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February 28, 2022
E-60264

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Response to Request for Additional Information for Application for Revision 6 of Certificate of Compliance No. 9358 for the Model No. TN-LC, Docket No. 71-9358

References: (1) TN Letter E-59073 "Application for Revision 6 of Certificate of Compliance No. 9358 for the Model No. TN-LC, Docket No. 71-9358," dated August 9, 2021.

(2) Letter to Don Shaw (TN) from Pierre Saverot (NRC), "Request for Additional Information for the Review of the Model No. TN-LC Package," dated December 21, 2021, Docket Number 71-9358, Enterprise Project Identifier (EPID) No. L-2021-LLA-0148

(3) NRC Certificate of Compliance for the Model No. TN-LC, USA/9358/B(U)F-96, Revision 5

In accordance with 10 CFR 71.38, TN Americas LLC (TN) made a submission of an application to revise Certificate of Compliance (CoC) No. 9358 for the TN-LC packaging [1]. The NRC requested additional information (RAI) needed to continue the review of the application [2].

Enclosure 1 provides the responses to the RAI. In addition to this enclosure providing the response to the RAI, details for a change to the material specification for the containment boundary O-rings that is not related to the RAI is included as Enclosure 2.

Preliminary changed SAR pages are provided as Revision 10B in Enclosure 3. A consolidated SAR Revision 10 will be submitted upon completion of the NRC review.

The changed pages are indicated by "Revision 10, 07/21" in the header of the page. Each changed page includes a revision bar adjacent to the changed content and the changes made relating to Revision 10B are gray shaded to distinguish them from the Revision 10A changes to the SAR. A public version of the Revision 10B SAR changed pages with proprietary information redacted is provided for public availability as Enclosure 4.

The NRC Electronic Information Exchange (EIE) system is used for submission of this application.

A set of computer calculation files is included as Enclosure 6. Enclosure 5 provides a listing of these computer files. Because the Enclosure 6 computer calculation files exceed the size limit allowed by the NRC EIE application process, they are provided separately on one computer disk.

Proposed changes to the NRC Certificate of Compliance [3] are annotated and provided as Enclosure 7.

Certain portions of this submittal include proprietary information. In accordance with 10 CFR 2.390, TN Americas is providing an affidavit (Enclosure 8) requesting that this proprietary information be withheld from public disclosure.

Should the NRC staff require additional information to support review of this application, please contact Peter Vescovi at 336-420-8325, or by email at peter.vescovi@orano.group.

Sincerely,

Don Shaw

Don Shaw
Licensing Manager
TN Americas LLC

Enclosures:

1. Responses to RAIs
2. Summary of Changes to TN-LC SAR Revision 10B Not Related to RAIs
3. TN-LC Transportation Package Safety Analysis Report Revision 10B Changed Pages (Proprietary Version)
4. TN-LC Transportation Package Safety Analysis Report Revision 10B Changed Pages (Public Version)
5. Listing of Computer Files Contained in Enclosure 6
6. Computer Files Associated with UFSAR Revision 10B (Proprietary)
7. Proposed Certificate of Compliance No. 9358, Revision 6 Markup
8. Affidavit Pursuant to 10 CFR 2.390

cc: Pierre Saverot, Senior Project Manager, U.S. Nuclear Regulatory Commission
Peter Vescovi, Licensing Engineer, TN Americas LLC
Kamran Travassoli, Project Manager, TN Americas LLC

Chapter 2 - Structural Evaluation:

RAI 2-1:

Evaluate applicable Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) scenarios for the Fuel Assembly Can (FAC).

In Section 1.4.5.2 of the application, the applicant discussed the TN-LC-1FA Basket contents which includes the new damaged fuel assembly can (FAC) intended for use with damaged fuel assemblies. The applicant only addressed thermal expansion of the FAC in this amendment request, particularly in Section 2.13.10.2 of the application. The staff could not find any additional evaluations for NCT or HAC for the FAC or a justification why it remains bounded by other analyses already performed.

It is not clear to the staff how the FAC will behave under NCT and HAC scenarios, mainly drop events. In particular, the applicant needs to clarify if impact loads have any effects (i.e., deformations) on the FAC which could then have potential effects on the fuel geometry and therefore on the criticality evaluations. The applicant needs to evaluate how the fuel assemblies, contained within the FAC, will continue to remain subcritical under the applicable NCT and HAC scenarios.

This information is required for the staff to determine compliance with 10 CFR 71.55(e), 71.71 and 71.73.

Response to RAI 2-1:

The fuel assembly can (FAC) is described in SAR Section 1.4.5.2.1 and Drawing 65200-71-91 of the TN-LC SAR.

Per SAR Section 1.4.5.2.1, the thickness of the FAC top lid (Item 5 of Drawing 65200-71-91) is 1.25 inches, and the thickness of the bottom cover (Item 2 of Drawing 65200-71-91) is 1.25 inches (including the thickness of the bottom spacer bars), both of which are greater than the axial gap of 1.00 inch between the basket length of 181.50 inches and cask cavity length of 182.50 inches. Therefore, both top lid and bottom cover are always confined within the fuel compartment, and cannot move outside the fuel compartment on either end, regardless of the basket axial location within the cask cavity during NCT or HAC.

The top lid and bottom cover are design features of the packaging that ensure that any fuel material is always confined within the fuel compartment square boundary, and fuel material cannot be released outside the fuel compartment square boundary into the space between the 1FA basket and TN-LC inner cavity. Figure 2-1-1 shows the top of the cask, where the fuel compartment square boundary is shown in green, the basket is shown in grey, and the FAC top lid is shown in blue. No fuel material can be released into the volumes marked in red because of the presence of the FAC top cover plate. A similar configuration with a gap between the basket and inner cavity could exist at the bottom of the cask if the basket were to slide to contact the TN-LC lid.

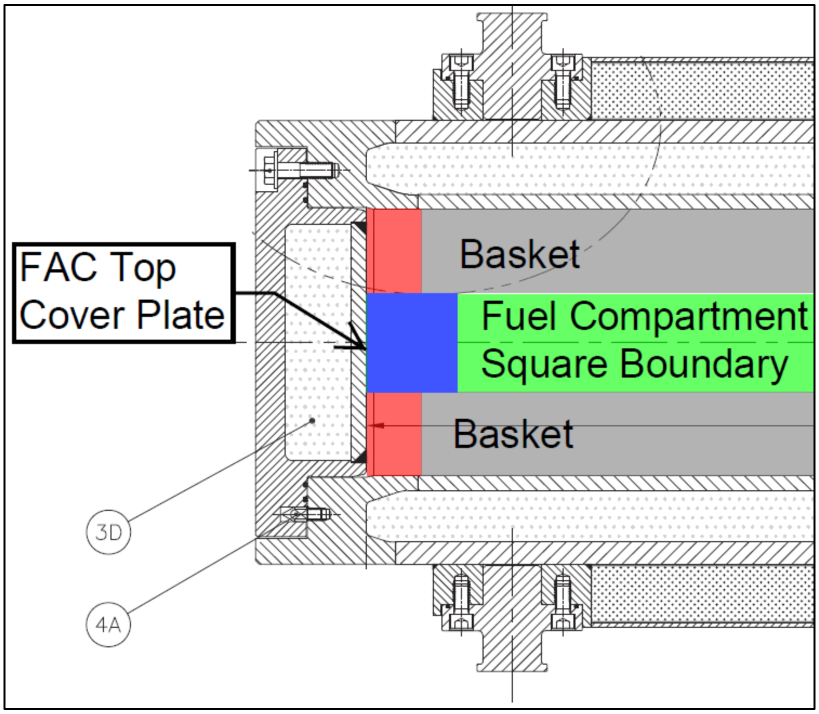

Figure RAI 2-1-1
Fuel Compartment Square Boundary (not to scale)

The function of the top lid and bottom cover would be accomplished even in the event of a failure of the welds that connect the top lid (Item 5 of Drawing 65200-71-91) to the top lid centering liner (Item 4 of Drawing 65200-71-91), or of the weld that connects the bottom cover (Item 2 of Drawing 65200-71-91) to the liner (Item 1 of Drawing 65200-71-91), or of the liner itself. The thicknesses of the top lid and bottom cover alone without the liner are sufficient to achieve the intended function of the FAC. The liner with bottom cover is an operational feature to allow for fuel handling with the FAC attached to the fuel assembly. The top lid and bottom cover perform the safety function of confining fuel material within the 1FA basket fuel compartment, and the liner is not required for the contents to remain subcritical.

The liner is not important in keeping the top lid and bottom cover in place to achieve the intended function of the FAC during transportation. For side drops, both the top lid and bottom cover are self-supported and self-loaded. Also, both plates are not slender; therefore, there is no risk of buckling. For end drop on the lid, the top cover plate is loaded with the fuel assembly through bearing on the TN-LC lid. There is no deformation of the top lid that would allow debris to escape from the fuel compartment. For bottom end drop, the bottom cover of the FAC is loaded by the fuel assembly, which may lead to some deformation of the plate because the spacer bars provide a point of contact for bending. During an end drop or side drop the debris is restricted from escaping the fuel compartment. This ensures debris will not accumulate in the gap between the basket and TN-LC ends. Therefore, the top lid and bottom cover perform the intended safety function described above during both NCT and HAC impacts.

The criticality evaluation for the 1FA basket with damaged PWR fuel assembly is provided as Appendix 6.10.5 in the SAR. The FAC structure is not required to demonstrate that the pressurized water reactor (PWR) contents remain subcritical during routine, normal, or accident transport conditions. The determination of the most reactive damaged fuel configuration from a cask-drop accident is performed in SAR Section 6.10.5.4.2.3. It assumes fresh water intrusion in conjunction with extended damaged fuels condition and fuel reconfiguration including single-ended, double-ended rod shear, missing rods and pitch expansion. The bent or bowed fuel rods scenario is bounded by the pitch expansion configuration, which assumes that the fuel rods remain within the lattice but not in its nominal fuel rod pitch. It is possible that the fuel rods may be crushed inward or bow outward to a certain degree. Rearrangement of the PWR fuel assembly has been evaluated by varying the fuel rod pitch from a minimum pitch that is limited by the clad outer diameter, and a maximum pitch that is limited by the 1FA basket fuel compartment inner width (i.e., the FAC liner is not modeled). Further cladding failure is considered assuming fully de-cladded fuel rods.

The presence of the FAC does not change the assumptions for fuel rearrangement that are used in the criticality evaluation. Any deformation of the FAC liner does not change the assumptions for fuel rod pitch, and the presence of the top lid and bottom cover confine the fuel assembly rods and any fuel debris from a damaged fuel assembly to the geometry of 1FA basket fuel compartment.

Drawing No. 65200-71-91 has been revised on sheet 1 to show construction details for Item 6, to show Items 1, 4, and 6 as not-important to safety, to allow Item 7 to be optional under certain conditions, and to make an editorial change. Sheet 2 of the drawing has been revised to show details between Items 1 and 6. In addition, Appendix 1.4.1 has been updated to reflect the drawing revision change to Revision 10.

Impact:

SAR Drawing No. 65200-71-91 and Appendix 1.4.1 have been revised as described in the response.

<u>Chapter 6 - Criticality Evaluation:</u>

RAI 6-1:

Revise the application to address the potential for fuel assembly misload considering the burnup, enrichment, and cooling time of the total population of discharged PWR fuel assemblies for the burnup credit analysis of the TN-LC package.

Section 6.10.5.8 of the SAR discusses the analysis to determine misload loading curves for the package which are shown in Tables 6.10.5-28 and 6.10.5-29 for the WE 14x14 and WE 17x17 fuel assembly classes, respectively.

However, the applicant does not discuss how the misload loading curves relate to the probability of having a misload, by comparing the loading curves to the characteristics of the discharged fuel population. NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” recommends that a severely underburned assembly for misload analysis should be chosen such that the misloaded assembly’s reactivity bounds 95 percent of the discharged PWR fuel population considered unacceptable for loading in the transportation package with 95- percent confidence. Demonstration that the misload loading curves bound the discharged fuel population is typically accomplished by comparing that curve with the characteristics of discharged PWR fuel assemblies allowed for loading in the package. The NRC finds the results of the most recent U.S. Energy Information Administration’s “Nuclear Fuel Data Survey” (RW-859) or later similar fuel data sources (i.e., GC-859), acceptable to estimate the discharged fuel population characteristics.

This information is required for the staff to ensure that the Model No. TN-LC package will meet the criticality safety requirements of 10 CFR 71.55 when loaded with the contents described in the application.

Response to RAI 6-1:

SAR Section 6.10.5.8 has been updated to add a discussion and new figures of misload loading curves vs. discharged spent fuel inventory (Figure 6.10.5-25 and Figure 6.10.5-26). Department of Energy (DOE) compiled fuel data is employed for this purpose; the data consists of existing and projected spent nuclear fuel assemblies that are potentially available for long-term storage at a repository. The data encompasses more than one hundred thousand entries of burnup and enrichment combinations from RW-859.

The study shows the misload loading curves bound 99%+ of the discharged population, and is shown in new Table 6.10.5-30. Note that the set of misload curves in Table 6.10.5-28 and Table 6.10.5-29 has been updated using ORIGEN-APR library with control rods insertion since the set of base loading curves with ORIGEN-APR library with control rods insertion is more penalizing (slightly lower enrichments) compared to that obtained with ORIGEN-APR library with BPR (see the Response to RAI 6-4 for further details).

Impact:

SAR Section 6.10.5.8 and Tables 6.10.5-28 and 6.10.5-29 have been revised as described in the response.

SAR Table 6.10.5-30 and Figures 6.10.5-25 and 6.10.5-26 have been added as described in the response.

RAI 6-2:

Revise the application to include procedural steps in the package operating procedures to compare the irradiation parameters of the fuel assembly being loaded to those used in the burnup credit isotopic depletion analysis.

Table 6.10.5-27 of the SAR provides a summary of the fuel irradiation parameters employed in the isotopic depletion analysis for fuel contents analyzed using burnup credit.

However, there is no discussion in the SAR regarding if these conditions are bounding of all PWR fuel. NUREG-2216 recommends that contents specifications tied to the actual reactor operating conditions may be needed unless the operating condition values used in the evaluation can be justified as those that produce the maximum k_{eff} values for the proposed contents. Staff notes that while the irradiation parameter values included in Table 6.10.5-27 are reasonably bounding for most PWR fuel assemblies, there may be assemblies in the discharged fuel population that exceed one or more of these parameter values, and which may produce a higher k_{eff} than considered in the applicant's analysis.

This information is required for the staff to ensure that the Model No. TN-LC package will meet the criticality safety requirements of 10 CFR 71.55 when loaded with the contents described in the application.

Response to RAI 6-2:

SAR Table 6.10.5-27 shows the depletion parameters employed in the pressurized water reactor (PWR) burnup credit approach for the criticality analysis performed in Appendix 6.10.5. The depletion parameters shown in Table 6.10.5-27 ensure the overall conservatism of the depletion model and are therefore the bounding set of isotopic concentrations employed in the criticality analysis. Section 6.10.5.6 of Appendix 6.10.5 has been updated to provide additional discussions related to the depletion parameters. A note has been added to Table 6.10.5-27 for clarification regarding the upper-end values shown for fuel temperature, moderator temperature, soluble boron concentration, and specific power. A requirement to compare the reactor operating parameters to values shown in Table 6.10.5-27 is added in Appendix 7.7.4.

Additional administrative controls have been added to SAR Section 7.7.4.1 TN-LC-1FA Basket Wet Loading. These administrative controls include a requirement to compare the reactor operating parameters for the irradiation period of the fuel assembly against those shown in Table 6.10.5-27 to ensure compliance with the isotopic depletion analysis.

Impact:

SAR Section 6.10.5.6 and Table 6.10.5-27 and Section 7.7.4.1 have been revised as described in the response.

RAI 6-3:

Revise the application to include additional administrative procedures in the package operating procedures to ensure that the TN-LC package will be loaded with fuel that is within the specifications of the approved contents.

The operating procedures for the TN-LC package do not include any procedures specific to loading fuel assemblies under the burnup credit requirements of Table 1.4.5-4a of the SAR. NUREG-2216 recommends procedures the applicant may consider in order to protect against misloads in transportation packages that rely on burnup credit for criticality safety, including the following:

- verification of the location of high-reactivity fuel (i.e., fresh or severely underburned fuel) in the SNF pool, both before and after loading,
- qualitative verification that the assembly to be loaded is burned (visual or gross measurement),
- quantitative measurement of any fuel assemblies without visible identification numbers,
- independent, third-party verification of the loading process, including the fuel selection process and generation of the fuel move instructions

This information is required for the staff to ensure that the Model No. TN-LC package will meet the criticality safety requirements of 10 CFR 71.55 when loaded with the contents described in the application.

Response to RAI 6-3:

SAR Section 7.1.1 of Chapter 7 and Section 7.7.4.1 of Appendix 7.7.4 have been updated for the package operations when burnup credit is employed for criticality safety.

Impact:

SAR Sections 7.1.1 and 7.7.4.1 have been revised as described in the response.

RAI 6-4:

Revise the application to clarify the irradiation conditions used in generating ORIGEN-ARP reactor libraries for the burnup credit analysis with respect to control rod or burnable poison rod assembly (BPRA) exposure.

Section 6.10.5.6 of the SAR discusses the generation of ORIGEN-ARP reactor libraries for use with the STARBUCS code sequence. This section states that two sets of libraries for each fuel assembly class were developed – one with BPRAs inserted into the fuel assembly guide tubes for the full irradiation period, and another with control rods inserted into the fuel assembly guide tubes for the first 15 GWd/MTU of irradiation.

However, the SAR does not include any discussion of which of these reactor libraries results in a higher k_{eff}, or which is used to generate the final burnup credit loading curves for each assembly class. The SAR should be revised to include results of reactivity comparisons using each reactor library, and demonstration that the reactor library that results in the highest k_{eff} is the one used to determine the loading curve for each fuel assembly class.

This information is required for the staff to ensure that the Model No. TN-LC package will meet the criticality safety requirements of 10 CFR 71.55 when loaded with the contents described in the application.

Response to RAI 6-4:

The loading curves shown in SAR Table 6.10.5-8 and Table 6.10.5-9 were developed, respectively, for WE 14x14 and WE 17x17 fuel classes using the ORIGEN-ARP library assuming burnable poison rods (BPR) inserted in fuel assembly during the entire irradiation history. A clarification has been added to SAR Section 6.10.5.4.3 for this purpose.

Two additional loading curves for WE 14x14 and WE 17x17 fuel classes using the ORIGEN-ARP library assuming control rod (CR) insertion as described in Section 6.10.5.6 are generated and shown in new SAR Tables 6.10.5-8A and 6.10.5-9A. Note that the maximum initial enrichments obtained with ORIGEN-ARP library assuming CR insertion are reduced compared to those with ORIGEN-ARP library assuming BPR. The maximum reduction in initial enrichment is less than 0.2 wt% U-235.

New SAR Table 1.4.5-4b has been added to Appendix 1.4.5 for the maximum planar average initial enrichment and minimum burnup combination (1FA PWR) with control rod insertion. SAR Section 7.1 and Table 7-1, and Section 6.10.5.1.2 have been updated regarding new Table 1.4.5-4b, accordingly.

Impact:

SAR Appendix 1.4.5, Section 6.10.5.4.3, Section 6.10.5.1.2, Section 7.1, and Table 7-1 have been revised as described in the response.

SAR Tables 1.4.5-4b, 6.10.5-8A, and 6.10.5-9A have been added as described in the response.

Summary of Changes to TN-LC SAR Revision 10B
Not Related to RAIs

<u>Change 1</u>:

Changed SAR (Chapter, Section, Appendix, Table, Drawing):

SAR Appendix 1.4.1

Description of Change:

Editorial correction to drawing revision numbers.

Justification:

Drawing revision numbers for drawings that were previously approved were corrected in Appendix 1.4.1 to reflect the latest drawings revisions. The previously approved CoC references the correct drawing revisions. There was no tracking used for this and other editorial changes.

<u>Change 2</u>:

Changed SAR (Chapter, Section, Appendix, Table, Drawing):

Drawing 65200-71-01

Description of Change:

Revise Drawing 65200-71-01 to update Note 17 to allow the following additional seal materials in addition to the current one: VM835-75, VM125-75 or VX065-75.

Justification:

V1289 compound for the seals is discontinued, and because there is no equivalence allowance, three compounds that exhibit similar temperature, elongation, and compression properties have been added as alternate seal materials.

Relevant properties for the proposed replacement seal specifications are compiled in the table below.

All three proposed alternate materials are acceptable for use at temperatures down to -40 °C.

All three alternate compounds are also rated up to 400 °F maximum steady state, and have been tested for the same short-term maximum temperature (482 °F) as the V1289 compound. All three exhibit similar properties changes after 70h at that temperature (note: for the TN-LC the temperatures only exceed 400 °F for 1 hour), except the elongation change for VX065 and VM125, but the drop in elongation is not deemed relevant because both materials start out with a better elongation than V1289 to begin with.

Finally, all three alternate compounds exhibit either better or similar compression set than the V1289 compound.

Based on this, these three alternate compounds are acceptable for use for the TN-LC seals.

Summary of Changes to TN-LC SAR Revision 10B
Not Related to RAIs

Material	Type A Hardness	Temp. range (°F)		Elongation (% min)	Heat aging - 70hr @250°C/482°F (ASTM D573)		Compression Set, % of Original Deflection (Max.)
		Min.	Max.		Hardness Change (pts)	Elongation Change (%)	
V1289 (discontinued)	75	-50	400	151	-1	7	11
VX065	75	-65	400	174	0	-13	5
VM125	75	-40	400	280	-2	-6	10
VM835	75	-40	400	215	3	8	13

See the following five pages for related information from Parker Hannifin Corporation.

Parker Hannifin Corporation
O-Ring & Engineered Seals Division
2360 Palumbo Drive
Lexington, KY 40509

Office 859 269 2351

11/03/21

Reference: V1289-75 Obsolete

Dear Valued Customer,

We regret to share that the polymer used to make V1289-75 has been discontinued by our supplier.
V1289 was used across several industries and also carries the pedigree of QPL (Qualified Producers List) for AMS7379.

V1289 is a 75 durometer fluorocarbon, ASTM D1418 type 3 classification for low temperature flexibility. At present there is not a straightforward replacement available but is in development and will be finalized in the coming months.

Until a replacement is commercialized, alternate low temperature fluorocarbon options are listed below V1289 in the table:

Material	Type A hardness	Temperature range	Tg
V1289	75	-50 to 400°F	-40°F (-40°C)
VX065	75	-65 to 400°F	-49°F (-45°C)
VG286	80	-45 to 400°F	-31°F (-35°C)
VG109	90	-50 to 400°F	-33°F (-36°C)
VM125	75	-40 to 400°F	-22°F (-30°C)
VM835	75	-40 to 400°F	-24°F (-31°C)

If you have any questions regarding the performance of V1289 compared to the other material options, please contact an Applications Engineer by calling 859-335-5101, or emailing OESmailbox@parker.com.

Sincerely,

Dorothy Kern
Applications Engineering Manager
O-Ring & Engineered Seals Division
dkern@Parker.com

Parker

Compound Data Sheet
O-Ring Division United States

LABORATORY TEST REPORT*

*Data compiled from previous reports

TITLE: Evaluation of Parker Compounds V1289-75, VX065-75, and VM125-75

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Kentucky 40509
(859) 269-2351

REPORT DATA

BASIC PROPERTIES	V1289-75	VX065-75	VM125-75
Hardness, Shore A, pts.	76	75	73
Tensile Strength, psi.	1766	1658	2301
Elongation, % min.	151	174	280
HEAT AGING:			
70 HRS. @ 250 C, ASTM D573			
Hardness chg. pts.	-1	0	-2
Tensile Strength chg, %	-9	-12	-13
Elongation chg, % max.	+7	-13	-6
COMPRESSION SET (Plied):			
22 HRS. @ 200 C , ASTM D395			
Percent of Original Deflect, Max.	+11	+5	+10
FLUID RESISTANCE (Service Fluid 101):			
70 HRS. @ 200 C, ASTM D471			
Hardness chg. pts.	-6	0	-5
Tensile Strength chg, %	-15	-14	-14
Elongation chg, % max.	-8	-13	+10
Volume chg, %	+9	+6	+10
Low Temperature			
TR-10. °C, ASTM D1329			
Report	-38	-45	-29

COMPOUND DATA SHEET

Parker O-Ring Division, North America

MATERIAL REPORT

Report Number: 92880
Date: 3/13/2013

Title: Evaluation of Parker Compound VM835-75

Elastomer Type: Fluorocarbon (FKM)

Purpose: To obtain typical test data.

Specification: ASTM D2000 M2HK710 A1-10 B38 E078 Z1 (Shore A Hardness 75 +/-5), Z2 Elongation 125% min, Z3 (Specific Gravity), Z4 (TR-10)

Color: Black

Recommended Temperature Range: -40°F to 400°F

Recommended For: Mineral oil and grease, ASTM No. 1 oil, IRM 902 oil, IRM 903 oil, non-flammable hydraulic fluids, silicone oils and greases, aliphatic hydrocarbons (propane, butane, natural gas), aromatic hydrocarbons (benzene, toluene), chlorinated hydrocarbons (trichloroethylene and carbon tetrachloride), gasoline, high vacuum, ozone, weather, and aging resistance.

Not Recommended For: Glycol based brake fluids, ammonia gas, amines, alkalis, superheated steam, and low molecular weight organic acids (formic and acetic acids).

Additional Approvals: N/A

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

REPORT DATA

Original Physical Properties	Test Method	Spec Limits	Test Results
(Z1) Hardness, Shore A, pts.	ASTM D2240	75 ±5	78
Tensile Strength, PSI (Mpa)	ASTM D412	1450 (10)	3059
(Z2) Ultimate Elongation, %	ASTM D412	125	215
(Z3) Specific Gravity	ASTM D297	as received	1.84
Fluid Resistance (Basic Requirement)			
IRM 903, 70 hrs @ 302°F			
Volume Change, %	ASTM D471	+10	+2
(A1-10) Heat Age			
70 hrs. @ 482°F			
Hardness Change, pts.	ASTM D573	+10	+3
Tensile Strength Change, %		-25	-22
Ultimate Elongation Change, %		-25	+8
(B38) Compression Set (Plied)			
22 hrs. @ 392°F			
Percent of Original Deflection, Max	ASTM D395 Method B	50	13
(E078) Fluid Resistance			
Service Fluid 101, 70 hrs @ 392°F			
Hardness Change, pts.	ASTM D471	-15 to +5	-8
Tensile Strength Change, %		-40	-6
Ultimate Elongation Change, %		-20	-1
Volume Change, %		0 to +15	+11
(Z4) Low Temperature Resistance			
TR-10, temperature °F , C	ASTM D1329	report	-22 (-30)

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

Enclosure 3 to E-60264

TN-LC Transportation Package Safety Analysis Report Revision 10B Changed Pages (Proprietary Version) Withheld Pursuant to 10 CFR 2.390

Enclosure 4 to E-60264

TN-LC Transportation Package Safety Analysis Report Revision 10B Changed Pages (Public)

Appendix 1.4.1
TN-LC Transport Package Drawings

Drawing Number	Title
65200-71-01 Revision *10*	TN-LC Cask Assembly (11 sheets)
65200-71-20 Revision 5	TN-LC Impact Limiter Assembly (2 sheets)
65200-71-21 Revision 2	TN-LC Transport Packaging Transport Configuration (1 sheet)

Drawing Number	Title
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)

Drawing Number	Title
65200-71-60 Revision 4	TN-LC-MTR Basket General Assembly (4 sheets)
65200-71-70 Revision 4	TN-LC-MTR Basket Fuel Bucket (2 sheets)

Drawing Number	Title
65200-71-80 Revision 4	TN-LC-TRIGA Basket (5 sheets)

Drawing Number	Title
65200-71-90 Revision 7	TN-LC-1FA Basket (5 sheets)
65200-71-91 Revision 0	*TN-LC Transportation Cask* *TN-LC-1FA PWR Fuel Basket* *Damaged Fuel Assembly Can (FAC) (3 sheets)*
65200-71-96 Revision 5	TN-LC-1FA BWR Sleeve and Hold-Down Ring (2 sheets)
65200-71-102 Revision 7	TN-LC-1FA 21 Pin Can Basket (4 sheets)

RAI 2-1

8 7 6 5 4 3 2 1

H G F E D C B A

Proprietary and Security Related Information
for Drawing 65200-71-01 Rev. 10A
Withheld Pursuant to 10 CFR 2.390

8 7 6 5 4 3 2 1

Proprietary and Security Related Information for Drawing 65200-71-91 Rev. 0B Withheld Pursuant to 10 CFR 2.390

Alternatively, in the absence of PRAs, burnup credit restrictions, as shown in Table 1.4.5-4a or Table 1.4.5-4b are required for 1FA PWR fuel transportation. The burnup/enrichment/cooling times are determined using PWR burnup credit approach and are required while transporting intact or damaged PWR fuel assembly in order to ensure that the maximum reactivity is subcritical and below the upper subcritical limit (USL). Note that burnup credit is not applicable to BW 15x15 fuel class.

The damaged fuel assembly can (FAC) is a stainless steel, square-section structure made of a sheet metal liner, a bottom closure welded to this liner, and a top closure lid, which simply sits on top of the can. There is a square structure welded to the bottom of the top lid, which slides inside the damaged fuel can liner. Because of this, the top closure lid can freely slide along the axis of the cask (within the cask cavity), but is captured by the body of the can in any direction perpendicular to the axis of the cask cavity (see Figure 1.4.5-7), and the top lid cannot come off of the damaged fuel can once the cask is closed because the axial gap between the damaged fuel can and the cask cavity is smaller than the length of the square structure welded under the lid.

The damaged fuel can top closure lid, liner, and bottom closure all have multiple drain holes to allow for efficient draining of the water from the can during operations.

Finally, the top and bottom closures are sufficiently thick (in relation to the available axial gaps within the cask cavity) to ensure that they are always constrained by the basket in any direction perpendicular to the cask axis.

1.4.5.2.2 BWR Fuel Assemblies

The TN-LC-1FA basket is designed to transport one intact BWR fuel assembly as specified in Table 1.4.5-6. Basket cell sleeves are used to reduce the area within the 1FA basket for BWR fuel. The BWR FQT is provided in Table 1.4.5-9. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable assembly average burnup is limited to 62 GWd/MTU. The maximum allowable heat load for the TN-LC-1FA basket loaded with a BWR fuel assembly is 2.0 kW.

1.4.5.2.3 Fuel Rods in the Pin Can

The TN-LC-1FA basket is designed to transport up to 21 intact *or damaged* light water reactor fuel rods in the pin can. This includes irradiated PWR, BWR, MOX, and EPR fuel rods. The maximum peak burnup for fuel rods is 90 GWd/MTU. Two designs are available, with cavity lengths of 180.24 in. *(4,578.1 mm)* or 169.55 in. *(4,306.6 mm)*. The pin can with the shorter cavity length is heavily shielded with lead at the ends, while the pin can with the longer cavity length does not feature axial lead shielding. The longer cavity pin can is used only for EPR pins, which are much longer than a standard fuel rod (an EPR rod is approximately 179.24 in. long). All other rods are transported in the shorter cavity pin can with heavy axial shielding.

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Table 1.4.5-1
PWR Fuel Specification for the Fuel to be Transported in the TN-LC-1FA Basket

PHYSICAL PARAMETERS: Fuel Class (1)(2)	Intact *or damaged* unconsolidated B&W 17x17, WE 17x17, CE 16x16, B&W 15x15, WE 15x15, CE 15x15, WE 14x14, WE 16x16, and CE 14x14 class PWR assemblies (without control components) that are enveloped by the fuel assembly design characteristics listed in Table 1.4.5-2. Reload fuel manufactured by the same or other vendors but enveloped by the design characteristics listed in Table 1.4.5-2 is also acceptable.
Maximum Assembly + PRA + *damaged FAC (as applicable)* Weight	1850 lbs *(839 kg)*
Damaged Fuel	*Assemblies are PWR assemblies containing fuel rods with known or suspected cladding defects greater that hairline cracks or pinhole leaks. The extent of damage in the fuel assembly, including non-cladding damage, is to be limited so that a fuel assembly maintains its configuration for normal conditions. Damaged fuel assemblies shall also contain top and bottom end fittings. Damaged fuel assemblies may also contain missing or partial fuel rods.*
Fissile Material	UO_2
Maximum Initial Uranium Content(4)	490 kg/assembly
Maximum Unirradiated Assembly Length	178.3 inches *(4,528.8 mm)*
THERMAL/RADIOLOGICAL PARAMETERS: Fuel Assembly Average Burnup, Enrichment and Minimum Cooling Time	Per Table 1.4.5-8 and Table 1.4.5-8a (applicable to TN-LC Unit 01)
Maximum Planar Average Initial Enrichment	5.0(3) wt.% U-235
Maximum Decay Heat(5)	3.0 kW per Assembly
Minimum B-10 content in poison plates loading	• 16.7 mg/cm^2 (Natural or Enriched Boron Aluminum Alloy / Metal Matrix Composite (MMC)) • 20.0 mg/cm^2 (Boral®)
Minimum number of absorber rods per PRA as a function of assembly class	Per Table 1.4.5-4 *(Note that the use of PRAs is optional)*
Burnup Credit Restriction in the absence of PRAs	*Per Table 1.4.5-4a or Table 1.4.5-4b*

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*Table 1.4.5-4*b
Maximum Planar Average Initial Enrichment/Minimum Burnup Combination – 1FA PWR – With Control Rod Insertion (1)

	WE 17x17, WE 16x16, WE 15x15, CE 14x14, CE 15x15 and CE 16x16 Fuel Assembly Classes			
Fresh Fuel	*2.90 wt. % U-235*			
Cooling Time	***5 Years***	***10 Years***	***15 Years***	***20 Years***
Burnup (GWd/MTU)	***Fuel Initial Enrichment (wt. % U-235)***			
5	*2.97*	*2.99*	*3.00*	*3.01*
10	*3.29*	*3.31*	*3.34*	*3.36*
15	*3.54*	*3.60*	*3.64*	*3.66*
20	*4.21*	*4.38*	*4.45*	*4.53*
25	*4.75*	*4.91*	*4.98*	*5.00*
30	*5.00*	*5.00*	*5.00*	

	WE 14x14 Fuel Assembly Class			
Fresh Fuel	*2.95 wt. % U-235*			
Cooling Time	***5 Years***	***10 Years***	***15 Years***	***20 Years***
Burnup (GWd/MTU)	***Fuel Initial Enrichment (wt. % U-235)***			
5	*3.20*	*3.20*	*3.21*	*3.23*
10	*3.57*	*3.57*	*3.59*	*3.59*
15	*3.81*	*3.86*	*3.90*	*3.90*
20	*4.48*	*4.62*	*4.71*	*4.78*
25	*5.00*	*5.00*	*5.00*	*5.00*

(1) Fuel assemblies with accumulated control rod insertion through the first 15 GWd/MTU. Fuel assemblies with accumulated control rod insertion greater than the first 15 GWd/MTU are not authorized.

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The criticality analysis is performed using the bounding Westinghouse (WE) 17x17 and WE 14x14 fuel classes identified in Table 6.10.4-2. The results of the WE 17x17 fuel class bound those of the WE 16x16, WE 15x15, Combustion Engineering (CE) 14x14, CE 16x16 and CE 15x15 fuel classes. The Babcock & Wilcox (BW) 15x15 fuel class is not included in the PWR burnup credit criticality analysis.

Intact or damaged fuel assembly is authorized for transportation in the 1FA PWR basket. Criticality calculations are performed to determine the minimum assembly average burnup as a function of initial enrichment and cooling time for the two bounding fuel assembly (FA) classes, which are listed in Table 1.4.5-4a *or Table 1.4.5-4b*. The calculations determine k_{eff} with the CSAS5 control module of SCALE 6.1.3 [3] for each assembly class and initial enrichment, including all uncertainties to assure criticality safety under all credible conditions.

The results of the evaluation demonstrate that the maximum k_{eff}, including statistical uncertainty, and appropriate biases associated with the burnup credit methodology is less than the USL determined from a statistical analysis of benchmark criticality experiments. The statistical analysis procedure includes a confidence band with an administrative safety margin of 0.05.

6.10.5.1.3 Criticality Safety Index

For the PWR fuel assembly payload, no HAC array models are developed (2N = 1). Therefore, per 10 CFR 71.59, N=0.5, and the CSI is 50/N = 100 for this payload. In the NCT array cases for the PWR fuel assembly payload, 5N=2.5 and three packages are modeled.

6.10.5.2 Package Fuel Loading

The 1FA PWR basket is capable of transporting one PWR fuel assembly. A detailed listing of the contents of the 1FA PWR basket is provided in Table 6.10.4-2.

For all the FA classes, control components (CCs) are also included as authorized contents. The only change to the package fuel loading to evaluate the addition of these CCs would be to replace the water in the guide tubes with $^{11}B_4C$. Since these CCs displace moderator in the assembly guide and or instrument tubes, an evaluation is not needed to determine the potential impact of storage of CCs that extend into the active fuel region on the system reactivity. The presence of these CCs such as control rod assemblies (CRAs), control element assemblies (CEAs) and burnable poison rod assemblies (BPRAs) will result in a reduction in the reactivity of the FAs. CCs that do not extend into the active fuel region of the assembly do not have any effect on the reactivity of the system as evaluated. Therefore, CCs are not included in any of the criticality models.

The criticality analysis is performed using two fuel assembly types, WE 14x14 STD, and WE 17x17 Robust Fuel Assembly (RFA)/LOPAR representing WE 14x14 and WE 17x17 fuel classes, respectively.

6.10.5.3 Model Specification

The evaluations are performed using SCALE 6.1.3 [3] and ENDF/B-VII nuclear data. The SCALE 6.1.3 capabilities used include automated sequences to produce problem-dependent multi-group cross-section data and analysis sequences for Monte Carlo neutron transport (CSAS5) and burnup-credit criticality safety (STARBUCS). The 238-group cross-section library based on the ENDF/B-VII nuclear data and the resonance cross-section methodology employing CENTRM are used.

6.10.5.4.3 Criticality Results

The study to determine the most reactive configuration considered four different scenarios: double shear, single shear, missing rods and pitch expansion. The results of the study are presented in *Table 6.10.5-6* and *Table 6.10.5-7* for WE 14x14 and WE 17x17 fuel assemblies, respectively. The results show that the configuration with fully expanded pitch is the most reactive configuration in both cases. Therefore, the fully expanded pitch is used for developing the loading curves. (It should be noted that the k_{eff} values in these cases are higher. These models are used only for comparative study and hence the value of k_{eff} need not be compared to any USL values).

Loading curves are generated for the WE 14x14 and WE 17x17 fuel assembly classes loaded in TN-LC cask. The BECT combinations are generated such that the k_{eff} is below the k_{BUC} values. This includes maximum allowable enrichment values corresponding to burnup values ranging from 0 to *30* GWd/MTU, cooling times 5, 10, 15 and 20 years.

The BECT for WE 14x14 fuel class are determined for single/array of package under HAC and are tabulated in *Table 6.10.5-8 and Table 6.10.5-8A using respectively ORIGEN-ARP cross-section libraries assuming BPR and CR insertions, as described in Section 6.10.5.6*. The BECT for WE 17x17 fuel class are determined for single/array of package under HAC and are tabulated in *Table 6.10.5-9 and Table 6.10.5-9A using ORIGEN-ARP cross-section libraries assuming BPR and CR insertions, respectively, as described in Section 6.10.5.6.* The results of the WE 17x17 fuel class bound those of the WE 15x15, CE 14x14, CE 16x16 and CE 15x15 fuel classes as WE 17x17 fuel class bounds those fuel classes per Table 6.10.4-8. The BW 15x15 fuel class is not included in the PWR burnup credit criticality analysis.

The criterion for sub-criticality is that:

$$k_{eff} + (\beta_i + \Delta k_i) + \Delta k_x < USL$$

where USL is the upper subcriticality limit established by an analysis of benchmark criticality experiments,

Δk_x is the code bias due to minor actinides and fission products, 0.1 x 0.015 = 0.0015,

And $(\beta_i + \Delta k_i)$ is the burnup-dependent bias and bias uncertainty for isotopics validation shown in Table 6-3 of [2].

$$k_{KENO} + 2\sigma_{KENO} + (\beta_i + \Delta k_i) + 0.0015 < 0.9424$$

6.10.5.5 Critical Benchmark Experiments and applicable biases

This section summarizes evaluations performed for benchmarking the various computer codes utilized in the criticality analysis. A description of the benchmarking analyses performed in support of the criticality analyses for the TN-LC transport cask where fresh fuel is considered is provided in Section 6.10.5.5.1. The description of the benchmarking analyses performed in support of burnup credit is provided in Section 6.10.5.5.2. For both fresh fuel and burnup credit evaluations, the USL is also determined.

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Additionally, a sensitivity and uncertainty (S/U) tool is used to generate a parameter that quantifies the similarity of the two systems. From [3]: "The technique compares the detailed sensitivity data for the two systems, giving greater weight to comparisons of sensitivities for nuclides and reactions with the highest nuclear data uncertainties." The correlation coefficient, c_k obtained from this process indicates the degree of similarity by the following standards: c_k greater than or equal to 0.9 indicates similar systems, c_k between 0.8 and 0.9 indicates marginally similar and less than 0.8 is not recommended for use. The system application is the TN-LC loaded with 1FA PWR basket, while the "experiment" is the GBC-32.

The TSUNAMI-3D module of SCALE 6.1.3, [3], has been used for performing the similarity analysis. TSUNAMI-3D calculations are performed for the application and experiment models. Isotopic number densities obtained from the STARBUCS outputs are used for the TSUNAMI-3D models as well. Direct perturbation calculations are used to confirm the adequacy of the sensitivity data files. Direct perturbation calculations involve varying the composition information around the nominal value and using the resulting k_{eff} value variations to calculate the total sensitivity. The direct perturbation results are compared with the TSUNAMI sensitivity results to confirm the adequacy of the sensitivity data. Finally, TSUNAMI-3D generates sensitivity data files (.sdf), which contains the energy-dependent sensitivity coefficients for each value of burnup.

The .sdf files generated by TSUNAMI-3D for the two systems in the previous step are used by TSUNAMI-IP to determine a c_k value at each burnup. The TSUNAMI-IP module of SCALE 6.1.3, [3], was used to calculate detailed k_{eff} uncertainty information for the application model. The correlation factor, C_k quantifies correlations in uncertainties by propagating the tabulated cross-section-uncertainty information to the calculated k_{eff} value of a given system via the energy-dependent sensitivity coefficients.

This evaluation demonstrates similarity by comparing the global parameters, as well as by determining the sensitivity and uncertainty. The c_k parameters generated from the sensitivity and uncertainty calculation, which indicate high degree of similarity, are provided in *Table 6.10.5-15*.

The results satisfy the requirements of Reference [2] thereby allowing the user to adopt the results from Table 3 of [2], in preparing system-specific loading requirements with burnup credit.

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6.10.5.9 Evaluations under NCT and HAC

This section describes the evaluations under HAC and NCT performed for the TN-LC transport package with the 1FA PWR basket.

6.10.5.9.1 Package Arrays under HACs

As the CSI is 100, the criticality analysis performed to demonstrate the compliance with the sub-criticality requirements of 10 CFR 71.55 (e), Section 6.10.5.9.3, also meets the sub-criticality requirements of 10 CFR 71.59 (a) (2).

6.10.5.9.2 Package Arrays under Normal Conditions of Transport

As the CSI is 100, the criticality analysis performed for an array of three undamaged packages described in Section 6.10.5.4.2.4 demonstrates the compliance with the sub-criticality requirements of 10 CFR 71.59 (a) (1). The analysis is conducted for WE 14x14 and WE 17x7 fuel classes. Note that BW 15x15 fuel class is added the analysis for conservatism. The results of the analysis are shown in Table 6.10.5-10. The highest k_{eff} is 0.2303.

6.10.5.9.3 Single Package Evaluation

The criticality analysis performed for a single package damaged PWR fuel under HAC in a fully flooded 1FA PWR basket is employed to demonstrate compliance with the sub-criticality requirements of 10 CFR 71.55 (e). The damaged PWR fuel under HAC model simulates a fuel reconfiguration scenario where fuel cladding, guide and instrument tubes, end fittings, and spacer grids are removed with the fuel rods in the most reactive configuration. The loading curves generated in Section 6.10.5.4.2.4 for the WE 14x14 and WE 17x17 fuel classes determined the maximum allowable enrichment values associated with burnup values ranging from 0 to *30* GWD/MTU, cooling times 5, 10, 15 and 20 years allowed for transportation. The analysis demonstrates the compliance with the sub-criticality requirements of 10 CFR 71.55 (e), 10 CFR 71.55 (d) and 10 CFR 71.55 (b) for the 1FA PWR basket in the TN-LC transport cask. The highest k_{eff} plus biases is 0.9418.

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6.10.5.10 References

1. 10 CFR 71, Packaging and Transportation of Radioactive Materials.

2. U.S. NRC, NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” August 2020.

3. Oak Ridge National Laboratory, RSIC Computer Code Collection, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011.

4. International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE), NEA-1486/15, NEA Nuclear Science Committee, September 2016.

5. U.S. NRC, NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” January 2001.

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8. U.S. NRC, NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analysis – Criticality (k_{eff}) Predictions,” J. M. Scaglione, D.E. Mueller, J.C. Wagner, and W.J. Marshall, April 2012.

9. U.S. NRC, NUREG/CR-6665, “Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel,” February 2000.

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11. “Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors,” ANSI/ANS-8.1-1998; R2007, an American National Standard, published by the American Nuclear Society, LaGrange Park, IL, 1998.

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13. U.S. NRC, NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” September 2008.

14. US NRC, NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Isotopic Composition Predictions,” April 2012.

15. US NRC, NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit,” January 2008.

16. *DOE Purchase Order DE-AF28-04RW12278, “Pool Inventories in Compact Disk Attachment,” May 21, 2004.*

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17. *U.S. NRC, Interim Staff Guidance-8, Revision 3, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks."*

18. *U.S. NRC, NUREG/CR-6800, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs," March 2003.*

All Indicated Changes are in response to RAI 6-2

Table 6.10.5-8A

Loading Curve for Damaged WE14 Fuel –Single/Array Package HAC – With Control Rod Insertion (1)

Damage Fuel under HAC						
Cooling Time	***Burnup (GWd/MTU)***	***Enrichment (wt.% U-235)***	k_{keno}	σ	k_{eff}	$k_{eff}+\Delta k_x+\Delta k_i$
0 years (Fresh Fuel)	*0*	*2.95*	*0.9354*	*0.0008*	*0.9370*	*0.9370*
5 years	*5*	*3.20*	*0.9241*	*0.0005*	*0.9251*	*0.9416*
	10	***3.57***	***0.9246***	***0.0005***	***0.9256***	***0.9419***
	15	*3.81*	*0.9233*	*0.0005*	*0.9243*	*0.9415*
	20	*4.48*	*0.9238*	*0.0004*	*0.9246*	*0.9415*
	25	*5.00*	*0.9205*	*0.0005*	*0.9215*	*0.9384*
10 years	*5*	*3.20*	*0.9234*	*0.0005*	*0.9244*	*0.9409*
	10	*3.57*	*0.9243*	*0.0005*	*0.9253*	*0.9416*
	15	*3.86*	*0.9234*	*0.0005*	*0.9244*	*0.9416*
	20	*4.62*	*0.9236*	*0.0005*	*0.9246*	*0.9415*
	25	*5.00*	*0.9147*	*0.0005*	*0.9157*	*0.9326*
15 years	*5*	*3.21*	*0.9235*	*0.0005*	*0.9245*	*0.9410*
	10	*3.59*	*0.9245*	*0.0005*	*0.9255*	*0.9418*
	15	*3.90*	*0.9237*	*0.0005*	*0.9247*	*0.9419*
	20	*4.71*	*0.9235*	*0.0005*	*0.9245*	*0.9414*
	25	*5.00*	*0.9121*	*0.0005*	*0.9131*	*0.9300*
20 years	*5*	*3.23*	*0.9237*	*0.0005*	*0.9247*	*0.9412*
	10	*3.59*	*0.9242*	*0.0005*	*0.9252*	*0.9415*
	15	*3.90*	*0.9227*	*0.0005*	*0.9237*	*0.9409*
	20	*4.78*	*0.9233*	*0.0005*	*0.9243*	*0.9412*
	25	*5.00*	*0.9090*	*0.0005*	*0.9100*	*0.9269*

(1) Fuel assemblies with accumulated control rod insertion through the first 15 GWd/MTU. Fuel assemblies with accumulated control rod insertion greater than the first 15 GWd/MTU are not authorized.

Table 6.10.5-9A
Loading Curve for Damaged WE17 Fuel –Single/Array Package HAC - With Control Rod Insertion (1)

Damage Fuel under HAC						
Cooling Time	***Burnup (GWd/MTU)***	***Enrichment (wt.% U-235)***	k_{keno}	σ	k_{eff}	$k_{eff}+\Delta k_x+\Delta k_i$
0 years (Fresh Fuel)	*0*	*2.90*	*0.9380*	*0.0008*	*0.9396*	
5 years	*5*	*2.97*	*0.9240*	*0.0005*	*0.9250*	*0.9415*
	10	***3.29***	***0.9248***	***0.0005***	***0.9258***	***0.9421***
	15	*3.54*	*0.9229*	*0.0005*	*0.9239*	*0.9411*
	20	*4.21*	*0.9233*	*0.0005*	*0.9243*	*0.9412*
	25	*4.75*	*0.9237*	*0.0005*	*0.9247*	*0.9416*
	30	*5.00*	*0.9077*	*0.0005*	*0.9087*	*0.9263*
10 years	*5*	*2.99*	*0.9234*	*0.0005*	*0.9244*	*0.9409*
	10	*3.31*	*0.9242*	*0.0005*	*0.9252*	*0.9415*
	15	*3.60*	*0.9235*	*0.0005*	*0.9245*	*0.9417*
	20	*4.38*	*0.9234*	*0.0005*	*0.9244*	*0.9413*
	25	*4.91*	*0.9237*	*0.0005*	*0.9247*	*0.9416*
	30	*5.00*	*0.8987*	*0.0005*	*0.8997*	*0.9173*
15 years	*5*	*3.00*	*0.9241*	*0.0005*	*0.9251*	*0.9416*
	10	*3.34*	*0.9246*	*0.0005*	*0.9256*	*0.9419*
	15	*3.64*	*0.9230*	*0.0005*	*0.9240*	*0.9412*
	20	*4.45*	*0.9233*	*0.0005*	*0.9243*	*0.9412*
	25	*4.98*	*0.9234*	*0.0005*	*0.9244*	*0.9413*
	30	*5.00*	*0.8928*	*0.0005*	*0.8938*	*0.9114*
20 years	*5*	*3.01*	*0.9244*	*0.0005*	*0.9254*	*0.9419*
	10	*3.36*	*0.9248*	*0.0005*	*0.9258*	*0.9421*
	15	*3.66*	*0.9233*	*0.0005*	*0.9243*	*0.9415*
	20	*4.53*	*0.9242*	*0.0005*	*0.9252*	*0.9421*
	25	*5.00*	*0.9213*	*0.0005*	*0.9223*	*0.9392*

(1) Fuel assemblies with accumulated control rod insertion through the first 15 GWd/MTU. Fuel assemblies with accumulated control rod insertion greater than the first 15 GWd/MTU are not authorized.

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Table 6.10.5-27
Operational Conditions Employed in T-DEPL Calculation (1)

Parameters	Value
Average Fuel Temperature (K)	1100
Average Moderator Density (g/cc)	0.63
Average Moderator Temperature (K)	610
Average Clad Temperature (K)	640
Soluble Boron Concentration (ppm)	1000
Specific Power (MW/MTU)	40
Down Time (Days)	0

(1) Bounding upper values for fuel temperature, moderator temperature, soluble boron concentration and specific power for depletion parameters

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Table 6.10.5-28
Misload Curve Developed for 5 Years Cooling Time – WE 14x14 Fuel Class

Cooling Time	*Burnup (GWd/MTU)*	*Enrichment (wt.% U-235)*	k_{keno}	σ	k_{eff}
0 years (Fresh Fuel)	*0*	*3.35*	*0.9668*	*0.0008*	*0.9685*
5 Years	*5*	*3.67*	*0.9539*	*0.0005*	*0.9549*
	10	*4.12*	*0.9536*	*0.0005*	*0.9546*
	15	*4.40*	*0.9531*	*0.0005*	*0.9541*
	20	*5.00*	*0.9448*	*0.0005*	*0.9458*

Table 6.10.5-29
Misload Curve Developed for 5 Years Cooling Time – WE 17x17 Fuel Class

Cooling Time	*Burnup (GWd/MTU)*	*Enrichment (wt.% U-235)*	k_{keno}	σ	k_{eff}
0 years (Fresh Fuel)	*0*	*3.25*	*0.9676*	*0.0008*	*0.9691*
5 Years	*5*	*3.42*	*0.9543*	*0.0005*	*0.9553*
	10	*3.80*	*0.9544*	*0.0005*	*0.9554*
	15	*4.11*	*0.9529*	*0.0005*	*0.9539*
	20	*4.93*	*0.9539*	*0.0005*	*0.9549*
	25	*5.00*	*0.9342*	*0.0005*	*0.9352*

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***Table* 6.10.5-30**
Misload Curve Vs DOE Inventory

DOE Inventory = 103475 PWR Fuel Assemblies		
Case	*Qualified # Assemblies*	*Percentage Qualified*
WE 14x14	*103417*	*99.96*
WE 17x17	*103395*	*99.92*

All Indicated Changes are in response to RAI 6-1

Proprietary Information on Pages 6.10.5-81 and 6.10.5-82
Withheld Pursuant to 10 CFR 2.390

Chapter 7
Package Operations

NOTE: References in this Chapter are shown as [1], [2], etc., and refer to the reference list in Section 7.5. A glossary of terms used in this Chapter is provided in Section 7.6.

This Chapter contains TN-LC cask loading and unloading procedures that are intended to show the general approach to cask operational activities. The procedures in this chapter are intended to show the types of operations that will be performed and are not intended to be limiting. Site-specific conditions and requirements may require the use of different equipment and ordering of steps to accomplish the same objectives or to meet acceptance criteria to ensure the integrity of the package.

A separate operations manual (OM) will be prepared for the TN-LC cask to describe the operational steps in greater detail. The OM, along with the information in this chapter, will be used to prepare the site-specific procedures that will address the particular operational considerations related to the cask.

7.1 TN-LC Package Loading

The use of the TN-LC cask to transport fuel offsite involves (1) preparation of the empty cask for use; (2) verification that the fuel assemblies or fuel rods to be loaded in the TN-LC cask with the appropriate fuel-specific basket meet the criteria set forth in this document; (3) installation of a basket into the cask; and (4) loading fuel or placing loaded fuel buckets or pin cans in an empty TN-LC cask with the appropriate fuel-specific basket.

Offsite transport involves (1) preparation of the loaded cask for transport; (2) assembly verification leakage-rate testing of the package containment boundary; (3) placement of the cask onto a transportation vehicle; (4) installation of the impact limiters and (5) closure of the transportation container.

During shipment, the package contains any one of the TN-LC basket designs with its authorized contents as described in Chapter 1, Appendices 1.4.2 through 1.4.5. TN-LC Unit 01 shall only be loaded with the TN-LC-1FA basket with one PWR fuel assembly (Table 1.4.5-8a *or Table 1.4.5-8b*) *(including a damaged fuel assembly can if transporting a damaged PWR fuel assembly)* or one pin can with up to 21 PWR fuel rods (Table 1.4.5-10a). Procedures are provided in this section for (1) transport of the cask directly from a site spent fuel pool and (2) transport of the cask directly from a site hot cell. Appendix 7.7 contains a sub-appendix for each basket design detailing its loading procedures.

7.1.1 TN-LC Cask Preparation for Loading

Procedures for preparing the cask for use after receipt at the loading site are provided in this section and are applicable for shipment of casks loaded with any one of the basket designs and its respective approved contents.

1. Upon arrival of the empty TN-LC Package at the receiving site, perform receipt inspection. Inspect for damage, verify tamper-indicating seal is intact and perform radiation survey.
2. Open the transport container, and remove the empty TN-LC package.

All Indicated Changes are in response to RAI 6-4

3. Remove the tamper-indicating seals.
4. Remove the impact limiters from the cask.
5. Prior to removing the lid, sample the cask cavity atmosphere. If removing the lid at this stage, inspect the lid seals and sealing surfaces and verify that the O-ring seals have been replaced within the last 12 months.
6. Remove the skid tie-down assembly.
7. Take contamination smears on the outside surfaces of the cask. If necessary, decontaminate the cask.
8. The lid, bottom plug and all drain/vent/test ports incorporate O-ring seals. O-ring seals may be reused. Prior to installation, the seals and sealing surfaces shall be inspected. Verify that the seals have been replaced within the last 12 months.
9. Remove the trunnion and pocket trunnion plugs.
10. Install the two lifting trunnions in place of the front trunnions plugs. Install the trunnion bolts and torque them to the torque specified on drawing 65200-71-01, Appendix 1.4.1, following the torquing sequence shown in Figure 7-1.
11. For the specific payload to be transported as part of the TN-LC package, verify that the basket type (TN-LC-NRUX, TN-LC-MTR, TN-LC-TRIGA, or TN-LC-1FA) and spacers, if required, are appropriate for the fuel to be transported.

NOTE: TN-LC Unit 01 shall only be loaded with TN-LC-1FA basket.

12. The candidate fuel assemblies/elements or fuel rods to be transported in a specific basket must be evaluated to verify that they meet the fuel qualification requirements of the applicable fuel specification as listed in Table 7-1. *For the transportation of fuel within the TN-LC-1FA where burnup credit is employed for criticality safety, additional administrative controls to prevent misloading are also outlined in Appendix 7.7.4.*

NOTE: TN-LC Unit 01 shall only be loaded with TN-LC-1FA basket *(equipped with the damaged fuel assembly can if loading a damaged PWR fuel assembly)* with one PWR fuel assembly or one fuel rod pin can with up to 21 PWR *or BWR intact or damaged* fuel rods.

13. Prior to being placed in service, the cask is to be cleaned or decontaminated, as necessary.
14. Remove the bottom plug assembly, inspect the seals and sealing surfaces, verify that the O-ring seals have been replaced within the last 12 months, lubricate and reinstall the bottom plug assembly, torquing the bolts to the torque specified on drawing 65200-71-01, Appendix 1.4.1.
15. Remove the two test ports, the drain port and the vent port, inspect the seals and sealing surfaces, verify that the O-ring seals have been replaced within the last 12 months, reinstall each port (hand tight). The vent port on the lid may be left partially threaded to facilitate draining operations in step 14 of Section 7.1.2. The ports covers may be reinstalled over the two test ports at this time.
16. Engage the cask trunnions with the cask lifting yoke.
17. Rotate the cask to a vertical orientation, lift the cask, and place the cask in the designated preparation area.

All Indicated Changes are in response to RAI 6-3

Table 7-1
Applicable Fuel Specification for Various Fuel Types

Basket Design	Applicable Fuel Specification from Chapter 1
TN-LC-NRUX	Table 1.4.2-1 and 1.4.2-2
TN-LC-MTR	Table 1.4.3-1 thru Table 1.4.3-3
TN-LC-TRIGA	Table 1.4.4.1 thru 1.4.4-5
TN-LC-1FA	Table 1.4.5-1 thru 1.4.5-14
TN-LC-1FA in Unit 01	Table 1.4.5-1, 1.4.5-2, 1.4.5-4, *or 1.4.5-4a or 1.4.5-4b* and either 1.4.5-8a (PWR fuel assembly) or 1.4.5-10a (up to 21 rods in pin can)

Table 7-2
Appendices Containing Loading Procedures for Various TN-LC Baskets

Basket Type	Subbasket Type	Appendix	Bottom Spacer Required?
TN-LC-NRUX	—	7.7.1, Sections 7.7.1.1-2	Yes
	—	7.7.1, Sections 7.7.1.1-2	Yes
TN-LC-MTR	—	7.7.2, Sections 7.7.2.1-2	Yes
TN-LC-TRIGA	—	7.7.3, Sections 7.7.3.1-2	Yes
TN-LC-1FA	1-PWR	7.7.4, Sections 7.7.4.1-2	Yes
	1-BWR	7.7.4, Sections 7.7.4.1-2	Yes
	Pin Can	7.7.4, Sections 7.7.4.1-2	No

Table 7-3
Appendices Containing Unloading Procedures for Various TN-LC Baskets

Basket Type	Subbasket Type	Appendix
TN-LC-NRUX	—	7.7.1, Sections 7.7.1.3-4
	—	7.7.1, Sections 7.7.1.3-4
TN-LC-MTR	—	7.7.2, Sections 7.7.2.3-4
TN-LC-TRIGA	—	7.7.3, Sections 7.7.3.3-4
TN-LC-1FA	1-PWR	7.7.4, Sections 7.7.4.3-4
	1-BWR	7.7.4, Sections 7.7.4.3-4
	pin can	7.7.4, Sections 7.7.4.3-4

All Indicated Changes are in response to RAI 6-4

- *Additional Administrative Controls for Burnup Credit*

  When burnup credit is employed for demonstration of criticality safety, additional administrative controls are required for verification of fuel assembly burnup and to prevent misloading. Fuel loading plans developed above shall also include these additional requirements:

  - *A requirement to compare the reactor operating parameters for the irradiation period of the fuel assembly against those shown in Table 6.10.5-27 to ensure compliance with the isotopic depletion analysis*
  - *A requirement for no fresh fuel in pool at time of loading, or verification that fuel being loaded into the cask is not fresh, either visually or by qualitative measurement*
  - *A pool audit prior to loading, including visual verification of assembly identification numbers*
  - *Identification of highly underburned and high reactivity (Table 6.10.5-28 and Table 6.10.5-29) fuel assemblies in the pool both prior to and after loading. Alternatively, the licensee can perform a misload evaluation using the methodology and criteria described in Section 6.10.5.8 to identify these highly underburned and high reactivity fuel assemblies. This evaluation will be subject to NRC review and approval*
  - *A requirement that assemblies without visible identification number must have a quantitative confirmatory measurement prior to loading.*

As described by the procedures above, multiple barriers are included to preclude misloading events.

- A fuel movement schedule is then written, verified, and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance with the fuel movement schedule.

- *If loading damaged fuel assemblies, verify that the fuel assembly can (FAC) is installed in the TN-LC-1FA Basket.*

All Indicated Changes are in response to RAI 6-3

Listing of Computer Files Contained in Enclosure 6

Disk ID No. (size)	Discipline	System/Component	File Series (topics)	Number of Files
Enclosure 6 One Computer Hard Drive Total (94.5 MB)	**Criticality**	**TN-LC_TRITON_CR (TRITON)**	**Folder: \TRITON-CR-WE14** Folder for TRITON inputs for WE14x14 fuel class with CR insertion during depletion (input files) - See description in Section 6.10.5.6 – Note that these TRITON inputs with CR insertion were not provided in the initial submission.	17
			Folder: \TRITON-CR-WE17 Folder for TRITON inputs for WE17x17 fuel class with CR insertion during depletion (input files) – See description in Section 6.10.5.6 – Note that these TRITON inputs with CR insertion were not provided in the initial submission.	17
		TN-LC_STARBUCS_CR	**Folder: \HAC-STARBUCS-CR-WE14** Samples of SCALE6.1 STARBUCS input/output files for HAC single package using ARP library with CR insertion (input and output files) – New Table 6.10.5-8A	12
			Folder: \HAC-STARBUCS-CR-WE17 Samples of SCALE6.1 STARBUCS input/output files for HAC single package using ARP library with CR insertion (input and output files) – New Table 6.10.5-9A	12

Enclosure 7 to E-60264

Proposed Certificate of Compliance No. 9358, Revision 6 Markup

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9358	~~5~~ 6	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.~~ 10

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~~ $\boxed{10}$	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 General Assembly (4 sheets)
65200-71-70 Revision 4	TN-LC-MTR Basket Fuel Bucket (2 sheets)
65200-71-80 Revision 4	TN-LC-TRIGA Basket (5 sheets)
65200-71-90 Revision ~~6~~ 7	TN-LC-1FA Basket (5 sheets)
65200-71-96 Revision 5	TN-LC-1FA BWR Sleeve and Hold-Down Ring (2 sheets)
65200-71-102 Revision 7	TN-LC-1FA Pin Can Basket (5 sheets)
65200-71-91 Revision 0	TN-LC Transportation Cask 1FA PWR Fuel Basket Damaged Fuel Assembly Can (FAC) (3 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 ^{235}U Enrichment ≥ 90% ^{235}U Mass ≤ 380 g	66,000	740
	165,000	1120
	330,000	1440
	495,000	1680
	660,000	1950
Type B ^{235}U Enrichment ≥ 90% 380 g < ^{235}U Mass ≤ 460 g	57,750	770
	144,375	1150
	288,750	1470
	433,125	1710
	577,500	1950
Type C 40% ≤ ^{235}U Enrichment < 90% ^{235}U Mass ≤ 380 g	29,330	740
	73,325	1120
	146,650	1440
	219,975	1690
	293,300	1940
Type D 19% ≤ ^{235}U Enrichment < 40% ^{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

Damaged Fuel assemblies are fuel assemblies containing missing or partial-length fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in a fuel assembly is to be limited such that it can be handled by normal means and that a fuel pellet is not able to pass through the gap created by the cladding opening. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.

Damaged Fuel rods are complete or partial-length fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rod is to be limited such that it can be handled by normal means and that a fuel pellet is not able to pass through the gap created by the cladding opening.

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).

or damaged

intact or damaged

(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.

Damaged PWR fuel assembly is authorized for transport only if confined by use of a Fuel Assembly Can (FAC).

Intact or damaged

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®.

may be used

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).

Alternatively, in the absence of PRAs, burnup credit restrictions as shown in Table 11a and Table 11b are required for transportation of PWR fuel assemblies. Note that burnup credit is not applicable to BW 15x15 fuel class.

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] + damaged FAC (as applicable)	One intact or damaged 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 (839 kg)
Fissile Material	UO_2
Maximum Initial Uranium Content[4]	490 kg/assembly
Maximum Unirradiated Assembly Length	178.3 inches (4,528.8 mm)
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)) • 20.0 mg/cm^2 (Boral®)
Minimum number of absorber rods per PRA as a function of assembly class	Per Table 11 (Note that the use of PRAs is optional)
Burnup Credit Restriction in the absence of PRAs	Per Table 11a or Table 11b

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

Insert Table 11a and Table 11b

Table 11a
Maximum Planar Average Initial Enrichment/Minimum Burnup Combination – PWR Fuel Assembly Classes

	WE 17x17, WE 16x16, WE 15x15, CE 14x14, CE 15x15 and CE 16x16 Fuel Assembly Classes			
Fresh Fuel	2.90 wt. % U-235			
Cooling Time	**5 Years**	**10 Years**	**15 Years**	**20 Years**
Burnup (GWd/MTU)	**Fuel Initial Enrichment (wt. % U-235)**			
5	3.04	3.05	3.06	3.08
10	3.37	3.40	3.42	3.44
15	3.66	3.70	3.74	3.76
20	4.43	4.53	4.61	4.65
25	4.87	5.00	5.00	5.00

	WE 14x14 Fuel Assembly Class			
Fresh Fuel	2.95 wt. % U-235			
Cooling Time	**5 Years**	**10 Years**	**15 Years**	**20 Years**
Burnup (GWd/MTU)	**Fuel Initial Enrichment (wt. % U-235)**			
5	3.26	3.26	3.27	3.28
10	3.65	3.65	3.66	3.68
15	3.92	3.96	4.00	4.03
20	4.67	4.80	4.86	4.93
25	5.00	5.00	5.00	5.00

Table 11b

Maximum Planar Average Initial Enrichment/Minimum Burnup Combination – PWR Fuel Assembly Classes – With Control Rod Insertion [(1)]

	WE 17x17, WE 16x16, WE 15x15, CE 14x14, CE 15x15 and CE 16x16 Fuel Assembly Classes			
Fresh Fuel	2.90 wt. % U-235			
Cooling Time	**5 Years**	**10 Years**	**15 Years**	**20 Years**
Burnup (GWd/MTU)	**Fuel Initial Enrichment (wt. % U-235)**			
5	2.97	2.99	3.00	3.01
10	3.29	3.31	3.34	3.36
15	3.54	3.60	3.64	3.66
20	4.21	4.38	4.45	4.53
25	4.75	4.91	4.98	5.00
30	5.00	5.00	5.00	

	WE 14x14 Fuel Assembly Class			
Fresh Fuel	2.95 wt. % U-235			
Cooling Time	**5 Years**	**10 Years**	**15 Years**	**20 Years**
Burnup (GWd/MTU)	**Fuel Initial Enrichment (wt. % U-235)**			
5	3.20	3.20	3.21	3.23
10	3.57	3.57	3.59	3.59
15	3.81	3.86	3.90	3.90
20	4.48	4.62	4.71	4.78
25	5.00	5.00	5.00	5.00

(1) Fuel assemblies with accumulated control rod insertion through the first 15 GWd/MTU. Fuel assemblies with accumulated control rod insertion greater than the first 15 GWd/MTU are not authorized.

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. • Up to 21 BWR MOX fuel rods with physical parameters as those listed in Table 13. • 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: Initial MOX composition	• ^{235}U Content in UO_2 : 0.5 ≤ ^{235}U ≤ 0.7 wt.% • Plutonium Content: Pu / (U + Pu) ≤ 7.0 wt.% • Initial ^{239}Pu Content in PuO_2 ≤ 60.0 wt.% • Initial ^{241}Pu Content in PuO_2 ≤ 7.5 wt.%
THERMAL/RADIOLOGICAL PARAMETERS: Initial MOX Composition for Fuel Qualification Burnup and Minimum cooling time for MOX rods	• ^{238}Pu / ^{239}Pu ≤ 4.0 wt.% • ^{239}Pu/ PuO_2 ≥ 50 wt.% • ^{241}Am / PuO_2 ≤ 70 ppm • $^{235}U/U$ ≥ 0.5 wt.% Per Table 21.
Maximum Decay heat per pin can	• 2.5 kW for the pin can with up to 21 rods • 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) • 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: 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: 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) • 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

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.

, one damaged PWR fuel assembly confined by use of a Fuel Assembly Can (FAC).

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

intact

intact

or damaged

(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.

790 (see table 13)

or damaged

intact

(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.

and Fuel Assembly Can (as applicable)

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

, one damaged PWR fuel assembly confined by use of a Fuel Assembly Can (FAC).

5(c) Criticality Safety Index (CSI):

and Fuel Assembly Can (as applicable)

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.

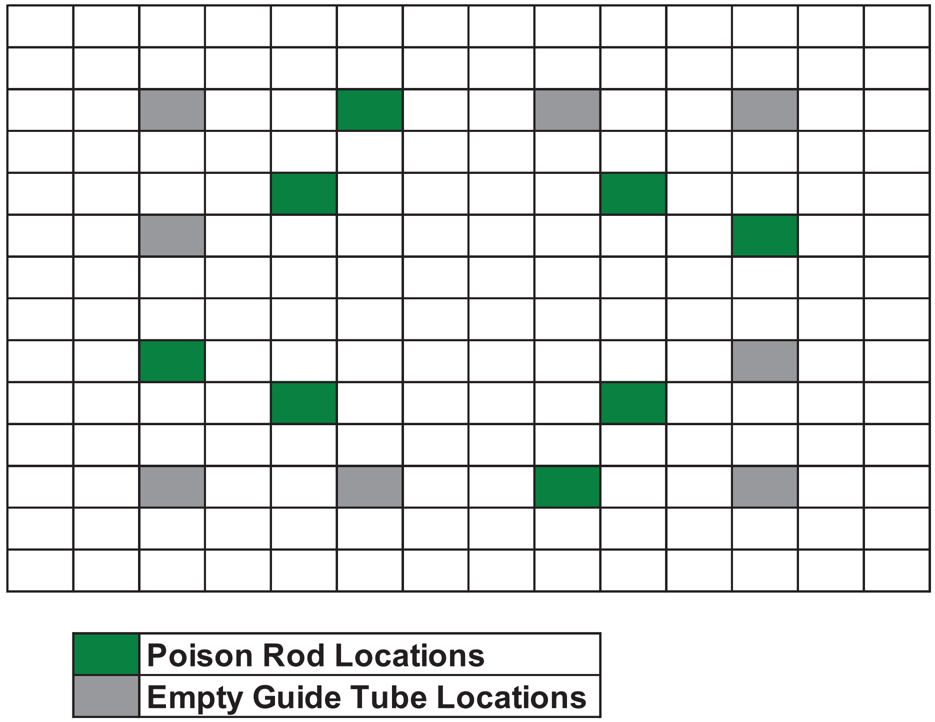
For Fuel Assemblies

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

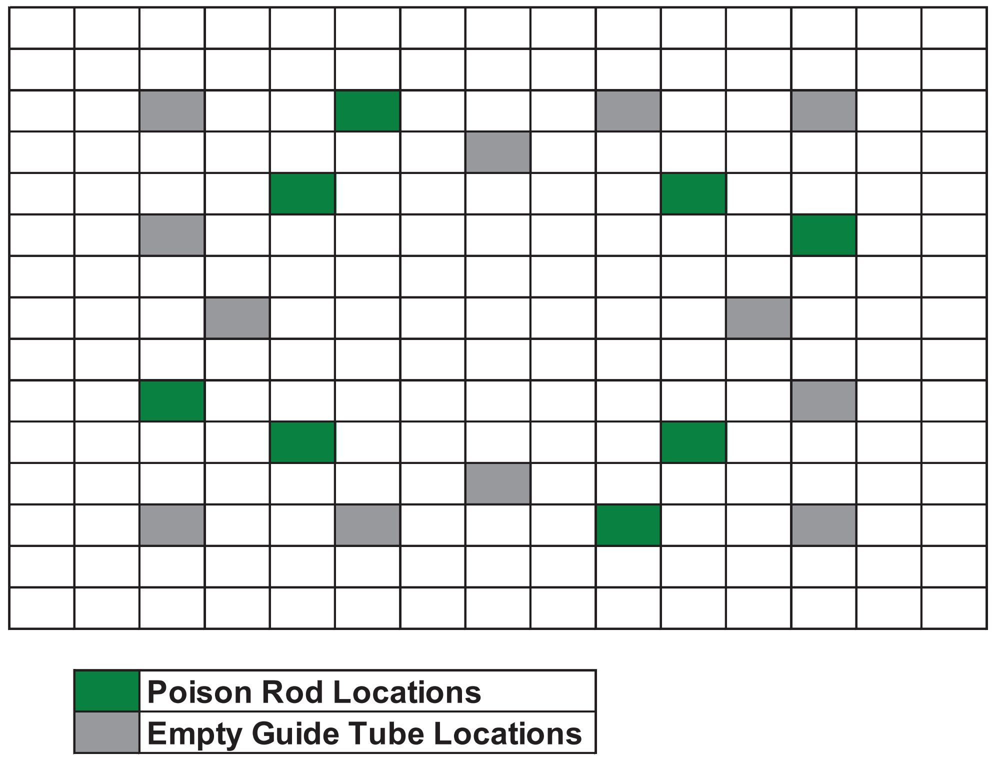
4. Burnup = Maximum burnup.
5. Enrichment = Rod Average Enrichment.
6. When transporting 21 or less fuel rods, the rods shall be placed in a specially designed pin can.
7. When transporting 9 or less fuel rods, the rods shall be placed in the 3x3 region of the pin can.
8. Fuel rods with a burnup greater 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).



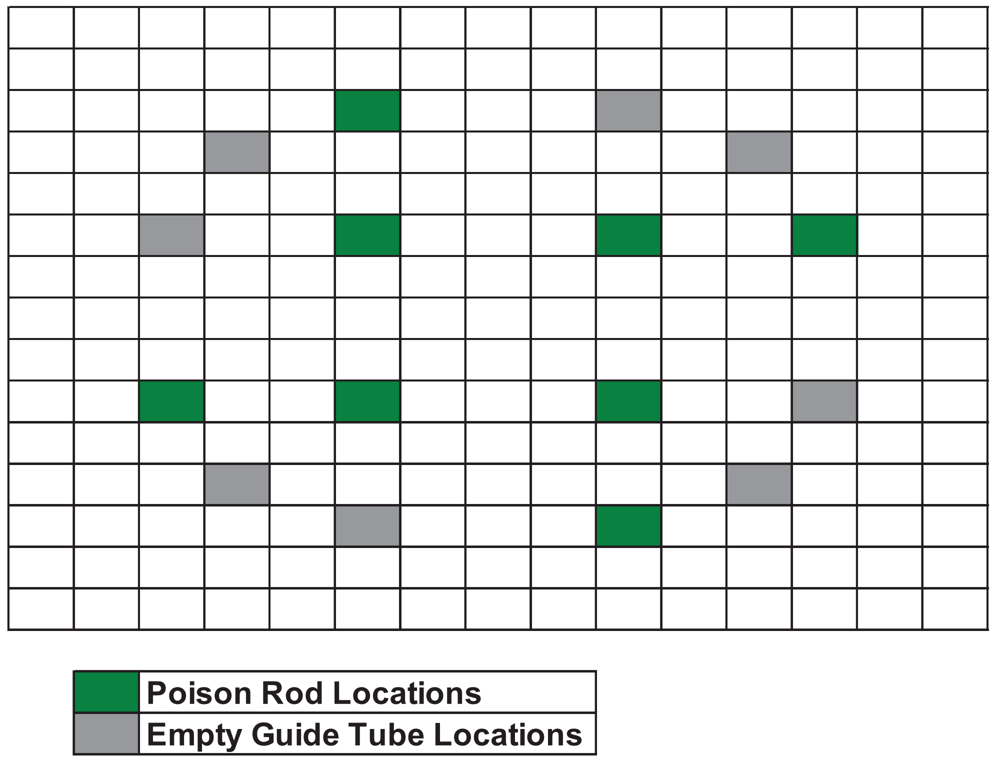
Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

Figure 1
PRA Insertion Locations for WE 14x14 Class Assemblies



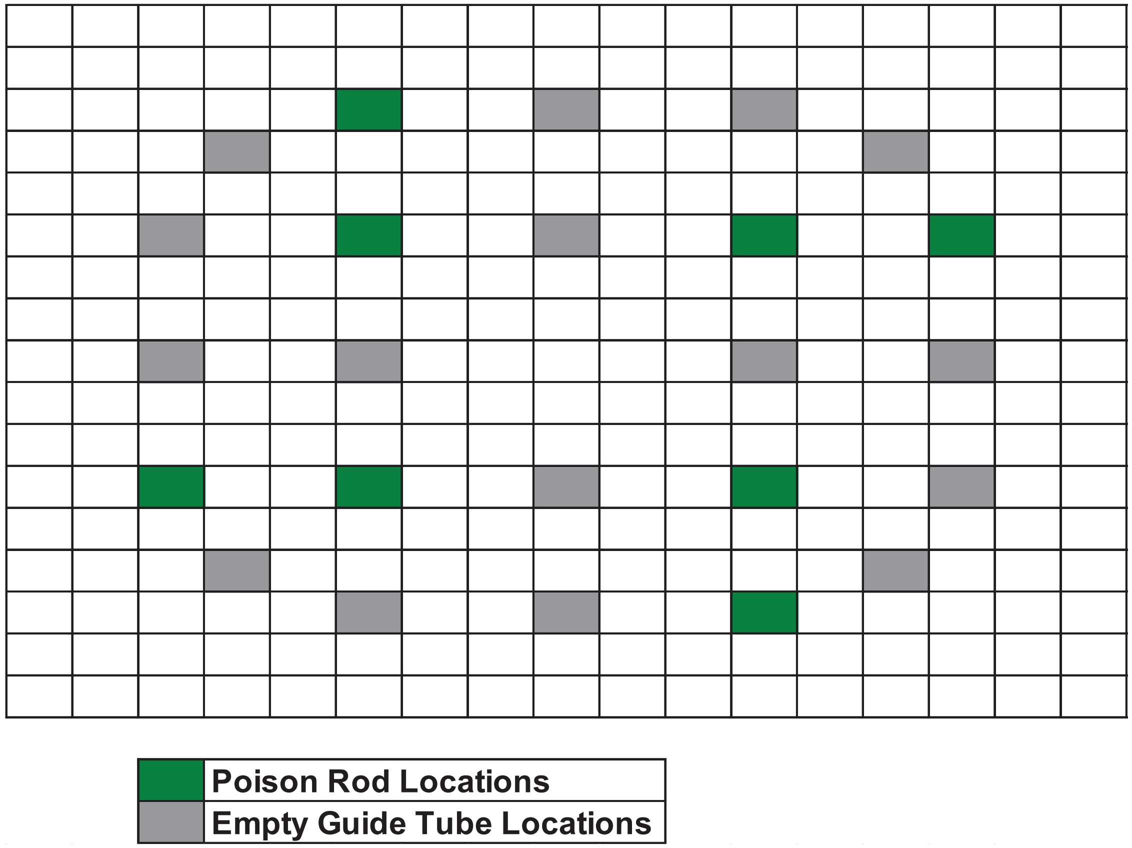
Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

Figure 2
PRA Insertion Locations for WE 15x15 Class Assemblies



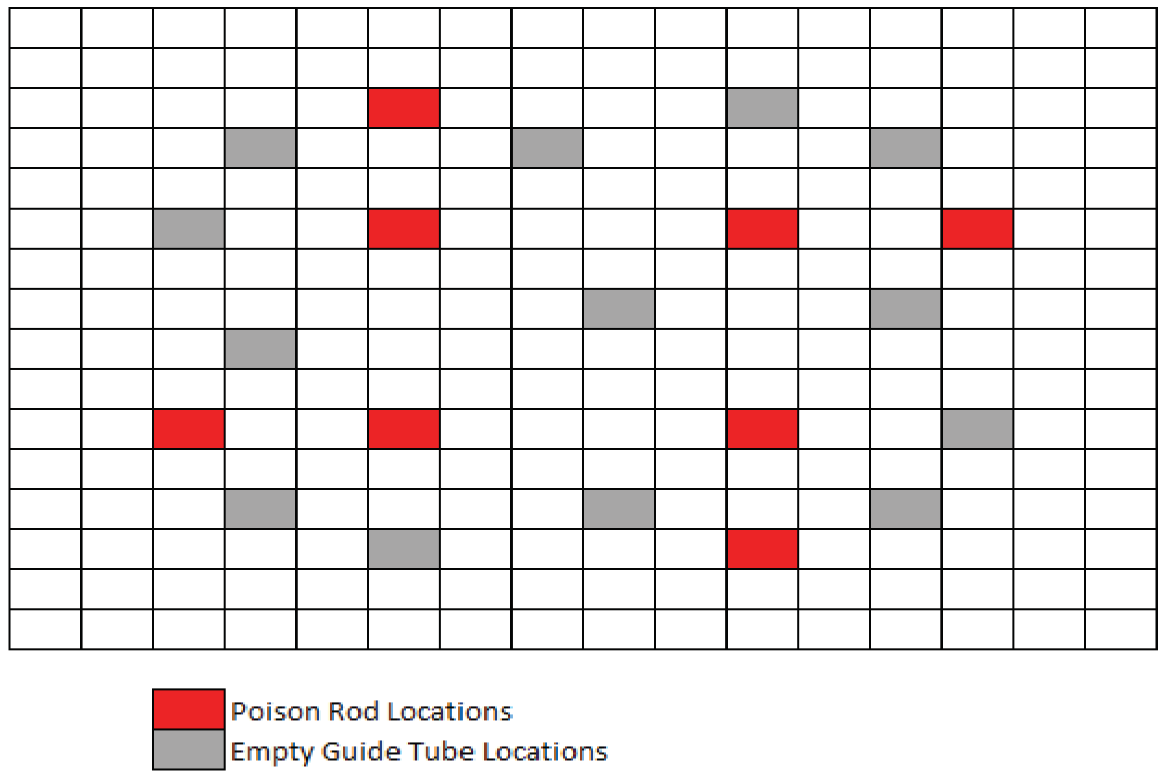
Note: This configuration indicates the relative location of the poison rods within the guide tubes and does not provide any other fuel class specific information. Any other configuration of poison rods that is rotationally symmetric is also acceptable.

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 1.0×10^{-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.

only if burn-up credit is not considered

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.

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	34	OF	34

REFERENCES

TN-LC Transportation Package Safety Analysis Report, Revision No. ~~9~~ $\boxed{10}$.

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

AFFIDAVIT PURSUANT TO 10 CFR 2.390

TN Americas LLC)
State of Maryland) SS.
County of Howard)

I, Prakash Narayanan, depose and say that I am Chief Technical Officer of TN Americas LLC, duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is listed below:

- Enclosure 3 - Portions of certain chapters and appendices of the Safety Analysis Report (SAR) for Certificate of Compliance No. 9358 TN-LC, Revision 10B, Docket 71-9358 (Proprietary Version)
- Enclosure 6 - Certain computer files associated with CoC 9358 (Proprietary)

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by TN Americas LLC in designating information as a trade secret, privileged, or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure involves certain design details associated with the SAR analyses and SAR drawings for the TN-LC System, which are owned and have been held in confidence by TN Americas LLC.

2) The information is of a type customarily held in confidence by TN Americas LLC and not customarily disclosed to the public. TN Americas LLC has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of TN Americas LLC because the information consists of descriptions of the design and analysis of a radioactive material transportation system, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with TN Americas LLC, take marketing or other actions to improve their product's position or impair the position of TN America LLC's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

A. Prakash

Prakash Narayanan
Chief Technical Officer, TN Americas LLC

Subscribed and sworn before me this 22nd day of February, 2022.

Notary Public
My Commission Expires 10 / 5 / 25

KHYNESYA TAYLOR
Notary Public
Howard County
Maryland
Commission Expires Oct. 5, 2025