in 5(b)(1)(iv) and limited in 5(b)(2)(iv) | 100 |
| For fuel rods in a pin can described in 5(b)(1)(iv) and limited in 5(b)(2)(iv) | 0 |

**Table 15**
**Fuel Qualification Table for a PWR Fuel Assembly**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/MTU | Enrichment (wt. % $^{235}U$) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 2.25 | 2.25 | 2.20 | 2.10 | 2.05 | 2.05 | 2.05 | 2.00 | 2.00 | 2.00 | 2.00 | 2.00 | 2.00 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 |
| 20 | 3.37 | 3.35 | 3.30 | 3.20 | 3.05 | 2.90 | 2.90 | 2.85 | 2.85 | 2.80 | 2.80 | 2.80 | 2.75 | 2.75 | 2.75 | 2.75 | 2.75 | 2.75 | 2.70 | 2.70 | 2.70 | 2.70 | 2.65 | 2.65 | 2.65 | 2.65 | 2.65 | 2.65 | 2.60 | 2.60 | 2.60 | 2.60 | 2.60 | 2.60 | 2.60 | 2.55 |
| 30 | | | 4.70 | 4.35 | 4.10 | 3.80 | 3.70 | 3.65 | 3.60 | 3.60 | 3.55 | 3.50 | 3.45 | 3.45 | 3.40 | 3.35 | 3.35 | 3.35 | 3.30 | 3.30 | 3.25 | 3.25 | 3.20 | 3.20 | 3.15 | 3.15 | 3.15 | 3.15 | 3.15 | 3.15 | 3.10 | 3.10 | 3.10 | 3.05 | 3.05 | 3.05 |
| 39 | | | | | | 4.95 | 4.85 | 4.75 | 4.65 | 4.55 | 4.45 | 4.40 | 4.35 | 4.25 | 4.20 | 4.15 | 4.10 | 4.00 | 3.95 | 3.95 | 3.90 | 3.85 | 3.80 | 3.75 | 3.70 | 3.70 | 3.70 | 3.65 | 3.65 | 3.60 | 3.55 | 3.55 | 3.50 | 3.50 | 3.50 | 3.50 |
| 40 | | | | | | | | | | | | 4.55 | 4.45 | 4.35 | 4.30 | 4.25 | 4.15 | 4.15 | 4.10 | 4.05 | 4.00 | 3.90 | 3.90 | 3.90 | 3.85 | 3.80 | 3.75 | 3.70 | 3.70 | 3.65 | 3.65 | 3.65 | 3.60 | 3.55 | 3.55 | 3.50 |
| 45 | | | | | | | | | | | | 5.40 | 5.25 | 5.15 | 5.05 | 4.95 | 4.85 | 4.80 | 4.70 | 4.60 | 4.55 | 4.50 | 4.45 | 4.35 | 4.35 | 4.30 | 4.20 | 4.15 | 4.10 | 4.10 | 4.05 | 4.00 | 3.95 | 3.95 | 3.90 | 3.85 |
| 50 | | | | | | | | | | | | 6.80 | 6.60 | 6.50 | 6.25 | 6.15 | 6.00 | 5.85 | 5.75 | 5.60 | 5.50 | 5.40 | 5.30 | 5.20 | 5.10 | 5.05 | 4.95 | 4.90 | 4.85 | 4.75 | 4.70 | 4.65 | 4.55 | 4.55 | 4.50 | 4.40 |
| 55 | | | | | | | | | | | | 8.85 | 8.60 | 8.30 | 8.05 | 7.85 | 7.65 | 7.35 | 7.15 | 7.00 | 6.80 | 6.65 | 6.45 | 6.30 | 6.20 | 6.05 | 5.90 | 5.85 | 5.70 | 5.65 | 5.50 | 5.45 | 5.35 | 5.30 | 5.25 | 5.15 |
| 60 | | | | | | | | | | | | 11.55 | 11.20 | 10.85 | 10.50 | 10.15 | 9.80 | 9.55 | 9.20 | 8.95 | 8.70 | 8.45 | 8.25 | 8.00 | 7.80 | 7.55 | 7.40 | 7.20 | 7.05 | 6.85 | 6.75 | 6.60 | 6.45 | 6.35 | 6.25 | 6.10 |
| 61 | | | | | | | | | | | | 12.15 | 11.80 | 11.45 | 11.10 | 10.70 | 10.35 | 10.10 | 9.75 | 9.45 | 9.20 | 8.90 | 8.65 | 8.35 | 8.20 | 7.90 | 7.75 | 7.55 | 7.40 | 7.20 | 7.00 | 6.85 | 6.75 | 6.55 | 6.50 | 6.40 |
| 62 | | | | | | | | | | | | 12.80 | 12.40 | 12.05 | 11.65 | 11.30 | 10.90 | 10.65 | 10.25 | 9.95 | 9.70 | 9.40 | 9.10 | 8.85 | 8.55 | 8.35 | 8.15 | 7.90 | 7.70 | 7.50 | 7.30 | 7.20 | 7.05 | 6.85 | 6.75 | 6.65 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 15a**
**Fuel Qualification Table for a PWR Fuel Assembly – Unit 1 Packaging - 3.10″ Lead Thickness**

(Minimum required years of cooling time after reactor core discharge)

| Burn-up, GWd/ MTU | Enrichment, wt. % U-235 | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 3.0 | 3.0 | 2.9 | 2.9 | 2.8 | 2.8 | 2.8 | 2.8 | 2.7 | 2.7 | 2.7 | 2.7 | 2.6 | 2.6 | 2.6 | 2.6 | 2.6 | 2.6 | 2.6 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.5 | 2.4 | 2.4 | 2.4 | 2.4 | 2.4 |
| 20 | 4.7 | 4.6 | 4.5 | 4.4 | 4.3 | 4.2 | 4.2 | 4.1 | 4.0 | 4.0 | 3.9 | 3.9 | 3.8 | 3.8 | 3.8 | 3.7 | 3.7 | 3.6 | 3.6 | 3.6 | 3.6 | 3.5 | 3.5 | 3.5 | 3.5 | 3.4 | 3.4 | 3.4 | 3.4 | 3.4 | 3.4 | 3.3 | 3.3 | 3.3 | 3.3 | 3.3 |
| 30 | | | 6.7 | 6.5 | 6.3 | 6.2 | 6.0 | 5.9 | 5.7 | 5.6 | 5.5 | 5.3 | 5.2 | 5.1 | 5.1 | 5.0 | 4.9 | 4.8 | 4.7 | 4.7 | 4.6 | 4.6 | 4.5 | 4.5 | 4.4 | 4.4 | 4.3 | 4.3 | 4.2 | 4.2 | 4.2 | 4.1 | 4.1 | 4.1 | 4.1 | 4.0 |
| 39 | | | | | | 7.1 | 6.9 | 6.7 | 6.6 | 6.4 | 6.3 | 6.2 | 6.0 | 5.9 | 5.8 | 5.7 | 5.6 | 5.5 | 5.4 | 5.4 | 5.3 | 5.2 | 5.2 | 5.1 | 5.0 | 5.0 | 4.9 | 4.9 | 4.8 | 4.8 | 4.8 | 4.7 | 4.7 | 4.6 | 4.6 | 4.6 |
| 40 | | | | | | | | | | | | 6.4 | 6.2 | 6.1 | 6.0 | 5.9 | 5.8 | 5.7 | 5.6 | 5.5 | 5.4 | 5.4 | 5.3 | 5.2 | 5.2 | 5.1 | 5.1 | 5.0 | 5.0 | 4.9 | 4.9 | 4.8 | 4.8 | 4.7 | 4.7 | 4.7 |
| 50 | | | | | | | | | | | | 9.6 | 9.4 | 9.2 | 8.9 | 8.7 | 8.5 | 8.3 | 8.1 | 7.9 | 7.8 | 7.6 | 7.5 | 7.3 | 7.2 | 7.1 | 6.9 | 6.8 | 6.7 | 6.6 | 6.5 | 6.4 | 6.3 | 6.2 | 6.2 | 6.1 |
| 55 | | | | | | | | | | | | 12.0 | 11.7 | 11.4 | 11.1 | 10.8 | 10.5 | 10.3 | 10.0 | 9.8 | 9.6 | 9.4 | 9.1 | 8.9 | 8.8 | 8.6 | 8.4 | 8.2 | 8.1 | 7.9 | 7.8 | 7.7 | 7.5 | 7.4 | 7.3 | 7.2 |
| 60 | | | | | | | | | | | | 14.8 | 14.4 | 14.1 | 13.7 | 13.4 | 13.0 | 12.7 | 12.4 | 12.1 | 11.8 | 11.5 | 11.3 | 11.0 | 10.7 | 10.5 | 10.3 | 10.1 | 9.8 | 9.6 | 9.4 | 9.3 | 9.1 | 8.9 | 8.8 | 8.6 |
| 61 | | | | | | | | | | | | 15.4 | 15.0 | 14.7 | 14.3 | 13.9 | 13.6 | 13.3 | 12.9 | 12.6 | 12.3 | 12.0 | 11.7 | 11.5 | 11.2 | 10.9 | 10.7 | 10.5 | 10.2 | 10.0 | 9.8 | 9.6 | 9.4 | 9.3 | 9.1 | 8.9 |
| 62 | | | | | | | | | | | | 16.1 | 15.7 | 15.3 | 14.9 | 14.5 | 14.2 | 13.8 | 13.5 | 13.1 | 12.8 | 12.5 | 12.2 | 11.9 | 11.7 | 11.4 | 11.1 | 10.9 | 10.7 | 10.4 | 10.2 | 10.0 | 9.8 | 9.6 | 9.4 | 9.3 |
| Enr. wt.% | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Note:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 16**
**Fuel Qualification Table for a BWR Fuel Assembly**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/ MTU | Enrichment (wt. % $^{235}U$) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.65 | 0.65 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 |
| 20 | 0.95 | 0.95 | 0.90 | 0.85 | 0.80 | 0.80 | 0.80 | 0.80 | 0.80 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 |
| 30 | | | 1.25 | 1.20 | 1.15 | 1.10 | 1.10 | 1.10 | 1.10 | 1.10 | 1.10 | 1.10 | 1.10 | 1.05 | 1.05 | 1.05 | 1.05 | 1.05 | 1.05 | 1.05 | 1.05 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 0.95 | 0.95 | 0.95 | 0.95 |
| 39 | | | | | | 1.40 | 1.40 | 1.40 | 1.35 | 1.35 | 1.35 | 1.35 | 1.30 | 1.30 | 1.30 | 1.30 | 1.30 | 1.25 | 1.25 | 1.25 | 1.25 | 1.25 | 1.20 | 1.20 | 1.20 | 1.20 | 1.20 | 1.20 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 |
| 40 | | | | | | | | | | | | 1.40 | 1.40 | 1.35 | 1.35 | 1.35 | 1.35 | 1.30 | 1.30 | 1.30 | 1.30 | 1.30 | 1.25 | 1.25 | 1.25 | 1.25 | 1.25 | 1.25 | 1.25 | 1.20 | 1.20 | 1.20 | 1.20 | 1.20 | 1.20 | 1.20 |
| 45 | | | | | | | | | | | | 1.60 | 1.60 | 1.60 | 1.55 | 1.55 | 1.55 | 1.55 | 1.50 | 1.50 | 1.50 | 1.50 | 1.50 | 1.50 | 1.50 | 1.50 | 1.50 | 1.45 | 1.45 | 1.45 | 1.45 | 1.45 | 1.45 | 1.40 | 1.40 | 1.40 |
| 50 | | | | | | | | | | | | 1.85 | 1.85 | 1.85 | 1.80 | 1.80 | 1.80 | 1.75 | 1.75 | 1.75 | 1.75 | 1.75 | 1.75 | 1.75 | 1.75 | 1.70 | 1.70 | 1.70 | 1.70 | 1.65 | 1.65 | 1.65 | 1.65 | 1.65 | 1.60 | 1.60 |
| 55 | | | | | | | | | | | | 2.10 | 2.10 | 2.10 | 2.05 | 2.05 | 2.05 | 2.00 | 2.00 | 2.00 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.95 | 1.90 | 1.90 | 1.90 | 1.90 | 1.90 | 1.85 | 1.85 | 1.85 | 1.85 |
| 60 | | | | | | | | | | | | 2.35 | 2.35 | 2.35 | 2.30 | 2.30 | 2.30 | 2.25 | 2.25 | 2.25 | 2.20 | 2.20 | 2.20 | 2.20 | 2.20 | 2.20 | 2.15 | 2.15 | 2.15 | 2.15 | 2.10 | 2.10 | 2.10 | 2.10 | 2.05 | 2.05 |
| 61 | | | | | | | | | | | | 2.40 | 2.40 | 2.40 | 2.35 | 2.35 | 2.35 | 2.30 | 2.30 | 2.30 | 2.25 | 2.25 | 2.25 | 2.20 | 2.20 | 2.20 | 2.20 | 2.20 | 2.20 | 2.20 | 2.15 | 2.15 | 2.15 | 2.15 | 2.10 | 2.10 |
| 62 | | | | | | | | | | | | 2.45 | 2.45 | 2.45 | 2.40 | 2.40 | 2.40 | 2.35 | 2.35 | 2.35 | 2.30 | 2.30 | 2.30 | 2.25 | 2.25 | 2.25 | 2.25 | 2.25 | 2.25 | 2.25 | 2.20 | 2.20 | 2.20 | 2.20 | 2.15 | 2.15 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 17**
**Fuel Qualification Table for 21 PWR/EPR Fuel Rods ($UO_2$)**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/MTU | Enrichment (wt. % $^{235}U$) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 20 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 30 | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 39 | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 40 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 45 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 50 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 55 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 60 | | | | | | | | | | | | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 61 | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.30 | 0.30 |
| 62 | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 65 | | | | | | | | | | | | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 70 | | | | | | | | | | | | | | | | | | | | | | | 0.50 | 0.50 | 0.50 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 |
| 75 | | | | | | | | | | | | | | | | | | | | | | | 0.65 | 0.65 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.60 | 0.50 | 0.50 | 0.50 | 0.50 |
| 80 | | | | | | | | | | | | | | | | | | | | | | | 0.85 | 0.85 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 |
| 85 | | | | | | | | | | | | | | | | | | | | | | | 1.05 | 1.00 | 1.00 | 1.00 | 0.90 | 0.90 | 0.90 | 0.90 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 |
| 90 | | | | | | | | | | | | | | | | | | | | | | | 1.25 | 1.25 | 1.25 | 1.15 | 1.15 | 1.15 | 1.10 | 1.10 | 1.10 | 1.00 | 1.00 | 1.00 | 1.00 | 0.95 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 17a**

**Fuel Qualification Table for 21 PWR Fuel Rods ($UO_2$) – Unit 1 Packaging - 3.10″ Lead Thickness**

(Minimum required years of cooling time after reactor core discharge)

| Burn-up, GWd/ MTU | Enrichment, wt. % U-235 | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.35 | 0.35 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 20 | 0.35 | 0.35 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 30 | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 39 | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 40 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 45 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 50 | | | | | | | | | | | | 0.35 | 0.35 | 0.35 | 0.35 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 55 | | | | | | | | | | | | 0.37 | 0.37 | 0.37 | 0.36 | 0.36 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 60 | | | | | | | | | | | | 0.50 | 0.49 | 0.48 | 0.47 | 0.47 | 0.46 | 0.45 | 0.45 | 0.44 | 0.44 | 0.43 | 0.43 | 0.42 | 0.42 | 0.41 | 0.41 | 0.41 | 0.40 | 0.40 | 0.40 | 0.39 | 0.39 | 0.39 | 0.38 | 0.38 |
| 61 | | | | | | | | | | | | 0.53 | 0.52 | 0.51 | 0.5 | 0.5 | 0.49 | 0.48 | 0.47 | 0.47 | 0.46 | 0.46 | 0.45 | 0.44 | 0.44 | 0.44 | 0.43 | 0.43 | 0.42 | 0.42 | 0.42 | 0.41 | 0.41 | 0.41 | 0.40 | 0.40 |
| 62 | | | | | | | | | | | | 0.56 | 0.55 | 0.54 | 0.53 | 0.53 | 0.52 | 0.51 | 0.5 | 0.49 | 0.49 | 0.48 | 0.48 | 0.47 | 0.46 | 0.46 | 0.45 | 0.45 | 0.44 | 0.44 | 0.44 | 0.43 | 0.43 | 0.43 | 0.42 | 0.42 |
| 65 | | | | | | | | | | | | | | | | | | | | | | | 0.56 | 0.55 | 0.55 | 0.54 | 0.53 | 0.53 | 0.52 | 0.51 | 0.51 | 0.50 | 0.50 | 0.49 | 0.49 | 0.49 |
| 70 | | | | | | | | | | | | | | | | | | | | | | | 0.71 | 0.70 | 0.69 | 0.68 | 0.67 | 0.67 | 0.66 | 0.65 | 0.65 | 0.64 | 0.63 | 0.62 | 0.62 | 0.61 |
| 75 | | | | | | | | | | | | | | | | | | | | | | | 0.87 | 0.86 | 0.85 | 0.84 | 0.83 | 0.82 | 0.8 | 0.79 | 0.79 | 0.78 | 0.77 | 0.76 | 0.75 | 0.75 |
| 80 | | | | | | | | | | | | | | | | | | | | | | | 1.04 | 1.03 | 1.01 | 1.00 | 0.99 | 0.97 | 0.96 | 0.95 | 0.94 | 0.93 | 0.92 | 0.91 | 0.90 | 0.89 |
| 85 | | | | | | | | | | | | | | | | | | | | | | | 1.24 | 1.22 | 1.20 | 1.18 | 1.16 | 1.15 | 1.13 | 1.12 | 1.10 | 1.09 | 1.08 | 1.06 | 1.05 | 1.04 |
| 90 | | | | | | | | | | | | | | | | | | | | | | | 1.47 | 1.44 | 1.41 | 1.39 | 1.37 | 1.34 | 1.32 | 1.3 | 1.29 | 1.27 | 1.25 | 1.23 | 1.22 | 1.20 |
| Enr. wt.% | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Note1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 18**
**Fuel Qualification Table for 9 PWR/EPR Fuel Rods ($UO_2$)**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/ MTU | Enrichment (wt. % $^{235}U$) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 20 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 30 | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 39 | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 40 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 45 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 50 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 55 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 60 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 61 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 62 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 65 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 70 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 75 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 80 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 85 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 90 | | | | | | | | | | | | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 19**
**Fuel Qualification Table for 21 BWR Fuel Rods ($UO_2$)**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/ MTU | Enrichment (wt. % $^{235}$U) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 20 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 30 | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 39 | | | | | | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 40 | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 45 | | | | | | | | | | | | 0.45 | 0.45 | 0.45 | 0.45 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 |
| 50 | | | | | | | | | | | | 0.60 | 0.60 | 0.60 | 0.60 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 |
| 55 | | | | | | | | | | | | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.70 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 | 0.65 |
| 60 | | | | | | | | | | | | 1.00 | 1.00 | 1.00 | 1.00 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 | 0.75 |
| 61 | | | | | | | | | | | | 1.05 | 1.05 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.75 | 0.75 | 0.75 | 0.75 |
| 62 | | | | | | | | | | | | 1.10 | 1.05 | 1.05 | 1.05 | 1.05 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 | 0.85 |
| 65 | | | | | | | | | | | | | | | | | | | | | | | 1.05 | 1.05 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 1.00 | 0.90 | 0.90 | 0.90 | 0.90 | 0.90 |
| 70 | | | | | | | | | | | | | | | | | | | | | | | 1.20 | 1.20 | 1.20 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 | 1.15 | 1.10 | 1.10 | 1.10 | 1.10 | 1.10 |
| 75 | | | | | | | | | | | | | | | | | | | | | | | 1.45 | 1.45 | 1.45 | 1.40 | 1.40 | 1.40 | 1.30 | 1.30 | 1.30 | 1.30 | 1.25 | 1.25 | 1.25 | 1.25 |
| 80 | | | | | | | | | | | | | | | | | | | | | | | 1.70 | 1.70 | 1.65 | 1.65 | 1.60 | 1.60 | 1.60 | 1.50 | 1.50 | 1.50 | 1.45 | 1.45 | 1.45 | 1.45 |
| 85 | | | | | | | | | | | | | | | | | | | | | | | 2.15 | 2.05 | 2.00 | 2.00 | 1.95 | 1.85 | 1.85 | 1.80 | 1.80 | 1.70 | 1.70 | 1.65 | 1.65 | 1.65 |
| 90 | | | | | | | | | | | | | | | | | | | | | | | 2.60 | 2.55 | 2.50 | 2.40 | 2.35 | 2.30 | 2.20 | 2.15 | 2.15 | 2.10 | 2.00 | 2.00 | 1.95 | 1.95 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 20**
**Fuel Qualification Table for 9 BWR Fuel Rods ($UO_2$)**

(Minimum required years of cooling time after reactor core discharge)

| Burnup, GWd/MTU | Enrichment (wt. % $^{235}$U) | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |
| 10 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 20 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 30 | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 39 | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 40 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 45 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 50 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 55 | | | | | | | | | | | | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 60 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 | 0.25 |
| 61 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.25 | 0.25 | 0.25 |
| 62 | | | | | | | | | | | | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 | 0.30 |
| 65 | | | | | | | | | | | | | | | | | | | | | | | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 | 0.35 |
| 70 | | | | | | | | | | | | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 |
| 75 | | | | | | | | | | | | | | | | | | | | | | | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 | 0.40 |
| 80 | | | | | | | | | | | | | | | | | | | | | | | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 | 0.45 |
| 85 | | | | | | | | | | | | | | | | | | | | | | | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 | 0.50 |
| 90 | | | | | | | | | | | | | | | | | | | | | | | 0.60 | 0.60 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 | 0.55 |
| | 0.7 | 0.8 | 0.9 | 1.2 | 1.5 | 2.0 | 2.1 | 2.2 | 2.3 | 2.4 | 2.5 | 2.6 | 2.7 | 2.8 | 2.9 | 3.0 | 3.1 | 3.2 | 3.3 | 3.4 | 3.5 | 3.6 | 3.7 | 3.8 | 3.9 | 4.0 | 4.1 | 4.2 | 4.3 | 4.4 | 4.5 | 4.6 | 4.7 | 4.8 | 4.9 | 5.0 |

Notes:

1. Explanatory notes and limitations regarding the use of this table follow Table 21.

**Table 21**

**Fuel Qualification Table for MOX PWR/BWR/EPR 21 Rods and MOX PWR/BWR/EPR 9 Rods**

| Burnup, GWd/MTHM | 9 Rods | | 21 Rods | |
|---|---|---|---|---|
| | 0.5 wt.% of $^{235}U$ | 0.7 wt.% of $^{235}U$ | 0.5 wt.% of $^{235}U$ | 0.7 wt.% of $^{235}U$ |
| 10 | 0.25 | 0.25 | 0.25 | 0.25 |
| 20 | 0.25 | 0.25 | 0.30 | 0.30 |
| 30 | 0.25 | 0.25 | 0.50 | 0.50 |
| 40 | 0.25 | 0.25 | 0.95 | 0.95 |
| 45 | 0.25 | 0.25 | 1.25 | 1.25 |
| 50 | 0.35 | 0.35 | 1.70 | 1.70 |
| 55 | 0.40 | 0.40 | 2.20 | 2.10 |
| 60 | 0.45 | 0.45 | 2.80 | 2.70 |
| 62 | 0.55 | 0.55 | 3.75 | 3.65 |

Notes:

1. Explanatory notes and limitation regarding the use of this table are provided on the following page.

Notes:

General

1. Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
2. For values not explicitly listed in the tables, round burnups **up** to the first value shown, round enrichments **down**, and select the cooling time listed.
3. $UO_2$ Fuel with an initial enrichment less than 0.7 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% $^{235}U$ is unacceptable for transportation.
4. Shaded areas in these Tables indicate fuel is not analyzed for loading.

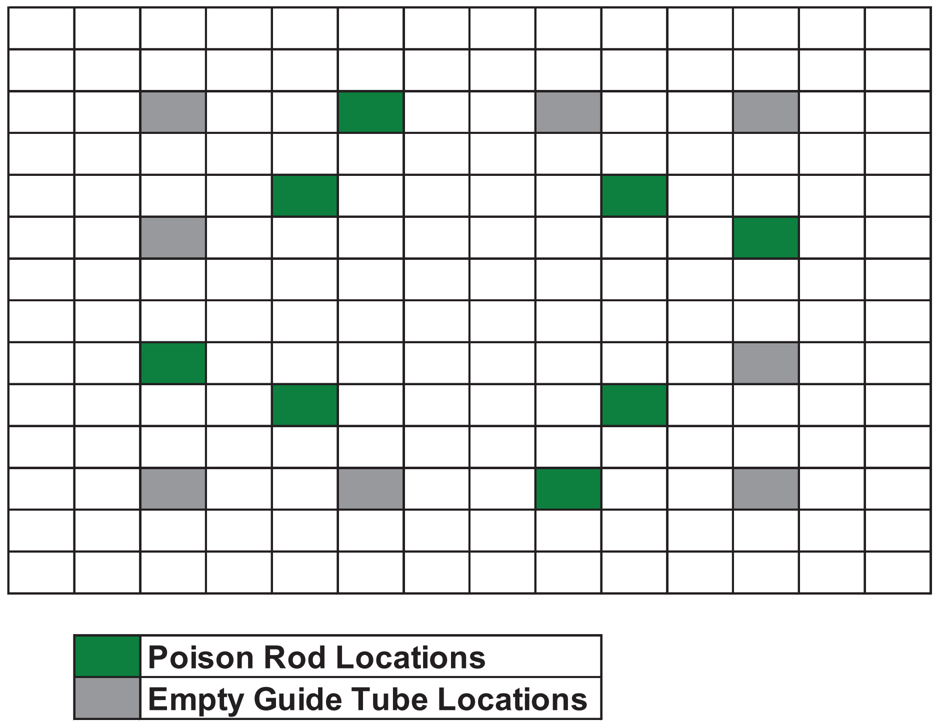
For Fuel Assemblies

1. Burnup = Assembly Average burnup.
2. Enrichment = Assembly Average Enrichment.
3. Fuel assembly with a burnup greater than 62 GWd/MTU is unacceptable for transportation.

For Fuel Rods

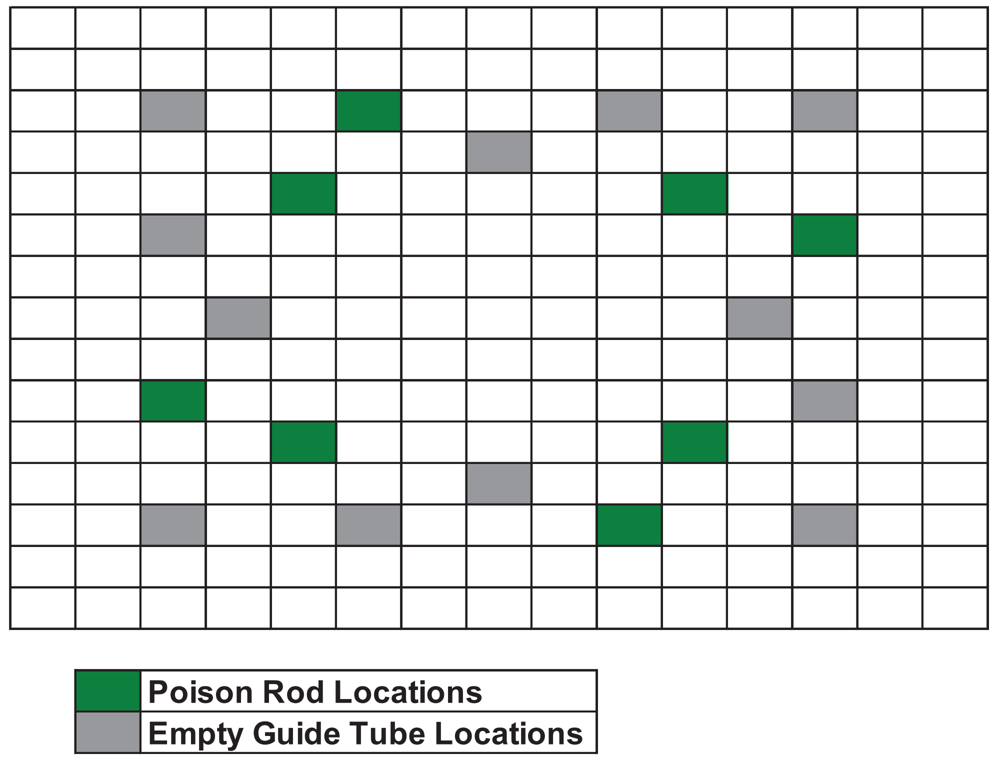
4. Burnup = Maximum burnup.
5. Enrichment = Rod Average Enrichment.
6. When transporting 21 or less fuel rods, the rods shall be placed in a specially designed pin can.
7. When transporting 9 or less fuel rods, the rods shall be placed in the 3x3 region of the pin can.
8. Fuel rods with a burnup greater than 90 GWd/MTU are unacceptable for transportation.

Example: Per Table 15, a PWR assembly with an initial enrichment of 4.85 wt.% $^{235}U$ and a burnup of 41.5 GWd/MTU is acceptable for transport after a 3.95-year cooling time as defined by 4.8 wt.% $^{235}U$ (rounding down) and 45 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).



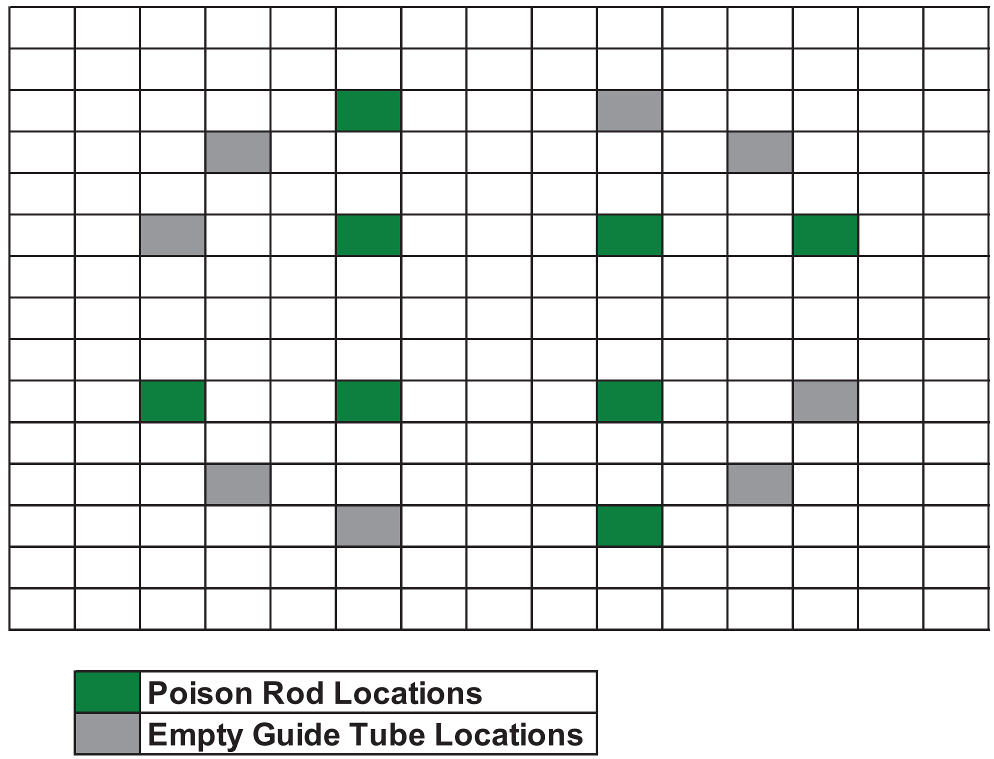

Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

**Figure 1**
**PRA Insertion Locations for WE 14x14 Class Assemblies**



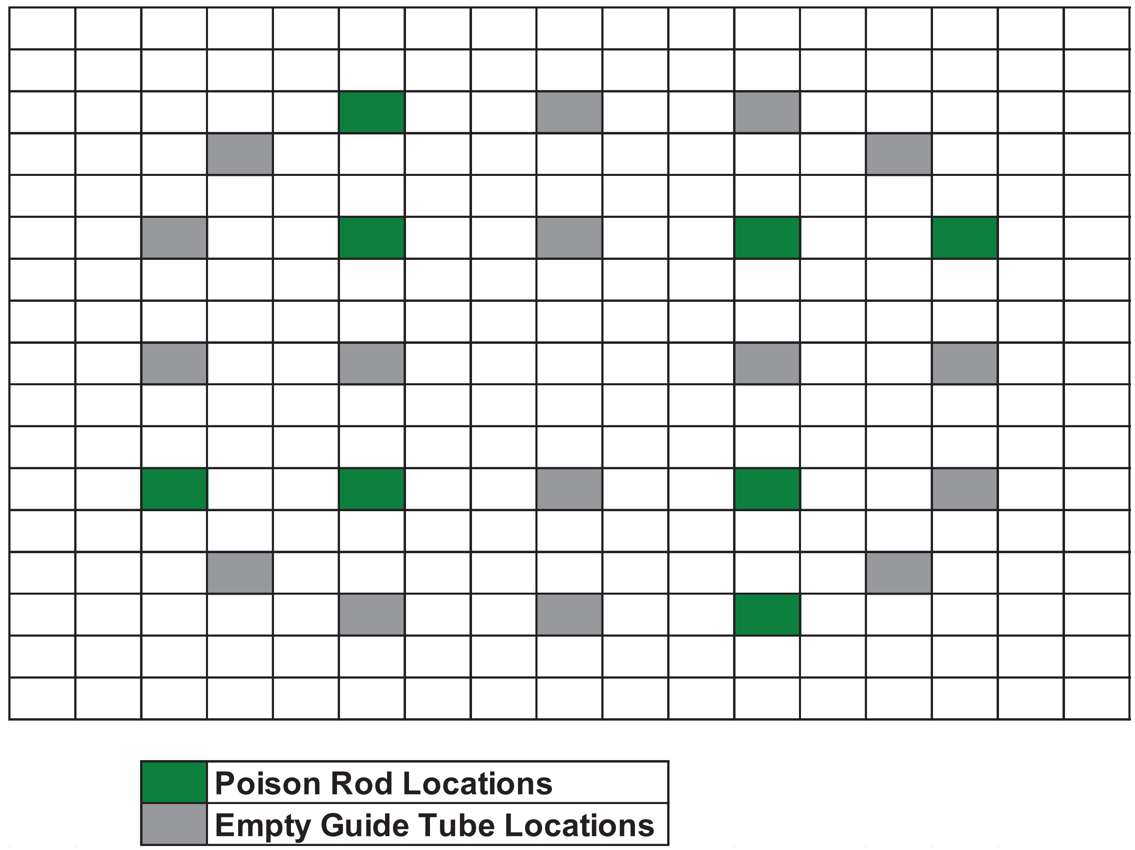

Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

**Figure 2**
**PRA Insertion Locations for WE 15x15 Class Assemblies**



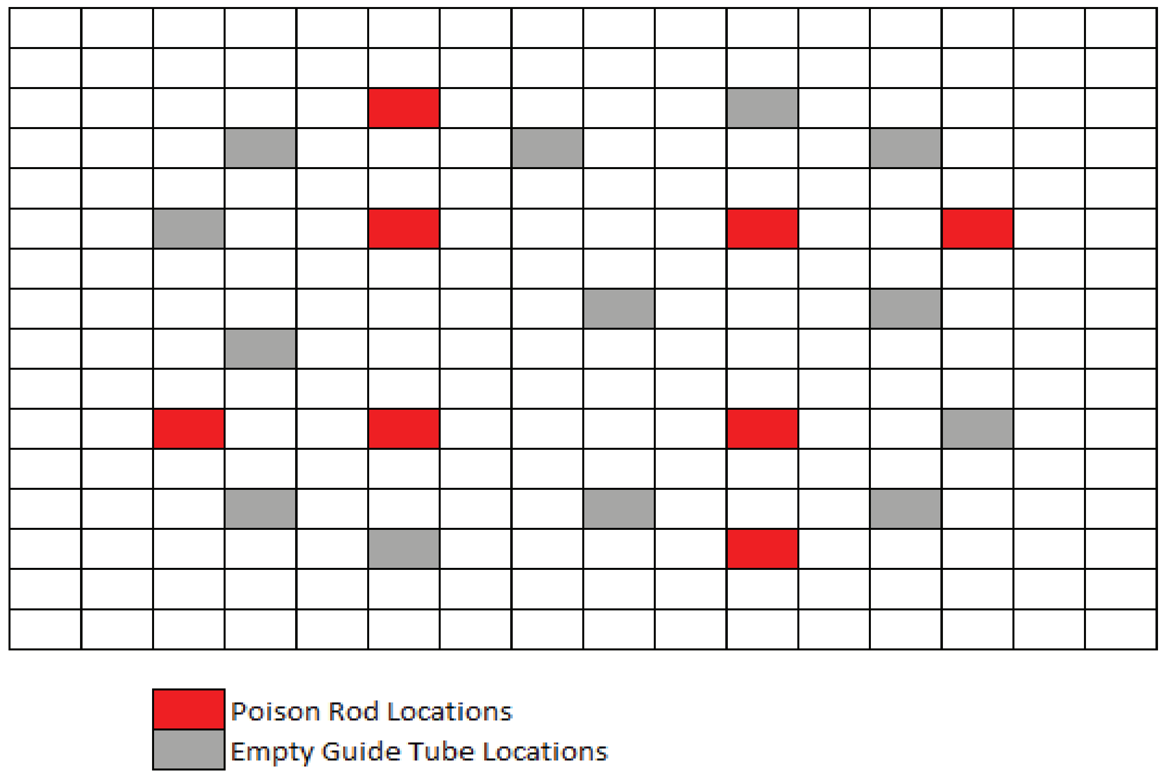

Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

**Figure 3**
**PRA Insertion Locations for BW 15x15 Class Assemblies**




Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

**Figure 4**
**PRA Insertion Locations for BW 17x17 and WE 17x17 Class Assemblies**




Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

**Figure 5**
**PRA Insertion Locations for WE 16x16 Class Assemblies**

6. 6. 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 of Chapter No. 7 of the application, and

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter No. 8 of the application.

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

8. Prior to the first shipment, the package shall be tested for the entire containment boundary, e.g., all base metal, all joining containment welds, vent port plug seal, drain port plug seal, lid seal, and bottom plug seal, in accordance with ANSI N14.5, by helium leakage rate testing to meet the leaktight criteria of $1.0 \times 10^{-7}$ ref-cm$^3$/sec for fabrication leakage tests.

9. Poison Rod Assemblies, required for shipment of PWR assemblies, shall be installed such that the active fuel length is covered by the absorber, and measures shall be taken against their inadvertent removal from the fuel assembly.

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

11. Expiration date: December 31, 2022.

UNITED STATES NUCLEAR REGULATORY COMMISSION

REFERENCES

TN-LC Transportation Package Safety Analysis Report, Revision No. 9.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John B. McKirgan

Digitally signed by John B. McKirgan
Date: 2020.12.22 07:24:01 -05'00'

John McKirgan, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: December 21, 2020

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

**SAFETY EVALUATION REPORT**
**TN AMERICAS LLC**
**Model No. TN-LC Package**
**Certificate of Compliance No. 9358**
**Revision No. 5**

## SUMMARY

By letter dated April 23, 2020, TN Americas LLC (the applicant) submitted an amendment request for Certificate of Compliance No. 9358 for the Model No. TN-LC package (ADAMS Accession No. ML20114E072). The staff performed an acceptance review of the application and transmitted, by letter dated June 5, 2020, a request for supplemental information (RSI) from the acceptance review. On July 9, 2020, the applicant provided responses to the RSIs (ADAMS Accession No. ML20191A365). On October 30, 2020, the applicant provided responses (ADAMS Accession No. ML20304A318) to the staff's request for additional information (RAI) letter dated September 22, 2020, and submitted a consolidated application, Revision No. 9, by letter dated December 10, 2020 (ADAMS Accession No: ML20345A113).

This amendment request included several changes as follows: The WE 16x16 was added to the list of fuel classes to be transported in the TN-LC-1FA Basket. Fuel designations were added for WE 17x17, WE 14x14, CE 16x16, WE 16x16 fuel classes. Licensing drawings were revised to include tolerances on Important to Safety (ITS) components and also to account for the "as fabricated" cask body of the TN-LC Unit 1 which has a reduction in its shielding capability due to localized areas where the lead thickness is less than the acceptance criterion. This non-conformance resulted in an "as-fabricated" measured cavity length is 182.12 inches compared to a minimum cask cavity length less than 182.50 inches per the original design. As a result of the reduced lead thickness condition for the TN-LC Unit 1, a new shielding evaluation assessed the impacts of the nonconforming thickness conditions with respect to the limiting dose rates for TN-LC-1FA PWR fuel assembly and PWR fuel rods.

The TN-LC package is designed and tested for a leak rate of $1 \times 10^{-7}$ ref $cm^3/s$, defined as "leak-tight" per ANSI N14.5. A reference air leak rate that allows periodic and maintenance testing to less than leak-tight is established that satisfies containment requirements for the 1FA PWR and BWR contents (assemblies and rods, MOX excluded). A new leak rate evaluation is included in this amendment request to determine an allowable leak rate for the 1FA contents, excluding MOX fuel. Operating Instructions and Acceptance Criteria and Maintenance Procedures were also revised to allow for less sensitive leak rate measurement methods and take into account NRC IN-2016-04, ANSI N14.5-2014 and leakage rate testing considerations.

NRC staff reviewed the application using the guidance in NUREG-1617 " Standard Review Plan for Transportation Packages for Spent Nuclear Fuel". The analyses performed by the applicant demonstrate that the package provides adequate structural, thermal, containment, shielding, and criticality protection under normal and accident conditions.

Based on the statements and representations in the application, and the conditions listed in the Certificate of Compliance, the staff concludes that the package meets the requirements of 10 CFR Part 71.

## 1.0 GENERAL INFORMATION

### 1.1 Packaging

The first fabricated TN-LC unit has a shielding non-conformance which resulted in a minimum lead thickness as verified by gamma scan that is less than the specified 3.38 minimum lead thickness, and this non-conformance resulted in a minimum cask cavity length of 182.12 inches, i.e., less than 182.50 inches.

A new material class was added for the impact limiter bolts, and the applicant has specified the number of fusible plugs that are designed to melt during a fire accident. All containment boundary welds are full penetration bevel or groove welds to ensure structural and sealing integrity. These full penetration welds are designed per ASME III Subsection NB and are fully examined by radiographic or ultrasonic methods in accordance with Subsection NB. Since the TN-LC package is designed to be leak-tight, no release calculations are performed except for the 1FA contents (see Appendix 4.6.1) in order to relax the pre-shipment leak- tightness criterion when shipping LWR fuel assemblies or rods.

### 1.2 Contents

The pin can was originally designed to allow the loading of 25 individual fuel pins. A design change, made during fabrication in order to improve the underwater operation of the lid, modified the four corner fuel pin tubes to accommodate lid fastening bolts and the 25-pin can is now referred only as a pin can, reducing the basket maximum capacity from 25 to 21 fuel rods

A new WE 16x16 fuel class was added as authorized contents and revised notes were provided for WE 17x17, W 14x14, CE 16x16, and WE 16x16 fuel designations. PRA requirements and insertion locations for WE 16x16 fuel were also added.

Staff notes that this amendment request is for PWR fuel transportation in the TN-LC Unit 1, no analysis has been made to transport TRIGA fuel in the Unit 1 packaging, and there currently are no plans to transport that fuel.

### 1.3 Materials

The materials of the TN-LC package are well specified in the application.

### 1.4 Drawings

Several modifications were made to the licensing drawings, particularly to include tolerances. The reduced cavity length of the TN-LC packaging was adjusted from “182.12” to “182.10 minimum”, and the pin can overall length is now limited to 181.91” maximum (versus a 181.92” maximum previously). The minimum thickness of the lead layer inside the lid was adjusted to 3.99” while the minimum thickness of the lead inside the bottom flange is now 3.49”. The missing length of item 8D: 6.00 +.05/-.01 was added to define the amount of lead shielding in the bottom plug. The quantity/material specification/quality category/code criterion of the dowel pins were changed to “A/R”/”Stainless Steel”/”NITS”/”NON-CODE” respectively to support ongoing basket modifications. A ±0.05” tolerance was added to thicknesses of the basket plates. The applicant also proposed to remove the designation of ‘Stk’ from the thickness dimension for the basket’s top and lateral frame components which surround the basket cavity, but specified a new tolerance and performed an analysis to show the impact of the new tolerances on package radiation levels, conservatively applying the effect of two times the specified tolerance. The results of that evaluation indicated that the radial package radiation levels would still be below the regulatory limits.

The packaging is constructed and assembled in accordance with the following drawings:

| | |
|---|---|
| 65200-71-01 Revision 9 | TN-LC Cask Assembly (11 sheets) |
| 65200-71-02 Revision 0 | TN-LC Transport Cask<br>Regulatory Plate (1 sheet) |
| 65200-71-20 Revision5 | TN-LC<br>Impact Limiter Assembly (2 sheets) |
| 65200-71-21 Revision 2 | TN-LC Transport Packaging<br>Transport Configuration (1 sheet) |
| 65200-71-40 Revision 4 | TN-LC-NRUX Basket<br>Basket Assembly (5 sheets) |
| 65200-71-50 Revision 4 | TN-LC-NRUX Basket<br>Basket Tube Assembly (5 sheets) |
| 65200-71-60 Revision 4 | TN-LC-MTR Basket<br>General Assembly (4 sheets) |
| 65200-71-70 Revision 4 | TN-LC-MTR Basket<br>Fuel Bucket (2 sheets) |
| 65200-71-80 Revision 4 | TN-LC-TRIGA Basket (5 sheets) |
| 65200-71-90 Revision 6 | TN-LC-1FA Basket (5 sheets) |
| 65200-71-96 Revision 5 | TN-LC-1FA BWR<br>Sleeve and Hold-Down Ring (2 sheets) |
| 65200-71-102 Revision 7 | TN-LC-1FA<br>Pin Can Basket (5 sheets) |

## 1.5 Evaluation Findings

A general description of the Model No. TN-LC package is presented in Section 1 of the package application, with special attention to design and operating characteristics and principal safety considerations. Drawings for structures, systems, and components important to safety are included in the application.

The application identifies the Quality Assurance Program and the applicable codes and standards for the design, fabrication, assembly, testing, operation, and maintenance of the package.

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. TN-LC package against 10 CFR Part 71 requirements for each technical discipline.

## 2.0 STRUCTURAL AND MATERIALS EVALUATION

The objective of the structural review is to determine that the information presented in the application, including the description of the packaging, design and fabrication criteria, structural material properties, and structural performance of the package design for the tests under normal conditions of transport (NCT) and hypothetical accident conditions (HAC), is complete and meets the requirements of 10 CFR Part 71.

TN Americas LLC included two changes to Chapter 2 “TN-LC Transport Package Structural Evaluation” in addition to editorial changes. The first change included a revision to the description to the pin can used for specific applications of the TN-LC-1FA basket. This revision was made to clarify a change from the original design of the pin can and does not impact the assumptions used in the structural evaluation of the basket.

The second change included a revision to the application to allow for ASME BPVC SA-540 GR. B23 or B24 CL. 1 for lid bolts, ram access cover bolts, and trunnion bolts. The justification for the change was to allow suppliers more flexibility in meeting material requirements including tests for fracture at low temperatures while providing the same capacity on stress values since both grades have the same design stress values.

The staff finds the proposed revision to allow for ASME BPVC SA-540 GR. B23 or B24 CL. 1 for lid bolts, ram access cover bolts, and trunnion bolts acceptable. The staff reviewed and confirmed that the design stresses for both grades are equivalent using the values from ASME BPVC.

Therefore, the proposed change will not impact the structural capacity of the package or its effectiveness in withstanding normal and abnormal conditions.

The staff also reviewed the other changes and determined that they do not impact the basis for the structural evaluation of the package. The staff concludes that the proposed changes will not impact the assumptions used to demonstrate that the package meets the regulatory requirements of 10 CFR Part 71 applicable to the structural review.

Because the contents may have high burnup (up to 90 GWd/MTU), the staff reviewed the O-ring stability against radiation. The O-ring is stable at $10^6$ – $10^7$ rads per year, with the exception of $10^4$ – $10^5$ rads per year for fluorocarbon-based O-rings. The application provided that the O-ring will be subject to radiation below these levels. Therefore, the staff finds it acceptable.

Based on the review of the statements and representations in the application for the TN-LC transportation package, the staff finds that the applicant adequately described and evaluated the materials performance of the TN-LC package. The staff concludes that the TN-LC package meets the package requirements of 10 CFR 71.31.

## 3.0 THERMAL EVALUATION

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

Section 3.1.2, “Decay heat,” of the application describes that the added content, intact unconsolidated WE 16x16 class PWR assemblies, in Section 1.2.2, “Contents,” and Table 1.4.5-1, “PWR Fuel Specification for the Fuel to be Transported in the TN-LC-1FA Basket,” of the application has a lower decay heat (2.5 kilowatt (kW)) than the decay heat used in the bounding thermal analysis (3.0 kW).

Section 3.6.7.1, “UO2 and MOX irradiated fuel assembly thermal conductivity,” of the application describes that the applicant concluded the previously analyzed UO2 irradiated fuel assembly thermal conductivity bounds the added WE 16x16 content, the staff finds this to be acceptable. Based on the staff’s review of the changes in Sections 3.1.2 and 3.6.7.1 of the application, the staff finds it to be acceptable that the thermal analysis remains bounding for the WE 16x16 class fuel assembly content.

### 3.1 Evaluation Findings

#### 3.1.1 Description of the Thermal Design

The staff has reviewed the package description and evaluation of the decay heat and concludes that they satisfy the thermal requirements of 10 CFR Part 71.

#### 3.1.2 Material Properties and Component Specifications

The staff has reviewed the material properties used in the thermal evaluation and concludes that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

## 4.0 CONTAINMENT EVALUATION

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

### 4.1 Description of the Containment System

The staff verified the applicant described the closure bolt torque requirements on the licensing drawing No. 65200-71-01 and that the closure bolt torque requirements were referenced in Chapter 7, “Package Operations,” of the application to provide reasonable assurance of positive closure.  The staff reviewed the changes to Drawing No. 65200-71-01 based on the applicant’s response to request for additional information (RAI) 8-1 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20304A319), and the staff finds these changes to be acceptable.  The staff reviewed the editorial drawing changes and based on the staff’s review, finds the changes to be acceptable.

4.2 Containment Analysis and Leak Rate Test Acceptance Criterion for 1FA Contents

The applicant described in Section 4.2.3, “Containment Criterion,” of the application for normal conditions of transport and 4.3.3, “Containment Criterion,” of the application for hypothetical accident conditions that for a shipment of 1FA contents (PWR, BWR, or pin can), a leak rate test criterion that is less restrictive than the leaktight criterion may be used for the periodic, maintenance, and pre-shipment leakage rate tests, which is established in Appendix 4.6.1, “Containment Reference Leak Rate for 1FA Contents (Excluding MOX),” of the application. These 1FA contents, where a leak rate test criterion that is less restrictive than leaktight has been established, shall not include MOX; therefore, any MOX contents shall use the leaktight criterion.

The applicant provided the radioactive release rates in Table 4.6.1-1, “10 CFR 71 Containment Criteria for Type B Transportation Packages,” of the application; the staff confirmed that the values in the table were consistent with, or in the case of Kr-85 more conservative than, the release rate requirements in 10 CFR Part 71.51(a)(1) and (2). Therefore, the staff finds the release rates to be acceptable.

The applicant described in Section 4.6.1.1.2, “Parameters,” of the application, that the contributions to the radionuclide releasable source term include fuel fines, volatiles, fission gases, and crud. The applicant provided the bounding $A_2$ values available for release in Table 4.6.1-2, “Bounding $A_2$ Values for Fission Gases, Volatiles and Fines,” of the application for the BWR and PWR 1FA content that was determined using the ORIGEN-ARP module of SCALE 6.0.

The applicant provided bounding release fractions in Table 4.6.1-3, “Bounding Release Fractions,” of the application. The staff reviewed the values in Table 4.6.1-3 and finds the values to use the bounding of the release fractions in NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,” and draft NUREG-2224, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel.”

The applicant provided the average cavity gas temperatures and pressures in Table 4.6.1-4, “TN-LC Cavity Gas Temperatures, Pressures and Properties,” of the application for NCT and HAC. The staff reviewed the temperature and pressure values for NCT and HAC and finds that the temperatures and pressures in units of psig to be consistent with Chapter 3, “Thermal Evaluation,” of the application. While the unit conversion from psig to atm.abs was incorrect, the pressure values that resulted were greater for NCT and HAC, and therefore conservative in the overall calculation. The staff also finds it to be acceptable that the 0.25 atm. abs downstream pressure was conservative, in addition to the use of the smallest seal cross-section for the hole length assumption. The staff also reviewed the fraction of fuel rods that experience cladding failure during NCT and HAC and finds those values to be the same or more conservative than the values in NUREG-1617 and draft NUREG-2224.

The applicant summarized the overall steps in Section 4.6.1.1.2 of the application to go from the containment requirements to the calculation of the reference air leakage rate and based on the staff’s review, the staff finds this to be acceptable. The staff reviewed the applicant’s calculations for releasable activity and activity density due to crud, fuel fines, volatiles, and gaseous fission products. The applicant summarized the results for NCT and HAC for the PWR fuel assembly in Table 4.6.1-5, “Activities and Activity Density per Release Type - PWR,” of the application and BWR fuel assembly in Table 4.6.1-6, “Activities and Activity Density per Release Type - BWR,” of the application. Based on the staff’s review of the applicant’s releasable activity, activity density calculations, and Tables 4.6.1-5 and 4.6.1-6 of the application, the staff finds the calculations to be acceptable.

The applicant calculated the allowable leakage rates based on the more limiting of the BWR and PWR content, for NCT and HAC in Section 4.6.1.3, “Determination of the Allowable Leakage Rates,” of the application; based on the staff’s review of those calculations, the staff finds them to be acceptable. The applicant then calculated the reference air leakage rates for NCT and HAC that the staff finds to be acceptable based on the staff’s confirmations using the equations in Appendix B of ANSI N14.5, “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment.” While the applicant should not have slightly rounded-up the reference air leakage rate acceptance criterion from 7.97·10-6 ref·$cm^3$/s to 8.0·10-6 ref·$cm^3$/s, the staff finds this to be acceptable based on the aforementioned conservatisms in the overall calculations. The applicant determined and the staff accepts that the reference air leakage rate is the more restrictive of the NCT and HAC, therefore LN = LR = 8.0·10-6 ref·$cm^3$/s, with a test sensitivity that is numerically less than 4.0·10-6 ref·$cm^3$/s.

The staff verified that the applicant updated the references throughout Chapters 4, 7, and 8 of the application, to ANSI N14.5-2014. In Chapter 7, “Package Operations,“ of the application, the staff verified that the applicant described that the O-rings were replaced within 12 months, and the assembly verification test was performed following the procedure in Section 7.4.1, “Assembly Verification Leakage Testing of the Containment Boundary,” of the application. The staff verified that the applicant described in Section 7.4.1 of the application the pre-shipment leakage rate test acceptance criterion and that it was consistent with ANSI N14.5-2014.

The staff reviewed the leakage rate test acceptance criteria for the periodic, pre-shipment, and maintenance leakage rate tests in Section 8.2.2, “Leakage Tests,” of the application and finds the values to be consistent with those described in Sections 4.2.3, “Containment Criterion,” 4.3.3, “Containment Criterion,” and Appendix 4.6.1 of the application. The staff verified the typical leakage rate test methods for the 1FA shipments described in Section 8.2.2 of the application, A.5.3 and A.5.4 of ANSI N14.5-2014 for the gas filled envelope and evacuated envelope, respectively, were capable of the acceptance criterion and test sensitivity for the periodic, pre-shipment, and maintenance leakage rate tests.

The staff also verified that the typical leakage rate test methods for the 1FA shipments described in Section 8.2.2 of the application, A.5.8 and A.5.9 of ANSI N14.5-2014 for the tracer gas sniffer techniques and tracer gas spray method, respectively, were capable of the acceptance criterion and test sensitivity for the pre-shipment leakage rate test. Based on the references to ANSI N14.5-2014 and the applicant provided updates to Chapters 4, 7, and 8 as described above, the staff finds that the application meets the containment criteria of ANSI N14.5-2014.

The staff reviewed the additional change to allow test personnel to be certified to editions of the American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” that are later than 2006, which is the edition referenced in ANSI N14.5-2014, and the staff finds this to be acceptable. The staff also reviewed the additional change to alternatively certify test personnel to International Organization for Standardization (ISO) 9712, “Non-destructive Testing – Qualification and Certification of NDT Personnel.” The applicant described the reason for the proposed alternative is that the TN-LC is used in overseas countries. The staff is not in a position to determine equivalency between Recommended Practice No. SNT-TC-1A to ISO 9712. However, the applicant highlighted the main differences in the additional change 8-A not associated with the RAIs, ADAMS Accession No. ML20304A319, and based on the staff's review of the main differences that staff finds that ISO 9712 meets the intent of ANSI N14.5-2014 to qualify and certify leakage rate testing personnel with a greater number of minimum training hours, a greater number of test questions, a greater near visual acuity, but a lower passing grade in comparison to Recommended Practice No. SNT-TC-1A.

The staff recommends that Recommended Practice No. SNT-TC-1A be used to certify leak test personnel on packages used in the United States because that is the document referenced in ANSI N14.5-2014.

### 4.3 Evaluation Findings

#### 4.3.1 Description of Containment System

The staff has reviewed the description and evaluation of the containment system and concludes that: (1) the application identifies established codes and standards for the containment system; (2) the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

#### 4.3.2 Containment under Normal Conditions of Transport

The staff has reviewed the evaluation of the containment system under normal conditions of transport and concludes that the package is designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71 (normal conditions of transport) the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system.

#### 4.3.3 Containment under Hypothetical Accident Conditions

The staff has reviewed the evaluation of the containment system under hypothetical accident conditions and concludes that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for hypothetical accident conditions, with no dependence on filters or a mechanical cooling system.

In summary, the staff has reviewed the Containment Evaluation section of the application and concludes that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71, and that the package meets the containment criteria of ANSI N14.5-2014.

## 5.0 SHIELDING EVALUATION

The purpose of the shielding review is to confirm that the package (the packaging together with its contents) meet the external radiation requirements in 10 CFR Part 71. The TN-LC package is designed to transport, among other things, commercial spent nuclear fuel in the form of an assembly or individual rods in a pin can. The certificate holder (also referred to as the applicant) has applied to revise the certificate and design of the package to incorporate various changes in the packaging components for shipment of commercial spent nuclear fuel and to specify distinct contents limits for the Unit 1 packaging, which has a minimum radial lead thickness that is less than the design-specified minimum. The staff used the guidance in NUREG-1617, the standard review plan for spent fuel packages, to conduct this review.

## 5.1 Shielding Design Description

### 5.1.1 Shielding Design Features

The package design includes lead in the package's radial side wall, lid and base for gamma shielding. It also includes a radial neutron shield. For the pin can, one optional design includes lead in the top and bottom of the pin can for additional gamma shielding for the spent fuel rod contents. Otherwise, the main components of the package that contribute to the shielding are made of stainless steel. Other components are credited in the shielding analyses as well. These components include the impact limiters, comprised of wood encased in a steel shell, and the aluminum components of the basket for commercial spent fuel. The applicant relies on the basket's aluminum components for positioning the contents in the package cavity. The applicant relies on the materials and the dimensions of the impact limiters to provide both shielding and distance. Thus, they are part of the shielding design. The applicant includes a discussion of spacers for axial positioning of the contents; however, the applicant does not describe these spacers, nor are they in any design drawing. The applicant's shielding analysis does not rely on them. The package design is described in Section 1 of the application, with design drawings contained in Appendix 1.4.1 and information specific to the design for the commercial spent fuel contents in Appendix 1.4.5.

For this revision request, the applicant made a variety of changes to the package design. A few changes are specifically for allowing the use of the Unit 1 packaging. The radial lead shielding in the Unit 1 packaging has a minimum thickness in various places that is less than the design-specified minimum. Thus, the applicant has proposed to revise the design description and drawings to include a minimum lead specification that is unique to the Unit 1 packaging. This new minimum is a little over a quarter of an inch thinner than the standard design specification.

The minimum cavity length of the Unit 1 packaging is also shorter than the standard design specification by almost a half of an inch. In the standard design, based on the nominal dimensions in the drawings, the radial lead shielding covers the full axial extent of the packaging cavity. This configuration is maintained in the Unit 1 packaging. The applicant noted that for the fabricated Unit 1 packaging, the top of radial lead shielding is slightly below the top end of the packaging cavity; however, it is consistent with design in the drawings for which the nominal dimensions show the top of the radial lead shielding and the package cavity being equal. The staff evaluated this variation from the nominal design with consideration of the other packaging changes described below. Because of the small size, the staff expects that this variation would not impact radiation levels at distance from the package side and would have only a minor impact on package surface radiation levels.

The applicant made several other changes to the package design as specified in the design drawings. These changes are applicable to the TN-LC package design in general, i.e., not just the Unit 1 packaging. The changes include removal of tolerances of some dimensions, changes in dimension designations in the drawings, removal of material or reduction of component thicknesses, modifications of components to remove materials in localized areas, removal of an option for the bottom end plug of the package's cask assembly, and changes to the pin can components.

The staff reviewed the design description and the design drawings and identified each of the changes. The staff reviewed each of the changes for potential impacts on the shielding design and performance of the package. Changes to the cask assembly of the packaging, related to the cavity length and radial lead's minimum thickness, apply only to the Unit 1 packaging, and the drawings clearly limit these changes to the Unit 1 packaging. The Unit 1 packaging may only contain a PWR $UO_2$ assembly and PWR $UO_2$ rods in a pin can. No other TN-LC package contents are allowed in the Unit 1 packaging. The staff is revising the certificate to limit the Unit 1 packaging contents in this regard. The applicant did not analyze any other contents for this particular packaging. Also, the applicant developed separate limits on burnup, enrichment, and cooling time specifications for the contents for the Unit 1 packaging which the staff will include in the certificate.

The package design had two options for the cask assembly's bottom plug. One option includes lead shielding that extends into the cask assembly's base. The other option was simply a steel plate that covered the penetration through the cask assembly's base but provided no other shielding. The applicant removed this second plug design option. The removal of this plug option only affects the commercial spent fuel rods contents in any TN-LC packaging, since the option was only used with those package contents. The commercial spent fuel rods contents are contained in a pin can. The pin can design previously had a bottom, post-like feature that contained lead and inserted into the space in the cask assembly's base that was unoccupied when the second plug design option was used. With this plug option removed, this post feature on the pin can is no longer necessary, so the applicant has removed it. Since the post feature fit into the cask assembly base, the positioning of the pin can within the packaging cavity remains unchanged and the amount of lead shielding beneath the pin can in the bottom plug area remains unchanged. Thus, the staff finds that the current shielding analysis for the pin can contents remains valid.

The applicant also modified the cask assembly design to have a different configuration for the valve for the neutron shield cavity. The valve and the housing around it now extend into and replace neutron shielding material. The staff reviewed the drawings and application regarding this change and finds the description is sufficient to determine the amount of neutron shielding material that has been replaced by the valve configuration. The amount is no more than the material replaced by the steel impact limiter attachment blocks that are in each end of the neutron shield. The shielding analysis explicitly accounts for those attachment blocks in the neutron radiation calculations. Since the valve configuration is limited in size (smaller than an attachment block), there is only one, and it would reduce gamma radiation in this location, the staff finds the change acceptable from a shielding standpoint and that the applicant's shielding analysis continues to be adequate with this design change.

In addition to these changes, the applicant proposed various changes to the impact limiters. Since the impact limiters are credited in the shielding analyses, the staff reviewed the changes for impacts to shielding. In particular, the staff evaluated materials and dimensional changes since the applicant's analysis credits the steel and wood materials and dimensions of the impact limiters. Changes included some that affect thicknesses and tolerances on dimensions. However, the impact of these changes is quite small.

The applicant also reduced the minimum density of the balsa wood and slightly increased the maximum moisture content of the wood components of the impact limiters. The moisture content change is likely to result in greater shielding from the impact limiter; however, the impact is expected to be small given the small change in this parameter. Also, the staff identified that the applicant's shielding analysis already assumes the impact limiter is made of balsa wood of the new minimum density. Thus, the analysis already covers this change. So, the staff finds the impact limiter design changes to be acceptable from a shielding perspective.

The applicant also made changes to the 1FA basket. The changes to this basket include changes to tolerances on the basket frames, addition of an optional recess and optional slots in the upper part of the basket frames, and addition of drain holes in the basket. The drain holes are small and have no bearing on radiation levels for the package due to their size.

The staff did consider there was the potential for impacts from the optional recess and slots, more so from the recess given it extended to a greater depth into the basket and was uniform around the upper portion of the basket cavity. The slots, while they go through the entire frame thickness, are still limited in width and do not extend down as far as the recess. Since these changes are not specific to the Unit 1 packaging, the staff evaluated these changes for effects on radiation levels from all the currently allowed contents of the 1FA basket. For hypothetical accident conditions, the margins to the limit are very large, so the staff only considered the effects of the changes for normal conditions. The staff focused on the limiting radiation level location, 2 meters from the vehicle side surface.

In evaluating the effects of these changes, the staff considered the axial dimensions of the fuel assemblies, the fuel rods and the pin can for the fuel rods, as the pin can has been modified in the revision request. Based on these considerations, the staff identified that the fuel rods in the pin can option with axial lead gamma shielding will remain below the affected area of the basket. So, any increases in radiation levels will be small. The recessed region of the basket will only extend into the upper hardware portions of the assembly contents and the upper plenum of the EPR rods in the pin can with the longer cavity. Thus, the radiation level impacts can be estimated by changes in transmission of cobalt-60 gammas, which would be in the activated hardware zones.

Based on the recess's width, the staff expects that the transmission rates (and so the radiation levels) will increase by about 34%. Since this increase will occur at the axial end of the package cavity, the staff applied the increase to the radiation levels at detector points consistent with that location, using the data for the axial distribution of the radiation levels at 2 meters from the vehicle side surface provided in the tables in Section 5.6.4.4.4 of the application. Based on this evaluation, the staff estimates that this increase in gamma radiation levels could cause peaks in the package radiation levels at side locations aligned with the top end of the package cavity; however, these peaks would not be large enough to become the maximum side radiation level for the package.

Even with the consideration of the reduced cavity length for the Unit 01 packaging, the staff expects the outcome to be the same. However, the staff finds that any future additional thinning of packaging components that are relied on for the shielding performance of the package will need to be analyzed to show the effects on package radiation levels.

The applicant also proposed to remove the designation of 'Stk' from the thickness dimension for the basket's top and lateral frame components which surround the basket cavity. The designation limits the under tolerance of their thickness to -0.01 inches. This limits the impact of the tolerances of the steel components in the packaging on radial radiation levels to a few percent. This is covered by the margin to the limits, even when accounting for the tolerances of other components and uncertainties in the analysis method that also increase radiation levels. Large enough tolerances could increase that impact to the point that the regulatory limits may not be met when other components' tolerances and method uncertainties are also considered.

Thus, the applicant specified a new tolerance and performed an analysis to show the impact of the new tolerances on package radiation levels, conservatively applying the effect of two times the specified tolerance. The results of that evaluation indicated that the radial package radiation levels would still be below the regulatory limits. The staff included this new tolerance in its evaluation of the shielding impact of package component tolerances and method uncertainties. The staff's evaluation indicated package radiation levels will remain below regulatory limits. Thus, the staff finds the new tolerances on the frames to be acceptable.

The applicant also modified the BWR sleeve and hold-down ring to add optional drain holes. As for the drain holes in the 1FA basket, these drain holes have no impact on packaging shielding performance.

The applicant made a number of changes to the pin can basket. These changes include the removal of the bottom post-like component described earlier. They also include: changes to the lid for lifting the pin can basket; extending the cavity length for both the long cavity (Option 1 and 2 cans, but referred to in this SER only as the Option 1 can) and short cavity (Option 3 can) options; extending the pin tube length for the Option 1 can; and modifications to the pin can's basket that cause the four corner locations to be unusable for loading fuel rods, reducing the basket maximum capacity from 25 to 21 fuel rods. The component on the lid for lifting the basket is shorter than the previous one. This together with the lengthened cavities can allow for the contents to move closer to the lid of the package. However, the impacts of this on top end radiation levels is covered by the margins to the limits.

The staff evaluated the effects of these changes on radial radiation levels in conjunction with the 1FA basket changes as described above. While the Option 1 can's basket cavity is lengthened, the length of the tubes the fuel rods are placed in is also extended by an equal amount. This ensures the steel of the tubes will continue to cover the fuel rods. While the Option 3 can's basket cavity is lengthened, the fuel tubes for this basket are not lengthened. Thus, the staff considered that this could allow for additional portions of the rods to be outside of the tubes and increase package radiation levels.

However, the drawings do include specifications about positioning of the fuel rods and limiting the total axial movement of the rods and the basket. Based on these considerations, the staff finds that the change should not result in a shielding impact. For the Option 3 can, which has lead shielding at its top and bottom, the top lead thickness is slightly reduced from before. The staff considered the effect of this slight reduction as part of the evaluations of the various packaging changes on package radiation levels described above. Other components of the pin can basket remain unchanged from before.

The applicant also modified various dimensions in the drawings to be reference dimensions only. For those dimensions that can be determined from non-reference dimensions or that appear elsewhere in the drawings as non-reference dimensions, the staff finds this designation change to be acceptable. In some cases, the needed dimensions are specified in notes on the drawing. Based on the staff's understanding, when a dimension is specified as a reference dimension and cannot be determined from other non-reference dimensions in the drawing, this can mean that the dimension is not considered important and so need not be inspected. The packaging includes components that are relied on for shielding or for maintaining the package contents' position relative to the packaging's shielding components. Dimensions for these components that are necessary to ensure these functions (shielding, positioning of contents) are important. Thus, the applicant modified the drawings further to ensure that dimensions ensuring these functions are adequately specified and are not reference dimensions. The staff reviewed the drawings and finds that the needed dimensions are adequately specified.

The applicant removed a note that was in the TN-LC cask assembly drawing and also in the pin can basket drawing that appeared to indicate that there may be an option for the CoC holder to approve packaging components with dimensions that may not meet the specifications in the drawings. While the option this note allowed may have been acceptable for package structural performance, it was not clear that the impacts on package shielding performance were considered for such an option. Thus, the applicant removed the note from these two drawings. With that option removed, the staff finds the drawings are adequately consistent with the shielding evaluation.

5.1.2 Codes and Standards

With regard to the design of the packaging, as has been the case with previous revisions to the package certificate, the applicant did not use any codes or standards. For the evaluation of the package's shielding performance, the applicant used the same computer codes and industry standard for conversion of gamma and neutron fluence rates to radiation levels that the applicant used in previous revisions. The shielding evaluation section of this section of the SER includes a description of these codes and this standard.

5.1.3 Summary Tables of Maximum External Radiation Levels

The TN-LC package has a variety of contents and multiple baskets for those contents. Tables of maximum dose rates for each basket and its contents are provided in the current revision of the application. Since the applicant did not consider that the packaging changes that apply generally to the package and its contents and the changes that apply generally to the 1FA basket and its contents impact radiation levels and shielding performance, the summary tables of maximum radiation levels for those baskets and contents remain unchanged. For the changes for the Unit 1 packaging, the applicant provided a table summarizing the maximum radiation levels for that packaging's allowable contents. That table is Table 5.6.4-66. This table includes only a summary of the maximum radiation levels for normal conditions of transport. The application does provide the maximum radiation levels for hypothetical accidents in Section 5.6.4.4.5.4.2 for the PWR $UO_2$ assembly contents and Section 5.6.4.4.5.4.4 for the PWR $UO_2$ rods contents. These are the only contents allowed in the Unit 01 packaging. Since the significant change for the Unit 1 packaging is the minimum thickness of the radial lead, only the maximum radiation level at 1 meter from the package side is provided for hypothetical accident conditions. The maximum radiation levels at 1 meter from the package's axial ends will remain bounding.

The staff reviewed the new Table 5.6.4-66 and the above-mentioned sections of the application. The reported values are below the respective radiation level limits in 10 CFR 71.47 for exclusive use and in 10 CFR 71.51(a) for hypothetical accident conditions. Using simple calculation techniques, the staff confirmed the maximum radiation level for the most limiting location (2 meters from the vehicle side) for the PWR $UO_2$ assembly contents. The values for those contents exhibit a trend versus the same contents in packagings that meet the design-specified minimum radial lead thickness that is consistent with the staff's expectations (as informed by the staff's calculations).

The application also shows values for the PWR $UO_2$ rods that are below the regulatory limits. For spots where radiation levels are not shown, the levels are bounded by those for the standard package design containing the EPR fuel rod contents. The staff also used simple calculation techniques to confirm the reported radiation levels at 2 meters from the vehicle side for the 25 PWR $UO_2$ rods contents. The proposed packaging and contents changes do not affect the axial ends in ways that would tend to increase radiation levels for those areas of the package. Therefore, the staff finds that the radiation levels for those areas of the package will continue to be bounded by the radiation levels for the standard package containing the 25 EPR rods contents. The staff's evaluation is described in greater detail in subsequent areas of this SER section.

5.2 Radioactive Materials and Source Terms

For the package design excluding the Unit 1 packaging, the package contents remain unchanged. Thus, the evaluations for those contents, including the definitions and calculation of their source terms remain unchanged and are not evaluated in this SER for the current revision request. For the Unit 1 packaging, only a subset of the allowable package contents in the 1FA basket is allowed. This is limited to the PWR $UO_2$ assembly contents and the PWR $UO_2$ rods contents.

The source terms for the contents are defined in terms of maximum burnup, minimum enrichment, and minimum cooling time. The burnup and enrichment are in terms of assembly average for the assembly contents and maximum burnup and rod average enrichment for the rod contents. Other parameters, such as maximum uranium mass also factor in, but these physical assembly and rod parameters are the same for the Unit 1 packaging as for the package design in general. While the standard package design has two separate sets of specifications of burnup, enrichment, and cooling time depending on the number of rods, the Unit 01 packaging has only one set of specifications for the rod's contents. These specifications must be met for the rod's contents, whether or not the number of rods in the package is less than the maximum number allowed.

The applicant selected a source term for each of the contents of the Unit 1 packaging that is expected to bound the source terms of the remaining assembly and rods contents, which are given in Tables 5.6.4-61 and 5.6.4-62 for the assembly and rods contents, respectively. These bounding source terms are specified in Section 5.6.4.4.5.3. The proposed allowable contents specifications for the assembly contents match those evaluated in the shielding analysis (see Table 1.4.5-8a). The proposed specifications for the rod's contents have longer cooling times than evaluated in the shielding analysis (see Table 1.4.5-10a) to ensure they are bounded by the contents approved for the standard package design.

The proposed specifications set the minimum cooling times for the rods contents to be no less than 0.05 years longer than the minimum cooling times for the rod's contents approved for the standard package design (see Tables 1.4.5-10 and 1.4.5-10a). The combinations of burnup, enrichment and cooling time are set to ensure that the radiation levels at 2 meters from the vehicle side do not exceed 8 mrem/hr for the assembly contents and 7.69 mrem/hr for the rod's contents, as determined using the applicant's response functions. Limiting the radiation levels to these values provides margin to the regulatory limit that can compensate for the uncertainties in the applicant's shielding analysis methods. The maximum radiation levels for the assembly and rods contents are given in Table 5.6.4-66 and exceed the above criteria, but they are also calculated in a different manner to identify the peak radiation levels around the package at 2 meters from the radial surface.

The applicant calculated the gamma source and spectrum and the neutron source for each burnup, enrichment and cooling time combination using the same codes and techniques used for the currently approved contents. The proposed contents specifications for the Unit 1 packaging contents only differ in cooling time. The evaluated burnups and enrichments remain the same. Thus, the staff finds that the methods remain acceptable to use in calculating the gamma and neutron sources for the Unit 1 packaging contents.

The staff evaluated the applicant's selection of the bounding assembly and rods contents' source terms. The staff used the ARP and ORIGEN modules of SCALE 6.2.3 to perform some comparisons to inform its review. Based on this review, the staff finds that the selected assembly source terms for normal conditions and for hypothetical accident conditions will bound the source terms of the other assembly contents for their respective conditions. The staff finds the selected rods source terms to be bounding for hypothetical accident conditions and reasonably bounding for the normal conditions.

The staff's evaluation indicated that other burnup, enrichment, and cooling time combinations in Table 5.6.4-62 might actually result in higher radiation levels; however, the difference was small (about 1%) and would be compensated by the margins to the limits, accounting for other method uncertainties, packaging tolerances, and conservatisms. Thus, the use of the applicant's selected source term is acceptable and reasonably bounding.

The staff also reviewed the normal conditions and hypothetical accident conditions' source terms for the Unit 1 packaging's rods contents and confirmed that they appropriately represent a source term from 25 PWR $UO_2$ rods, as opposed to a full assembly. As part of confirmatory calculations, the staff calculated the source strength and spectrum for the active fuel zone of the assembly. Scaling that source to 25 fuel rods, the staff got a reasonably similar source strength and spectrum to that reported by the applicant for both the normal conditions and hypothetical accident conditions' sources.

## 5.3 Shielding Model and Specifications

For the changes that apply to more than just the Unit 1 packaging, the applicant continues to rely on the existing analyses with the existing shielding model and specifications to demonstrate compliance with the regulatory radiation level limits. Based on a review of the changes to the packaging components as described in the SER section on shielding design features, the staff finds that to be acceptable. However, as stated in that SER section, future additional changes to packaging properties (e.g., thinning of components credited for shielding) should be evaluated to demonstrate the radiation level limits will not be exceeded, performing calculations with revised models as needed.

The staff identified that the applicant only modeled the radial lead shielding at its minimum thickness. This means the applicant modeled the lead in the package lid and base at their nominal dimensions. The staff notes that the application includes a study for the sensitivity of the radiation levels at the package's bottom axial end to the lead thickness in the package base (nominal vs. the originally specified minimum thickness).

However, this study did not use the contents with the highest radiation levels for this part of the package. It also did not include the tolerances of the steel components in the package base. For these reasons, the staff finds it does not provide an indication as to whether radiation levels for all package contents would exceed regulatory limits when tolerances are considered, for both lead and steel components, for this part of the package.

The staff performed some simple calculations that showed the radiation levels would increase by about 20% at the package ends with the lead in the lid and base modeled at the minimum thicknesses originally specified for the drawings for each component. For the top end of the package, this presented no issue because there is significant margin to each of the limits at the top of the package. However, for the bottom end of the package, this was not the case. As shown in Table 5.6.4-1 of the application, the radiation level on the package surface and the vehicle surface at the bottom end of the package (taken to be the same location) is 178 mrem/hr. The table also shows that the radiation level is due almost entirely to gamma radiation. It also shows that the BWR assembly contents result in the bounding radiation levels at the bottom end of the package.

A 20% increase in the radiation level at the bottom end package surface results in a radiation level above 200 mrem/hr, which is the limit for the vehicle surface. It is also the limit for the package surface since it is the same location and, if this package is transported as other spent fuel packages are, the impact limiters are not in an enclosure. To alleviate this issue, the applicant revised the packaging design to increase the minimum thicknesses of the lead in the packaging lid and base, including in the plug in the bottom end of the packaging, to be no less than 0.01 inches less than the nominal thicknesses. The staff estimates that accounting for the new minimum lead thicknesses would only result in an approximately 1.5 percent increase in the radiation levels at the package axial ends.

Tolerances on the thicknesses of the steel components of the lid and base also have an influence on radiation levels at the package’s axial ends. The applicant revised the package drawing to specify a minimum total thickness for these steel components for both the lid and the base. The minimum total thicknesses allow for a total steel thickness that is no more than 0.1 inches less steel versus nominal for the lid and for the base.

The staff estimates that accounting for this difference in steel thickness together with the lead at its specified minimum thickness would increase radiation levels by almost 12 percent at each of the package’s axial ends. Even with this increase, the radiation levels at the axial bottom surface of the package remain below the 200 mrem/hr limit, though there is not much margin.

Thus, the staff finds reasonable assurance that the radiation levels for this package surface and vehicle surface will not exceed the regulatory limit. The staff also finds reasonable assurance that the radiation levels at 2 meters from the vehicle side surfaces are still appropriate to use to determine the allowable package contents, including for BWR fuel. However, the staff notes that contents or packaging changes in future CoC revision requests may result in the need to reevaluate the radiation levels for this part of the package and confirm that the above staff findings remain valid.

The model and specifications for the Unit 01 packaging are the same as for the standard TN-LC packaging, with the exception of the radial lead being at a thinner minimum thickness (3.10 inches versus 3.38 inches). The staff notes that the Unit 01 packaging’s cavity is also a little shorter than the minimum specified in the design drawings for the standard TN-LC packaging. This would result in packaging contents being closer to the package lid and thus result in some increase in radiation levels at the top end of the package.

However, as noted previously, the margins to the limits at the top end are significant and are more than adequate to compensate for any increase in radiation levels arising from the change in proximity of the package contents to the package lid. Based on these considerations together with the considerations noted for the other packaging component changes (e.g., see the SER section on shielding design features), the staff finds that the model described in the application for the Unit 01 packaging is acceptable.

## 5.4 Shielding Evaluation

### 5.4.1 Methods

The applicant used the same methods and codes for the analyses for the Unit 1 packaging as the applicant used for analyzing the currently approved packaging design and contents. Thus, the types of input and output data remain unchanged. The applicant continued to use the staff-accepted conversion factors for converting fluence rates to radiation levels (i.e., the conversion factors in ANSI/ANS 6.1.1-1977).

Radiation level calculations involved the use of the MCNP code, as before. However, the applicant's method is not to use MCNP to calculate radiation levels for every proposed combination of burnup, enrichment, and cooling time for the proposed contents. As before, the applicant used MCNP to create response functions, which are radiation levels at a particular point or surface location per starting particle of a particular energy bin for a particular geometry and material properties of the packaging and the contents. For the spent fuel contents' neutron source, the spectrum is dominated by the curium nuclides for which the spectrum can be represented easily and simply in MCNP.

Thus, for the neutron source, the response function is for the total neutron source versus a response function for each energy bin of the neutron spectrum. This approach simplifies the radiation level calculation process by allowing for the gamma source in each energy bin to be multiplied by its respective response function and the results summed over the gamma spectrum. The total neutron source is also multiplied by the response function for neutron radiation levels and for radiation from secondary gammas, the results of which are summed with the gamma radiation level to obtain the total radiation level.

This process is applied for each location around the package that is needed to demonstrate compliance with the regulatory limits for both normal conditions of transport and hypothetical accident conditions. In the applicant's analysis, as before, the radiation levels at 2 meters from the vehicle side (at the axial center of the contents' active fuel region) are the most limiting of the radiation level locations around the package. So, the applicant developed response functions for this location.

The applicant used the response functions to generate the table of allowed burnup, enrichment, and cooling time combinations, based on a target radiation level, which is some value below the regulatory limit (e.g., 8 mrem/hr for PWR $UO_2$ assemblies in the Unit 1 packaging). The applicant also used the response functions to rank the contents by radiation level to identify the bounding source term. The applicant then used this bounding source term in a full MCNP calculation to confirm radiation levels will not exceed the regulatory limits.

For the Unit 1 packaging and its contents, the applicant provided the response functions for the gamma spectra of the PWR $UO_2$ fuel assembly contents and the PWR $UO_2$ fuel rod contents in Tables 5.6.4-59 and 5.6.4-60, respectively. Since the lead shielding thickness is the only difference between the Unit 01 packaging and the standard packaging design, the neutron response functions for the assembly contents and fuel rods contents for this location will not change. Although the design changes reduce the pin can basket's capacity from 25 fuel rods to 21 fuel rods, the applicant continued to perform the analysis for the basket containing 25 fuel rods. Thus, the previously developed neutron response functions for these contents were used for the Unit 1 packaging analysis.

For the calculation of radiation levels for the spent fuel's gamma source, as done previously, the applicant only created response functions for four of the energy bins in the gamma spectrum, those that have a maximum energy of 1.33, 1.66, 2.5, and 3.0 MeV. The applicant stated that these four energy bins account for 95 percent of the total gamma radiation level. In reviewing the analysis, including the source terms, for the Unit 01 packaging, the staff considered that these four energy bins may not always account for 95 percent of the gamma radiation levels. In looking at the 25 fuel rod source term in Table 5.6.4-64 for normal conditions, the source in the 2.0 MeV energy bin was as large as the source in the 3.0 MeV energy bin. The sources in the 0.8 and 1.0 MeV bins were much larger. For the hypothetical accident source, the 2.0 MeV source was larger than the 3.0 MeV source, with the 0.8 and 1.0 MeV energies' sources being significantly larger. The staff identified a similar pattern for the fuel assembly source terms in Table 5.6.4-63.

The staff also performed some simple calculations in MicroShield (v. 12.00) for a few different source terms to evaluate the relative contribution of different energy bins. The models used shielding geometries (thicknesses) that represent the configuration of the pin can in the packaging with a detector at 2 meters from the vehicle's side surface. The results of this evaluation showed that the four energy bins selected by the applicant do contribute significantly to package radiation levels. However, there are cases where one or two of the energy bins contribute very little and where the total contribution of the four selected bins is less than 90 percent of the total primary gamma radiation level. Also, other energy bins (0.8, 1.0, and 2.0 MeV bins) contribute significantly to radiation levels in various cases, contributing as much or more than at least the 3.0 MeV bin.

The staff notes that the review guidance in Section 5.5.2.1 of NUREG-1617, the standard review plan for spent fuel packages, indicates that gammas of energies between 0.8 and 2.5 MeV contribute significantly to gamma radiation levels for typical packages. The staff's evaluation indicates this is so for the TN-LC Unit 1 packaging and for the TN-LC package design in general. Even applying the applicant's own information regarding design basis gamma source terms, response functions, and analyses that use the full spectrum, the staff identified instances where these four energy bins captured less than 90 percent of the primary gamma radiation levels (e.g., the functions for 25 EPR rods capture less than 90 percent of those rods' primary gamma radiation levels).

As noted above, however, the applicant uses the response functions to identify the bounding source term. The applicant uses that source term in a full MCNP calculation to demonstrate compliance with the regulatory radiation level limits. The applicant uses the full gamma spectrum of the design basis source in that MCNP calculation. Thus, any influence of the response functions capturing more or less of a source's primary gamma radiation is limited to the selection of the bounding source term.

As also noted above, for the allowed and proposed contents that the staff reviewed, the staff found that the applicant's selected sources were adequately bounding. In one instance other sources may result in slightly higher radiation levels (about 1 percent). The staff expects that a review of the package's remaining commercial spent fuel contents would show a similar result.

While the applicant's analysis results show margins to the regulatory limits, these margins are also needed to compensate for various aspects of the applicant's analysis method that can underpredict the radiation levels (at 2 meters from the vehicle side). These aspects include the use of nominal dimensions of the packaging's steel components instead of their minimum design dimensions, azimuthal variations in the contents and packaging components that result in higher radiation levels, and impacts of how the neutron shielding was modeled (including use of nominal versus minimum thickness and other modeling aspects). The outcome of the staff's evaluation of the bounding source terms adds a little more to that (about 1 percent).

The staff evaluated the effects of these aspects of the analysis, applying them to the radiation levels shown in Table 5.6.4-66 for the Unit 1 packaging. The staff also considered the impacts of the relevant analysis aspects on the radiation levels for the standard package design in Table 5.6.4-32. The staff recognized that the applicant also calculated radiation levels using mesh tallies and that, based on information in the application, the radiation levels shown in Tables 5.6.4-32 and 5.6.4-66 are the maximum total radiation levels at 2 meters from the vehicle side, including consideration of the mesh tally results. Since the mesh tallies would, by their nature, account for azimuthal variation of the radiation levels around the package, the staff determined that the applicant's results in these two tables already adequately account for that variation.

Thus, the staff did not include the effect of that variation in its evaluation since doing so would mean that the effect was included twice. The staff's evaluation indicated that the radiation levels for the standard package design will remain below the regulatory limits with the proposed packaging design changes, though the margins for some of the package contents are quite small. The staff's evaluation also indicated that the radiation levels for the Unit 1 packaging and its contents will not exceed the regulatory limits; however, the evaluation for the Unit 1 packaging also had to account for the analysis's conservatism in calculating radiation levels for 25 rods (versus the maximum capacity of 21 rods) to show regulatory limits would not be exceeded for the rods contents. The staff's evaluation indicated that the margins to the limits for the Unit 01 packaging and its contents are also quite small.

As noted previously, changes to the pin can basket have reduced the number of fuel rods it can hold from 25 to 21. The applicant has not changed the analyses for the pin can basket and its contents. For the Unit 1 packaging analysis, the applicant continued to model the pin can basket as containing 25 fuel rods. This is acceptable because the analysis uses the source for 25 fuel rods, which will bound the source strength for 21 rods.

While the lack of the four rods (from the four corner spots in the pin can basket) also results in less shielding of the interior rods, the staff expects that the reduction in source term will have at least as large of an impact on radiation levels as will the loss of shielding for the interior rods in these areas such that any difference in radiation levels will be a reduction in radiation levels on the package surface and at the limiting 2 meter location.

5.4.2 External Radiation Levels

In reviewing the analyses for the Unit 1 packaging and its proposed contents, the staff considered the changes in the packaging's radial lead shielding and the source terms used for the analysis. The staff estimated that the reduction in the minimum radial lead thickness from the standard TN-LC design specification to that specified for the Unit 1 packaging would result in an approximately 43 percent increase in package gamma radiation levels. While the bounding source terms for the proposed Unit 1 packaging contents are less than the design basis source terms for the standard TN-LC packaging design, it was not immediately obvious that the difference was enough to compensate for the reduction in the radial lead's minimum thickness.

The staff used simple calculations that involved ratios of the Unit 1 packaging source term to the design basis source term. The staff also used and ratios of the transmission through the packagings' total half-value thicknesses for each energy bin at each respective packaging's (Unit 1 vs. standard design) minimum radial lead thickness. Using a MicroShield calculation to determine the relative contribution to package radiation levels from each bin, the staff weighted the product of the source term and transmission ratios of each energy bin by the fraction it contributed to package radiation levels. For energy bins where the result was overly large, but gammas of that energy contribute negligibly to package radiation levels, the staff ignored the results of those bins in the summation of the results over the gamma spectrum. Since the margins to the hypothetical accident limit are substantial and the radiation level limit at 2 meters from the vehicle side for normal conditions is limiting, the staff evaluated radiation levels for this latter case.

The staff used that total relative change in gamma radiation levels and the difference in neutron source strengths between the design basis and the Unit 1 packaging sources to adjust the radiation levels for the PWR $UO_2$ assemblies in the standard package design. The staff's adjusted radiation levels provided an estimate for the radiation levels for the Unit 1 packaging, which were consistent with the applicant's calculated values for the 2-meter side location in Table 5.6.4-66. The gamma radiation levels increased for the Unit 01 packaging versus the standard package design, by about 40 percent. However, for the PWR $UO_2$ assembly contents, the neutron radiation levels are substantial for the package. The decrease in the neutron source, and the associated decrease in neutron and secondary gamma (from neutron interactions with packaging materials) radiation levels was enough to compensate for the increase in gamma radiation levels.

The staff considered that for the hypothetical accident conditions a 43 percent increase in the applicant's calculated radiation levels in Table 5.6.4-2 would remain below the regulatory limit with significant margin remaining, showing that the design basis PWR $UO_2$ assembly source term in the Unit 1 packaging will not exceed that limit. Therefore, the Unit 01 packaging with its proposed PWR $UO_2$ assembly contents with a smaller source term will not exceed the radiation level limit for hypothetical accident conditions.

Thus, based on a review of the applicant's analyses and the results of the staff's own simple calculations, the staff finds reasonable assurance that the Unit 1 packaging with its proposed PWR $UO_2$ contents will have radiation levels that do not exceed the regulatory limits. The staff's finding also includes consideration of aspects of the applicant's analysis (e.g., steel components at nominal thicknesses vs. the minimum dimensions allowed by the design) that would increase the package radiation levels. The margins to the limits continue to be sufficient to compensate for the effects of those aspects of the analysis method and package design, though residual margins will be quite small.

The staff performed simple MicroShield calculations to evaluate the proposed Unit 1 packaging's PWR $UO_2$ fuel rods contents. The staff calculated radiation levels at 2 meters from the vehicle side for the normal conditions' design basis gamma source in the standard package design. The staff did the same calculation for the bounding normal conditions' gamma source for the Unit 1 packaging design. The result was a relative increase of the gamma radiation levels by almost 24 percent.

Using that relative change, the staff adjusted the radiation levels for the design basis source in the standard design to estimate the radiation levels for the Unit 01 packaging with its bounding source. Accounting for the differences in neutron source strengths and associated neutron and secondary gamma radiation levels, the staff calculated a radiation level that was similar to the applicant's calculated value in Table 5.6.4-62a.

As previously noted, the staff identified a concern with the BWR assembly contents and their radiation levels in the standard package design. The applicant's Table 5.6.4-1 shows the radiation levels for the bottom end are based on the BWR assembly contents. The package surface and vehicle surface are the same surface and so have the same limit, i.e., 200 mrem/hr. The radiation levels in Table 5.6.4-1 are based on calculations that use the nominal thicknesses of the lead and steel components in the cask assembly's base.

However, the staff estimated that using the originally specified lead minimum thickness increased those radiation levels by about 20 percent and resulted in the regulatory limit being exceeded. Accounting for the steel components' tolerances would further increase the radiation levels. However, with the newly specified minimum lead thickness for the package (cask assembly) base and the newly specified minimum total thickness of the steel components in the package base, the staff finds reasonable assurance that the radiation levels will not exceed the regulatory limit for the allowed BWR contents, though the margin is quite small.

### 5.4.3 Confirmatory Calculations

As previously described, the staff performed a variety of confirmatory calculations as part of this review. These calculations involved use of the ORIGEN and ARP modules of the SCALE 6.2.3 code system for calculating gamma and neutron source terms. They also involved use of MicroShield calculations to determine relative radiation level changes and contributions of different portions of the gamma spectrum to total gamma radiation levels.

The staff's calculations also included simple hand calculation techniques, estimating relative changes in shielding capability and radiation levels with half value thicknesses for packaging materials. With these methods, the staff performed evaluations to confirm the applicant's selection of bounding source terms, evaluate changes in radiation levels between the standard package design and the Unit 1 packaging, evaluate impacts of packaging tolerances and other analysis method aspects, and determine radiation level estimates for compliance with the regulatory limits.

### 5.5 Evaluation Findings

Based on a review of the information and representations provided in the application and the staff's confirmatory calculations, the staff has reasonable assurance that the TN-LC package and its contents, including the proposed Unit 1 packaging and its contents, satisfy the shielding requirements and limits in 10 CFR Part 71.

## 6.0 CRITICALITY EVALUATION

The purpose of the criticality review is to confirm that the package together with its contents meet the requirements in 10 CFR Part 71 for criticality safety. The TN-LC package is designed to transport, among other things, commercial spent nuclear fuel in the form of an assembly or individual rods in a pin can. The certificate holder (also referred to as the applicant) has applied to revise the certificate and design of the package to incorporate various changes in the packaging components for shipment of commercial spent nuclear fuel, to add specifications for a new class of spent fuel assemblies (the WE16x16 class), and to specify distinct specifications for the Unit 1 packaging. The staff used the guidance in NUREG-1617, the standard review plan for spent fuel packages, to conduct this review.

## 6.1 Packaging Component Design Changes

The applicant has proposed several modifications to the packaging components. These changes include items such as removal of or changes to the tolerances of various packaging components' dimensions, modifying designations of dimensions, reduction of component thicknesses, localized removal of materials, and changes to the pin can basket. This section describes the staff's evaluation of these changes.

The changes include modifications to the cask assembly and the impact limiters. These changes had the potential to affect all the TN-LC contents. In its review, the staff identified that all the changes to the cask assembly, with the exception of the removal of the optional bottom plug configuration, did not have any effect on criticality safety due to the nature of the changes. The removed option for the bottom plug only affected the commercial spent fuel rods in the pin can basket. However, the removal of this option was accompanied by removal of the bottom, post-like feature from the pin can basket.

Thus, the potential for an effect on criticality safety as a result of changes in axial positioning of the contents relative to the radial packaging components credited in the criticality analyses was alleviated; there is no change in axial positioning of contents versus packaging components. Thus, from a criticality safety perspective, the staff finds the changes to cask assembly have no impact on criticality safety and are acceptable.

The modifications to the impact limiters included some tolerance changes and materials specifications changes among others. These changes also had the potential to affect all the TN-LC contents. In its review, the staff identified that none of the criticality analyses for any of the package contents credited the impact limiters in any way, with the exception of the normal conditions of transport array for the MTR contents.

In that analysis, the applicant credits the spacing provided by the impact limiters between packages. In its review of the impact limiter drawing, the staff determined that none of the changes affected the spacing between packages provided by the impact limiters. Thus, from a criticality perspective, staff finds the changes to the impact limiters have no impact on criticality safety and are acceptable.

The remaining changes affect only the 1FA basket, the BWR sleeve and hold-down ring, and the pin can basket. These components are used only with the commercial spent nuclear fuel contents that can be shipped in the package. The BWR sleeve and hold-down ring is the component with the fewest changes. The changes for this component include the option for having drain holes. These would be very small, localized changes. Furthermore, the applicant's criticality analysis evaluates for optimum moderation conditions within the package, including non-uniform flooding of the package. Given their expect size and that they would only help to ensure more uniform flooding or draining of the package cavity, the staff finds these changes have no impact on the criticality safety function of the package and so are acceptable.

The changes to the 1FA basket include optional drain holes, an optional recess in the top end of the basket frames (top and lateral), optional slots in the top end of the basket, changing dimensions of the basket cavity (formed by the top and lateral frames) and the width of the frames to be reference dimensions, and changing the specified tolerances for the frames' thickness. The staff's evaluation of the drain holes in the BWR sleeve and hold-down ring applies to the drain holes in the 1FA basket. The optional slots are limited in area, but they occur in the top of each frame and extend up to two inches down into the basket cavity. They penetrate through the entire thickness of the frames but will not result in removal of neutron absorber plate material attached to the exterior surfaces of the frames. The optional recess uniformly removes 0.25 inches from the interior surfaces of the basket's frames and extends up to three inches down into the basket cavity. The staff considered that these losses of materials had the potential to affect the criticality safety function of the package for the commercial spent fuel contents and necessitate changes to the criticality analysis.

In evaluating the potential effects of the optional slots and recess, the staff considered the axial dimensions and positioning of the fuel assembly, including end hardware, and fuel rod contents relative to the locations of these component changes. The staff also considered the applicant's analysis for the effect of vertical positioning of the BWR assembly within the BWR sleeve and hold-down ring. The hold-down ring sits above the sleeve, has a larger interior cavity, and has thinner walls than the sleeve. It also extends further down into the basket cavity than the optional slots and recess in the 1FA basket frames.

The applicant's analysis showed there was no effect on reactivity for the BWR assembly positioned with a portion of the assembly in the hold-down ring axial zone versus being entirely within the sleeve axial zone. For the assembly contents, the staff evaluation indicated that the notches and recess could not extend into the active fuel zone of the assemblies regardless of the axial movement allowed. For the fuel rods contents, the only contents of concern were the EPR rods since only these rods are allowed in the pin can with the longer cavity (i.e., Option 1 and 2 cans, referred to only as Option 1 can). The short cavity pin can (i.e., Option 3 can) includes several inches of lead in the top that precludes the fuel rods from reaching the axial location of the slots and recess.

The staff determined that the EPR rods' plenum zone length ensured that the rods' active fuel zone could not extend into the axial zone of the slots and recess in the 1FA basket frames. In the criticality analyses, the applicant modeled anything beyond the axial length of the active fuel as water. For the infinite arrays for the BWR assembly contents and fuel rods contents, the applicant placed a reflective boundary at the axial ends of the model (at the ends of the active fuel). Thus, based on these considerations, the staff finds that the optional slots and recess have no impact on the package's criticality safety function and so are acceptable.

The staff considered the proposed change to the tolerance for the 1FA basket frames' thickness and the changing of the frames dimensions and cavity dimensions to reference dimensions. Since the drawing specifies the cavity, or compartment, minimum dimensions in a note in the drawing and the thickness of the frames is shown as a regular dimension (i.e., it has not been changed to a reference dimension), the staff finds changing the dimensions of the frames and cavity to reference dimensions to be acceptable; there is no effect on the package's criticality safety function. The tolerance on the thickness of the frames can be important to the criticality analysis, affecting the amount of neutron absorption by these steel components.

The applicant's criticality analysis currently includes an evaluation for a specific tolerance for the frames' thickness. Thus, though the initial proposal was to remove the tolerance, the applicant specified a new tolerance for the frames' thickness. This tolerance is the same as that which the applicant used in the package criticality analysis demonstrating the impact of packaging tolerances. Thus, the staff finds the newly specified tolerance to be acceptable for ensuring the packaging performs its criticality safety function as analyzed.

The changes to the pin can basket include: the previously described removal of the base, post-like component; modifications to the lid, particularly the components for lifting the pin can basket; modifications that preclude loading of fuel in the four corner tubes of the 5x5 tube array; lengthening of the basket's cavity in both the Option 3 and Option 1 cans; and extending the fuel tube length for tubes in the Option 1 can. The changes to the lifting components and lengthening of the pin can basket's cavity length can allow for greater axial movement of the pin can basket and the fuel rod contents. This is a potential criticality safety concern in so far as the axial location changes the packaging components credited in the criticality analysis that result in keeping the package subcritical.

Thus, staff included these changes to the pin can basket in its evaluation of the optional slots and recess being added to the 1FA basket described previously. The modification that reduces the capacity of the pin can basket reduces the quantity of fuel in the basket and allows for more water to be in the basket. However, the staff notes the maximum package reactivity is 0.5452 (application Table 6.10.4-1), which is significantly below the upper subcritical limit of 0.9420. Thus, even if this change was to result in an increase in reactivity, the package will still be significantly subcritical.

Also, the steel fuel tubes are important as they absorb neutrons and help to reduce reactivity. With the lengthening of the cavity of the pin can basket, there could be the potential for more of the fuel rods to move outside of the fuel tubes, reducing the tube's effect on reactivity. The applicant extended the length of the fuel tubes in the Option 1 can's basket by approximately the same length as the extension of the can's cavity; thus, for the Option 1 can, the potential concern is alleviated. For the Option 3 can, the fuel tubes' length remains as before. Thus, the potential may exist for more of the fuel rods in the Option 3 can to be outside of the can's basket's fuel tubes. In evaluating the potential concern for the Option 3 can, the staff again noticed the significant margin between the current maximum reactivity for these contents and the upper subcritical limit. While the package reactivity could be increased for the contents in the Option 3 can, the staff considers that the current margins more than compensate for that potential increase and the package will still be significantly subcritical.

Beyond the analyses supporting previous CoC revisions, the applicant performed a few additional calculations to support the conclusions of those previous analyses. However, those analyses relate to only a limited number of the packaging changes proposed in the current revision request. Additionally, the applicant proposed to reduce the maximum allowed enrichment for the CE15x15 PWR assembly contents for which the applicant's analysis indicates that k-effective ($k_{eff}$) is maximized.

The applicant reduced the enrichment slightly from 3.7 weight percent to 3.6 weight percent. This reduction results in reduced $k_{eff}$ values and more margin to the upper subcritical limit.

Based on this increased margin and the staff's own review of the changes as described in this SER section, the staff finds that the package's criticality safety function is maintained with the proposed changes to the packaging components.

## 6.2 Package Contents Changes

The applicant proposed to add a new PWR assembly class to the allowed package contents in the 1FA basket. This class is referred to as the WE16x16 class, based on the Westinghouse 16x16 assembly design. The applicant proposed assembly specifications for this design that are identical to the WE17x17 assembly class contents, with the exception of having fewer fuel rods and guide tubes. The specifications include identical specifications for poison rod assembly rods (number, diameter, $B_4C$ content, and positions within the assembly). Based on these specifications, the applicant stated that the WE16x16 assembly class would be bounded by the WE17x17 assembly class. The staff reviewed the specifications of the new assembly class and confirmed that those specifications important for criticality safety (e.g., maximum pellet outer diameter, maximum rod pitch, minimum cladding thickness, minimum cladding outer diameter) are the same. With the same requirements for poison rod assembly absorber rods, these specifications would tend to result in the WE16x16 class assembly being bounded by the WE17x17 class assembly. The staff also performed a few simple calculations using SCALE 6.2.3's CSAS6 criticality code sequence. The results of those calculations confirmed this condition. Thus, the staff finds that the package with WE16x16 assembly class contents as either an assembly or as individual rods will be subcritical.

The staff also noticed that the applicant proposed a change to the minimum cladding outer diameter specified for the CE16x16 PWR contents. This diameter is less than the previously accepted and analyzed diameter, which will have the effect of increasing reactivity. The applicant therefore performed an additional calculation for the design basis poison rod assembly configuration for which the $k_{eff}$ results in Table 6.10.4-21 apply. The applicant determined that $k_{eff}$ did in fact increase with the smaller diameter. The analyzed configuration, however, is not the configuration upon which the final $k_{eff}$ results for demonstrating subcriticality are based. Those results are given in Table 6.10.4-33 and are higher than the values in Table 6.10.4-21.

Since the results in Table 6.10.4-33 are from calculations with the SCALE code on a computer with a different operating system, a factor is applied to the values in Table 6.10.4-33 to adjust for that difference, which yields the $k_{eff}$ values that are used to demonstrate subcriticality. The staff finds that the applicant should have done the new calculation for the reduced cladding outer diameter for the configuration used for Table 6.10.4-33. However, even applying the increase that the applicant identified in Table 6.10.4-21 to the $k_{eff}$ for the CE16x16 assembly in Table 6.10.4-33 with the adjustment factor for the difference in computer system, the end result would still be subcritical with margin. However, with the applicant's proposed reduction in allowed maximum enrichment for the CE15x15 assembly contents, the CE16x16 assembly contents may now result in the maximum $k_{eff}$ for the package containing a commercial PWR assembly.

As noted above, the applicant also reduced the maximum enrichment of the CE15x15 assembly contents from 3.7 weight percent to 3.6 weight percent. Since this reduction in enrichment leads to a decrease in $k_{eff}$ and increases the margin to the upper subcritical limit, the staff finds the change to be acceptable. The staff notes, however, that the overall increased margin may not be as significant as the results of that change would indicate because, as described above, the changes for the CE16x16 assembly content may mean that the CE16x16 assembly now results in the maximum $k_{eff}$.

6.3 Unit 1 packaging

The Unit 1 packaging differs from the standard TN-LC design in two ways. One is that the cavity length is less than the minimum specified in the design drawings for the standard design. The second is that the minimum thickness of radial lead in the cask assembly wall is less than the minimum specified in the design drawings for the standard design. The applicant has also limited the Unit 01 packaging contents to only allow PWR $UO_2$ assembly contents and PWR $UO_2$ fuel rods contents.

Since the applicant only models the packaging portions that are within the active fuel axial zone, the cavity length has no effect on the criticality safety function of and analysis for the package.

If anything, it could tend to restrict any potential axial movement of the package contents more so than in the standard packaging design. For the BWR assembly contents, for example, the applicant has analyzed the effect of axial positioning since the package configuration for this content includes a sleeve and hold-down ring, with the hold-down ring in the upper portion of the basket being a thinner component than the sleeve. That analysis showed that the axial positioning of the BWR assembly had no impact on package reactivity.

With regard to the lead thickness, while this could in theory result in closer proximity of the contents of adjacent packages in package arrays (due to thinner packaging walls), the thinner areas in the Unit 1 packaging's radial lead are described as being localized versus being uniform throughout the radial lead shielding. Even so, the applicant did a calculation to show the impact of uniform thinning of the lead shielding. Table 6.10.4-11 shows the impact to be negligible. Thus, the staff finds that there would be no effect from the thinner lead shielding in the Unit 1 packaging on the package's criticality safety function or the analysis for it. Thus, the staff also finds it acceptable that the application does not include an evaluation of the Unit 1 packaging changes in terms of criticality safety.

6.4 Licensing Basis

In the course of its review, the staff identified that the application (in Section 1.1.2.1) specifies that the licensing basis for the package in terms of criticality includes evaluation of fuel reconfiguration for both normal conditions of transport and hypothetical accident conditions. This is to be done for both research and commercial spent fuel.

For the commercial spent fuel criticality analyses in Appendix 6.10.4, the staff identified that fuel reconfiguration was only analyzed for the hypothetical accident condition analyses. Since in all other respects (e.g., optimum internal moderation, credited packaging components, tolerances) the criticality models are the same for the normal conditions and hypothetical accident conditions, the staff identified that following the licensing basis approach for the package array analyses would result in the normal conditions array exceeding the upper subcritical limit unless the array size was reduced to the same size as the hypothetical accident array (i.e., one package). In doing so, per 10 CFR 71.59, the value of 'N' for the normal conditions array would be less than 0.5, which is precluded by the regulation. Thus, the array is three packages.

So, following the licensing basis would result in a package that cannot be shown to be subcritical for the minimum normal condition array size given the current modeling assumptions for that array. Therefore, the staff determined that something needs to be done to rectify this situation. The applicant's analysis needs to be modified in a way that follows the applicant's own stated licensing basis and shows that the normal conditions array will be subcritical. Modifications to the array analysis will need to have appropriate justification and still use a model that is consistent with or bounding for the impacts of the normal condition tests in 10 CFR 71.71.

The staff finds that the current normal conditions array model includes very conservative assumptions that when modified in a manner that would still be consistent with or bounding for the normal conditions tests' impacts would result in the three-package normal conditions array being subcritical.

Therefore, the staff finds that this is not a safety concern that requires immediate correction. However, this should be addressed in the next revision request that involves changes to the criticality analysis; if not resolved as part of the revision request submittal, the staff will request that the issue be resolved through the RAI process. The applicant is aware of this concern and understands that it needs to be addressed.

6.5 Evaluation Findings

Based on a review of the information and representations provided in the application and the staff's confirmatory evaluations, the staff has reasonable assurance that the TN-LC package and its contents, as modified in the application, satisfy the criticality safety requirements in 10 CFR Part 71.

## 7.0 PACKAGE OPERATING PROCEDURES

The application provides a description of package operations, including package loading and unloading operations, and the preparation of an empty package for shipment. Loading and unloading procedures show a general approach to perform operational activities because site-specific conditions may require the use of different equipment and loading or unloading steps.

The staff reviewed the Operating Procedures in Chapter 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation. Several changes were proposed and justified by the applicant on the cask preparation before loading, wet loading, preparation before transport, cask unloading, as well as assembly verification leakage testing of the containment boundary.

Since the revision request adds a specific packaging unit design, the Unit 1 packaging, with limits on the contents that can be shipped in that packaging, the package operations have been updated to ensure that only the contents analyzed and approved for shipment in the Unit 1 packaging will be loaded into it. Based on its review, the staff finds the package operations as modified are sufficient to ensure the package is operated consistent with its design and in a way that ensures the package will meet the shielding and criticality safety requirements of 10 CFR Part 71 during actual transportation activities.

The staff verified that the applicant updated the references throughout Chapters 4, 7, and 8 of the application, to ANSI N14.5-2014. In Chapter 7, "Package Operations," of the application, the staff verified that the applicant described that the O-rings were replaced within 12 months, and the assembly verification test was performed following the procedure in Section 7.4.1, "Assembly Verification Leakage Testing of the Containment Boundary," of the application. The staff verified that the applicant described in Section 7.4.1 of the application the pre-shipment leakage rate test acceptance criterion and that it was consistent with ANSI N14.5-2014.

Leakage test is performed in accordance with ANSI N14.5, edition 2014, which requires that test personnel shall be certified according to SNT-TC- 1A, edition 2006. The applicant stated that the TN-LC package may be used in overseas countries so that available test personnel may be certified according to ISO 9712 instead of SNT-TC-1A. The applicant demonstrated that a certification of test personnel according to ISO 9712 is similar to SNT-TC-1A as the differences are very limited and cannot lead to question the qualification of test personnel for leakage test.

On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the package in accordance with 10 CFR Part 71.

## 8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The lower limit Balsa density was changed from 10 to 7 $lb/ft^3$, and the moisture content range was changed from 6-10% to 6-12%, and redwood crush stress upper limit from 7500 to 7000 psi. Such a change had been previously approved by staff for a similar package design (Certificate of Compliance No.9302, Revision No. 9, April 17, 2019, ADAMS Accession No. ML19112A168)

The staff reviewed the leakage rate test acceptance criteria for the periodic, pre-shipment, and maintenance leakage rate tests in Section 8.2.2, "Leakage Tests," of the application and finds the values to be consistent with those described in Sections 4.2.3, "Containment Criterion," 4.3.3, "Containment Criterion," and Appendix 4.6.1 of the application. The staff verified the typical leakage rate test methods for the 1FA shipments described in Section 8.2.2 of the application, A.5.3 and A.5.4 of ANSI N14.5-2014 for the gas filled envelope and evacuated envelope, respectively, were capable of the acceptance criterion and test sensitivity for the periodic, pre-shipment, and maintenance leakage rate tests.

The staff also verified that the typical leakage rate test methods for the 1FA shipments described in Section 8.2.2 of the application, A.5.8 and A.5.9 of ANSI N14.5-2014 for the tracer gas sniffer techniques and tracer gas spray method, respectively, were capable of the acceptance criterion and test sensitivity for the pre-shipment leakage rate test. Based on the references to ANSI N14.5-2014 and the applicant provided updates to Chapters 4, 7, and 8 as described above, the staff finds that the application meets the containment criteria of ANSI N14.5-2014.

The staff reviewed the additional change to allow test personnel to be certified to editions of the American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," that are later than 2006, which is the edition referenced in ANSI N14.5-2014, and the staff finds this to be acceptable. The staff also reviewed the additional change to alternatively certify test personnel to International Organization for Standardization (ISO) 9712, "Non-destructive Testing – Qualification and Certification of NDT Personnel."

The applicant described the reason for the proposed alternative is that the TN-LC is used in overseas countries. The staff is not in a position to determine equivalency between Recommended Practice No. SNT-TC-1A to ISO 9712. However, the applicant highlighted the main differences in the additional change 8-A not associated with the RAIs, ADAMS Accession No. ML20304A319, and based on the staff's review of the main differences that staff finds that ISO 9712 meets the intent of ANSI N14.5-2014 to qualify and certify leakage rate testing personnel with a greater number of minimum training hours, a greater number of test questions, a greater near visual acuity, but a lower passing grade in comparison to Recommended Practice No. SNT-TC-1A. The staff recommends that Recommended Practice No. SNT-TC-1A be used to certify leak test personnel on packages used in the United States because that is the document referenced in ANSI N14.5-2014.

The criteria for replacing the O-ring seals were changed from 6 months to 12 months.

The applicant proposed modifications to the gamma shield test in Section 8.1.6.1 of the application to add testing for lead gamma shielding that is precast and installed in its proper place in the packaging. This includes the lead shielding in the cask assembly, except the radial lead shielding which is poured in place (which is already addressed by the gamma shield test), and the lead shielding in the short cavity pin can. The acceptance testing for the precast lead shielding consists of two parts, each with its own acceptance criteria. The first is that the raw lead block shall be of uniform thickness and have an area that is known to be free of voids and defects. That this specified area is known to be free of voids and defects is verified by x-ray radiography that shows no indications of intensity variations. Either a gamma scan or x-ray radiography (or both) is used to examine the lead block. Only portions of the block that show count rates that do not exceed the count rate for the area known to be without voids and defects will be used to create the precast lead shielding components of the TN-LC packaging. Once these portions of the lead block are machined into the components, dimensional checks will confirm that they have a thickness that is no less than the minimum thickness specified in the drawings for the packaging component for which they are to be used. These drawings are incorporated by reference into the certificate of compliance.

The staff reviewed the test and acceptance criteria descriptions. While different from the testing and criteria for poured lead shielding, they do ensure both the integrity (i.e., lack of voids and defects) and thickness of the precast lead meet the design specifications. This is the same outcome for the testing and acceptance criterion for poured lead shielding. They are consistent with the testing and acceptance criteria and actions that the staff has accepted for this kind of test. Thus, the staff finds they are adequate to ensure the gamma shielding of the as-fabricated packaging will perform as designed and ensure the packaging shielding performs as evaluated to meet the radiation requirements and limits in 10 CFR Part 71.

The staff finds the descriptions of the tests and criteria are adequate and consistent with descriptions of tests and criteria for other packages the NRC has approved. The staff also reviewed the packaging drawings and confirmed that specifications in the drawings regarding verifying the packaging's lead shielding (confirming minimum thickness and lack of voids) are consistent with the acceptance test methods in Section 8.1.6.1 of the application. The staff has assurance that actions will be taken to ensure that only packagings with lead shielding components that meet the specifications in the certificate of compliance, including the drawings, as determined by compliance with the tests' acceptance criteria, will be used for transportation.

Based on the review of the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71, and that the maintenance program is adequate to assure packaging performance during its service life.

**CONDITIONS**

The following changes were made to the Certificate of Compliance:

(a) Item No. 3a was modified to list the new address of TN Americas LLC.

(b) Item No. 3b was modified to reference the consolidated revision No. 9 of the application.

(c) Condition No. 5(a)(2) specifies that the first fabricated packaging, Unit 1, shall only be loaded with the TN-LC 1FA basket.

(d) Condition No. 5(a)(3) was modified to include the latest revisions of the licensing drawings.

(e) Condition No. 5(b)(1) was modified to update the numbering of the FQT Tables, add the WE 16x16 fuel design characteristics in Tables 8, 9, and 11 of the CoC, modify the number of rods from 25 to 21 in Table 12 and in the notes to Table 13.

(f) Condition No. 5(b)(2)(iv) was modified to correct a typographical error while Condition No. 5(b)(2) (v) was added to limit the contents of the first fabricated package, Unit 1, to one intact PWR fuel assembly, or up to 21 intact PWR (excluding MOX) fuel rods in a pin can. The condition also specifies that when transporting 9 or fewer fuel rods, the rods shall be placed in the center 3x3 region of the pin can.

(g) Tables 15a and 17 a were added to the Fuel Qualification Tables for a PWR Fuel Assembly - 3.10″ Lead Thickness and 21 PWR Fuel Rods (UO2) - 3.10″ Lead Thickness, respectively. The corresponding notes were also updated. Table 5 was added to show the PRA insertion locations for WE 16x16 Class Assemblies.

The References section of the certificate was updated to reflect the consolidated Revision 9 of the application. The expiration date of the certificate was not changed.

**CONCLUSION**

Based on the statements and representations contained in the application, and the conditions listed above, the staff concludes that the Model No. TN-LC package has been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9358, Revision No. 5.