January 2012

Revision LWT-12A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Docket No. 71-9225

NAC INTERNATIONAL

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February 3, 2012

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attn: Document Control Desk

Subject: Submission of a Request for an Amendment of Certificate of Compliance (CoC) No. 9225 for the NAC-LWT Cask to Incorporate SLOWPOKE fuel as Authorized Content

Docket No. 71-9225

Reference:
1. Model No. NAC-LWT Package, U.S. Nuclear Regulatory Commission (NRC) Certificate of Compliance (CoC) No. 9225, Revision 55, March 23, 2010
2. Safety Analysis Report (SAR) for the NAC Legal Weight Truck Cask, Revision 41, NAC International, April 2010

NAC International (NAC) hereby submits a request to amend the Model NAC-LWT Package CoC to incorporate the following changes:

- Include SLOWPOKE fuel elements as authorized content (up to eight canisters per cask, up to 100 undamaged and/or damaged SLOWPOKE fuel rods per canister);
- Include the shipping configuration of SLOWPOKE fuel rods loaded into 5 x 5 or 4 x 4 tube arrays with four tube arrays placed into each SLOWPOKE screened canister;
- Incorporate editorial changes in the NAC-LWT SAR; and
- Incorporate two new license drawings and update nine license drawings with minor corrections

The screened SLOWPOKE canisters may be loaded in two MTR-28 basket modules (top and top intermediate) with the three center row fuel cells fitted with cell block spacers.

This submittal includes two copies of this transmittal letter and Revision LWT-12A changed pages to the Reference 2 SAR. The changed pages incorporate the requested amendment and the applicable evaluations performed to justify the addition of the proposed contents to the authorized shipping configurations of the CoC. Attachment 1 contains a brief summary of the changes to the SAR for the amendment. Consistent with NAC administrative practice, this proposed SAR revision is numbered to uniquely identify the applicable changed pages. Revision bars mark the SAR text changes on the Revision LWT-12A pages.

NMSS01

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NAC INTERNATIONAL

Minor editorial changes have been made throughout the document, however these changes are separately described in Attachment 1 and the affected pages are included in the amendment request. The editorial changes do not affect the technical content of this submittal or the existing information in Reference 2. This submittal includes nine revised license drawings and two new license drawings. Attachment 2 to this transmittal letter lists all drawing changes in detail. The included List of Effective Pages identifies the current revision level of all pages in the Reference 2 SAR.

In order to better facilitate the review process, NAC is providing the Revision LWT-12A change pages with appropriate backing pages. Consequently, a number of Revision 41 pages are included. In accordance with NAC's administrative practices, upon final acceptance of this application, the LWT-12A changed pages will be reformatted and incorporated into the next revision of the NAC-LWT SAR.

In this amendment request, the proposed changes to the authorized contents are described in Chapter 1. The structural, thermal, shielding and criticality evaluations documenting the suitability of the NAC-LWT packaging for the requested content are presented in SAR Chapters 2, 3, 5 and 6, respectively. Chapter 7 has also been revised to address the operational and loading requirements.

Attachment 3 to this transmittal letter includes the requested changes to Reference 1.

Approval of the amendment to Reference 1 is requested by June 1, 2012, to support US Department for Energy/Savannah River Site shipping schedules planned for the summer of 2012.

If you have any comments or questions, please contact me on my direct line at 678-328-1274.

Sincerely,

Anthony L Patko

Anthony L. Patko
Director, Licensing
Engineering

Attachment 1 – List of Changes, NAC-LWT SAR, Revision LWT-12A, SLOWPOKE Amendment
Attachment 2 – List of Drawing Changes, NAC-LWT SAR, Revision LWT-12A, SLOWPOKE Amendment
Attachment 3 – Proposed Changes for Revision 56 of Certificate of Compliance No. 9225 for the NAC-LWT Cask SLOWPOKE Amendment

Enclosure

ED20120023

Attachment 1

List of Changes

NAC-LWT SAR, Revision LWT-12A

SLOWPOKE Amendment

List of Changes, NAC-LWT SAR, Revision LWT-12A

Note: The List of Effective Pages and the Chapter Tables of Contents, including the List of Figures and the List of Tables, were revised as needed to incorporate the following changes.

Chapter 1

- Page 1-1, second paragraph, bulleted list – added the last bullet stating, "up to eight (8) SLOWPOKE Fuel Canisters."
- Page 1-2, Approved Contents table – added the last row in the Approved Contents table for "SLOWPOKE Fuel Rods in Fuel Canisters" at 0.0 CSI
- Page 1-4, Table 1.1-1 – added the second to last bullet stating, "up to eight (8) SLOWPOKE Fuel Canisters each containing up to 100 undamaged and/or damaged SLOWPOKE fuel rods"
- Page 1.1-2, bulleted list – added the second to last bullet stating, "up to 800 SLOWPOKE undamaged and/or damaged fuel rods contained in up to eight (8) fuel canisters (up to 100 fuel rods each); or,"
- Pages 1.1-2 through 1.1-3 – added the last paragraph on page 1.1-2, which carries over to the top of page 1.1-3, describing the loading of SLOWPOKE undamaged and/or damaged fuel rods
- Page 1.2-5, Section 1.2.3, item 4 – added "5.0 W for SLOWPOKE fuel rods" to the last sentence in item 4, after MOATA bundles and before solid, nonfissile, irradiated hardware
- Page 1.2-7 – added item 21 to the top of the page to describe the SLOWPOKE fuel packaging
- Pages 1.2-18 through 1.2-19 – added Section 1.2.3.12, "SLOWPOKE Fuel Rods"
- Pages 1.2-20 through 1.2-37 – included only for text flow change effecting page numbering
- Page 1.2-38 – added Figure 1.2.3-19, "SLOWPOKE Fuel Element"
- Pages 1.2-39 through 1.2-53 – included only for text flow change effecting page numbering
- Page 1.2-54 – added Table 1.2-14, "SLOWPOKE Fuel Rods"
- Note: Two new drawings were added to the end of Section 1.4: Drawing 315-40-156, Rev. 1, "Canister Assembly SLOWPOKE Fuel"; Drawing 315-40-158, Rev. 0, "Legal Weight Truck Transport Cask Assy, SLOWPOKE Fuel"

Chapter 2

- Page 2.2.1-2, Table 2.2.1-1, Footnote 1 – added the third sentence in the first footnote to state: "The maximum weight for the SLOWPOKE canister is 23 pounds."
- Page 2.6.12-7, Section 2.6.12.6 – changed the number of MTR modular basket assembly configurations listed in the first sentence from four to five

- Page 2.6.12-8, first paragraph – added the second full sentence on the page to state, "The fifth configuration is for up to 800 SLOWPOKE intact and/or damaged fuel rods contained in canisters"; added the sixth and seventh full sentences in the first paragraph to state: "Up to four (4) SLOWPOKE canisters may be loaded in each of the top and the upper intermediate MTR-28 basket modules with the center fuel cells blocked. The lower intermediate and bottom basket modules are installed as axial spacers."; added "or Table 1.2.3-14" to the last sentence in the first paragraph
- Pages 2.6.12-95 through 2.6.12-106 – added Section 2.6.12.13, "SLOWPOKE Fuel Canister Assembly," including Table 2.6.12-3 and Figures 2.6.12-9 through 2.6.12-10
- Page 2.6.12-107 – changed the section number from 2.6.12.13 to 2.6.12.14 for the "Conclusion" section and added "and +8.5 as reported in Section 2.6.12.13 for the SLOWPOKE fuel canister assembly" to the end of the second to last sentence.
- Page 2.7.7-8, Section 2.7.7.6 – changed the number of MTR modular basket assembly configurations listed in the first sentence from four to five; added the fourth, fifth, and sixth sentences in the first paragraph in the section to state: "The fifth MTR basket configuration is for SLOWPOKE fuel rods contained in canisters. For SLOWPOKE fuel, up to four (4) canisters (each containing up to 100 SLOWPOKE intact and/or damaged fuel rods) may be loaded in both the top and upper intermediate MTR-28 basket modules. The lower intermediate and bottom MTR-28 basket modules are installed as axial spacers."
- Pages 2.7.7-74 through 2.7.7-88 – added Section 2.7.7.15, "SLOWPOKE Fuel Canister Assembly," including Tables 2.7.7-3 through 2.7.7-5 and Figures 2.7.7-2 through 2.7.7-7

Chapter 3

- Page 3.1-1, Section 3.1, third paragraph – added the last four sentences in the third paragraph to support the SLOWPOKE fuel content
- Page 3.1-2 – included only for text flow change effecting page numbering
- Pages 3.4-37 through 3.4-38 – added Section 3.4.1.18, "SLOWPOKE Fuel Analytical Model" and Section 3.4.1.18.1, "SLOWPOKE Fuel Heat Transfer Analyses Results" to support the SLOWPOKE fuel content
- Page 3.4-39, Section 3.4.2 – added the last sentence in Section 3.4.2 to reference a new table (Table 3.4-27) added to support the SLOWPOKE fuel content
- Pages 3.4-40 through 3.4-99 – included only for text flow change effecting page numbering
- Page 3.4-100 – added a new table – Table 3.4-27, "Maximum Component Temperatures – SLOWPOKE Fuel in MTR Basket" – to support the SLOWPOKE fuel content
- Page 3.5-13 – added Section 3.5.3.16, "Evaluation of SLOWPOKE Fuel Contents" to support the SLOWPOKE fuel content
- Pages 3.5-14 through 3.5-19 – included only for text flow change effecting page numbering

Chapter 5

- Pages 5.3.20-1 through 5.3.20-28 – Added new Section 5.3.20, SLOWPOKE Fuel Configuration, including Figures 5.3.20-1 through 5.3.20-9 and Tables 5.3.20-1 through 5.3.20-9

Chapter 6

- Page 6-1, Section 6 – added "or up to 800 SLOWPOKE rods" in the second to last sentence of the first paragraph, and added the last row to the "Criticality Safety Index" table at the bottom of the page for the SLOWPOKE fuel rods
- Page 6.1-5 – added the last (third) paragraph on the page to summarize the SLOWPOKE fuel rod analyses
- Page 6.5.1-1 – changed the title of Section 6.5 from "Critical Benchmarks" to "Criticality Benchmarks" for consistency; removed the last sentence in the first paragraph under Section 6.5 since the section references listed in the sentence are no longer accurate; changed the title of Section 6.5.1 from "PWR and BWR Fuel Assemblies" to "CSAS25 Criticality Benchmark for LEU LWR Oxide Fuel"
- Page 6.5.2-1 – changed the title of Section 6.5.2 from "MTR and DIDO Fuel Elements" to "CSAS25 Criticality Benchmarks for Research Reactor Fuel Elements (MTR and DIDO)", and added the second sentence in the section to state that TRIGA fuel is not included in this section
- Page 6.5.3-1 – changed the title of Section 6.5.3 from "TRIGA Fuel Elements" to "CSAS25 Criticality Benchmarks for TRIGA Fuel Elements"
- Pages 6.5.4-1 through 6.5.4-46 – moved Section 6.7.2 in its entirety to Section 6.5.4; changed the section title to Section 6.5.4, "MCNP Criticality Benchmarks LEU Oxide and MOX LWR Fuels" (was previously Section 6.7.2, "Critical Benchmarks"); subsection, figure, and table numbers (and their references) were changed accordingly
- Pages 6.5.5-1 through 6.5.5-15 – added Section 6.5.5, "MCNP Criticality Benchmarks for Research Reactor Fuels," including subsections 6.5.5.1 through 6.5.5.3, Figures 6.5.5-1 through 6.5.5-4, and Tables 6.5.5-1 through 6.5.5-5
- Pages 6.7.2-1 through 6.7.2-16 – added a new section, Section 6.7.2, "SLOWPOKE Fuel Rods," including subsections 6.7.2.1 through 6.7.2.3, Figures 6.7.2-1 through 6.7.2-5 and Tables 6.7.2-1 through 6.7.2-9 (the previous Section 6.7.2 was moved to Section 6.5.4)

List of Changes, NAC-LWT SAR, Revision LWT-12A (cont'd)

Chapter 6 Appendices

- Pages 6.6.17-1 through 6.6.17-7 – added Section 6.6.17, "SLOWPOKE Fuel MCNP Input," including Figures 6.6.17-1 and 6.6.17-2

Chapter 7

- Pages 7.1-57 through 7.1-62 – added Section 7.1.13, "Procedures for Dry Loading of MTR-28 Basket Modules Containing SLOWPOKE Fuel Canisters into the NAC-LWT Cask"
- Pages 7.1-63 through 7.1-64 – added Section 7.1.14, "Procedures for Loading AECL SLOWPOKE Fuel Rod Contents Into the SLOWPOKE Fuel Canister"
- Page 7.2-6 – changed the 7.2.3 section title from "Procedure for Wet Unloading of MTR, TRIGA, DIDO, ANSTO, or PULSTAR Fuel Basket Contents" to "Procedure for Wet Unloading of MTR, TRIGA, DIDO, ANSTO, PULSTAR, or SLOWPOKE Fuel Basket Contents"; changed the first sentence from "The procedure for the unloading of MTR, TRIGA, DIDO, ANSTO, ANSTO-DIDO, or PULSTAR fuel basket contents from the package in a spent fuel pool is as follows" to "The procedure for the unloading of MTR, TRIGA, DIDO, ANSTO, ANSTO-DIDO, PULSTAR, or SLOWPOKE fuel basket contents from the package in a spent fuel pool is as follows"
- Page 7.2-7, step 19 – added "or SLOWPOKE" to the first sentence and changed all instances of "DFCs" to "fuel canisters"
- Page 7.2-9 – changed Section 7.2.4 title from "Procedure for Dry Unloading of MTR, TRIGA, DIDO, ANSTO, or PULSTAR Fuel Contents" to "Procedure for Dry Unloading of MTR, TRIGA, DIDO, ANSTO, PULSTAR, or SLOWPOKE Fuel Contents"; changed the first sentence to add "or SLOWPOKE"
- Page 7.2-10, step 18 – added "or SLOWPOKE" to the first line

List of Changes, NAC-LWT SAR, Revision LWT-12A (Non-SLOWPOKE related editorial changes)

Chapter 6

- Page 6.1-5 – changed "critical benchmark" to "criticality benchmark" in the last sentence of the first paragraph for consistency
- Page 6.4.8-3, Section 6.4.8.9, last sentence of the first paragraph – changed "critical benchmarks" to "criticality benchmarks" for consistency
- Page 6.4.9-3, Section 6.4.9.5, third and fourth sentences – changed "critical benchmarks" to "criticality benchmarks" for consistency
- Page 6.5.2-2 – made an editorial correction to the criticality safety limits equation description to correct "Δk_{BU}"
- Page 6.5.4-2, last sentence – changed "critical benchmarks" to "criticality benchmarks" for consistency
- Page 6.7.1-1, Section 6.7.1, first sentence – changed "critical benchmarks" to "criticality benchmarks" for consistency

Chapter 7

- Page 7.1-4, step 9 – deleted the second sentence in step 9 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Page 7.1-15, step 9 – deleted the second sentence in step 9 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Page 7.1-36, step 9 – deleted the second sentence in step 9 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Page 7.1-40, step 8 – changed "50 ± 10 inch-pounds" to "50 ± 2 ft-lbs" in order to provide the proper value for the DFC drain plug torque.
- Page 7.1-43, step 9 – deleted the second sentence in step 9 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Page 7.1-47, step 10 – deleted the second sentence in step 10 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Page 7.1-54, step 9 – deleted the second sentence in step 9 that stated "If leakage is detected or suspected, verify shield tank fluid level and correct, as required" because the sentence is not applicable to cask field operations
- Pages 7.3-1 through 7.3-2, Section 7.3 – deleted Section 7.3, "Procedures for Preparation of the Empty Package for Transport" per NRC recommendation to remove parts of the SAR that address handling of empty cask

Attachment 2

List of Drawing Changes

NAC-LWT SAR, Revision LWT-12A

SLOWPOKE Amendment

Drawing 315-40-04, LWT Transport Cask Lid Assembly SAR, Revision 12

1. B.O.M., Item 7, Revise Description to "Shamban S11214-0460"; was "Shamban S11214-460."
2. B.O.M., Item 2, Revise Drawing No. to "315-40-08-19"; was "315-40-08-21."
3. For Item #3, O-ring, delete "Viton" from material, and "Parker 3-904V747-75" from the description, and add reference to "Drawing No. 315-40-02-6."
4. For Item #4, Thread Insert, delete information in the material, specification and description columns, and add reference to "Drawing No. 315-40-08-13."
5. Revise "Material" for Item # 8, O-ring, from "STSTL" to "X-750."
6. Title Block, Revise scale callout to "N.T.S."; was "1/4."
7. Zone A7 Detail B-B, Revise Scale callout to "Scaled"; was "Scale: 2/1" and Zones C6 & C8 Sections E-E & D-D, Revise Scale callout to "Scaled"; was "2/1."
8. Title Block, Delete weight callout "938 lbs."
9. Title Block, Update to current title block.

Note: Items 3, 4, and 5 were incorporated prior to Revision 12.

Drawing 315-40-087, Canister Lid Assembly, Sealed Failed Fuel Can, TRIGA Fuel, Revision 7

1. Revise Delta Note 3, to: "...Article NG-5350."; was "...Article NB-5350."
2. Item 12 (Lift Lug) in Zone C6, change "R.5" dimension to "R.3" and Zone C7, add "TYP" to 45° X .13 dimension.

Drawing 315-40-088, Canister Body Assembly, Sealed Failed Fuel Can, TRIGA Fuel, Revision 3

1. Revise Delta Note 3, to: "...Article NG-5350."; was "...Article NB-5350." And Zone E7, Delete Delta symbol 3 callout.
2. Revise Title Block scale factor in "Scale" box to "N.T.S."; was "FULL", delete word "NOTED" in "Weight" box and Zones C4 & D4 Delete weight callout for Assy 99 & 98.
3. Replace border and title block with newer version.

Drawing 315-40-104, Legal Weight Truck Transport Cask Assy, PWR/BWR Rod Transport Canister, Revision 6

1. B.O.M., Items 14 & 15, Delete Qty "4" from Assy 96.

Drawing 315-40-128, Legal Weight Truck Transport Cask Assy, TPBAR Shipment, Safety Analysis Report, Revision 4

Sheet 1:

1. Add "N.T.S." to Title Block.

Sheet 2:

2. Revise Title Block to "N.T.S."; was "1/8."

Drawing 315-40-129, Canister Body Assembly, Failed Fuel Can, PULSTAR, Revision 2

1. Revise Delta Note 3, to: "...Article NG-5350."; was "...Article NB-5350."
2. Revise Title Block on drawing to "Canister Body Assembly, Sealed Failed Fuel Can, Pulstar"; was "Canister Body Assembly, Failed Fuel Can, Pulstar."
3. Zone E7, Delete Delta symbol 3 callout.
4. Add "N.T.S." to scale callout in title block.

Drawing 315-40-130, Assembly, Failed Fuel Can, PULSTAR, Revision 2

1. Revise Title block on drawing to "Assembly, Sealed Failed Fuel Can, Pulstar"; was Assembly, Failed Fuel Can, Pulstar."
2. Title block scale callout, Add "N.T.S."

Drawing 315-40-133, Transport Cask Assembly, PULSTAR Shipment, LWT Cask, Revision 2

1. Title block scale callout, Add "N.T.S." on sheets 1 & 2.

Sheet 1:

2. Revise Delta Note 5 to "...Article NG-5350."; was "...Article NB-5350."
3. B.O.M., Item 11, Revise Name to "Sealed Failed Fuel Can"; was "Failed Fuel Can."

Sheet 2:

4. Revise Delta symbol 6 to 5.

Drawing 315-40-134, Body Weldment, Screened Fuel Can, PULSTAR Fuel, Revision 2

1. Revise Delta Note 1 to "...Article NG-5350."; was "...Article NB-5350."
2. Title block scale callout, Add "N.T.S."

Attachment 3

Proposed Changes for Revision 56 of Certificate of Compliance

No. 9225 for NAC-LWT Cask

SLOWPOKE Amendment

Drawings (new)

CoC Page 4 of 31:

LWT 315-40-156, Rev. 1 (Sheets 1-4)
"Canister Assembly SLOWPOKE Fuel"

LWT 315-40-158, Rev. 0
"Legal Weight Truck Transport Cask Assy SLOWPOKE Fuel"

Drawings (revised)

CoC Page 2 of 31:

LWT 315-40-04, Rev. 10 **to** Rev. 12

CoC Page 3 of 31:

LWT 315-40-104, Rev. 5 (Sheets 1-3) **to** Rev. 6 (Sheets 1-3)
LWT 315-40-129, Rev. 1 **to** Rev. 2
LWT 315-40-130, Rev. 1 **to** Rev. 2
LWT 315-40-133, Rev. 1 (Sheets 1-2) **to** Rev. 2 (Sheets 1-2)
LWT 315-40-134, Rev. 1 **to** Rev. 2

CoC Page 30 of 31:

315-40-128, Rev. 3 **to** Rev. 4 (Sheets 1-2)

CoC Sections (new)

CoC Page 19 of 31:

5.(b)(1) Type and form of material (continued)

(xviii) Damaged or undamaged SLOWPOKE fuel rods, including fuel pellets, pieces, and debris as specified below.

Parameter	Limiting Values
Maximum cask heat load (W)	5
Maximum canister heat load (W)	0.625
Payload limit (lb/canister)	23.1
Maximum ^{235}U per rod (g)	2.800
Maximum U per rod (g)	3.111
Minimum cool time (yr)	14
Maximum burnup (GWd/MTU or	30
wt% ^{235}U Depletion	4.5

CoC Page 27 of 31:

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

(xix) For the SLOWPOKE fuel described in Item 5.(b)(1)(xviii):

Up to 100 SLOWPOKE fuel rods (or the equivalent quantity of damaged material) may be loaded per SLOWPOKE canister in accordance with NAC Drawing No. 315-40-156 utilizing either a 4x4 or 5x5 tube array or any combination, thereof. Up to 4 SLOWPOKE canisters may be loaded within a 28 MTR fuel basket assembly with the three center fuel cells blocked. Only the top and top intermediate fuel baskets may be loaded with SLOWPOKE fuel. Cask configuration to be in accordance with NAC Drawing No. 315-40-158.

CoC Sections (revised)

CoC Page 29 of 31:

5(c) Criticality Safety Index (CSI)

For (canned) SLOWPOKE fuel described in 5(b)(1)(xviii) and limited in 5.(b)(2)(xix)	0.0

6. Known or suspected damaged fuel assemblies (rods) or elements, and fuel with cladding defects greater than pin holes and hairline cracks are not authorized, except as described in Items 5.(b)(1)(x); 5.(b)(1)(xiv); 5.(b)(1)(xviii); 5.(b)(2)(iv)(d); 5.(b)(2)(vi); 5.(b)(2)(vii)(a); 5.(b)(2)(vii)(b); 5.(b)(2)(viii); 5.(b)(2)(ix); 5.(b)(2)(x); 5.(b)(2)(xi); 5.(b)(2)(xiv); 5.(b)(2)(xv); and 5.(b)(2)(xix).

CoC Page 31 of 31

19. Revision 55 of this certificate may be used until December 31, 2013.

REFERENCES

NAC International, Inc., application dated January 22, 2010.

NAC International, Inc., supplements dated February 9 and 23 and March 3, 2010 and February 3, 2012.

January 2012

Revision LWT-12A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Docket No. 71-9225

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LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

Table of Contents

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List of Tables

List of Drawings

* Packaging Unit Nos. 1, 2, 3, 4 and 5 are constructed in accordance with this revision of drawing.

List of Drawings (continued)

1 GENERAL INFORMATION

This chapter of the NAC International, Legal Weight Truck spent fuel shipping cask (NAC-LWT) Safety Analysis Report (SAR) presents a general introduction to, and description of, the NAC-LWT cask. Terminology used throughout this report is presented in Table 1.1-1.

Shipment of the NAC-LWT cask by truck, ISO container, and/or by railcar, as a Type B(U)F-96 package, as defined in 10 CFR 71.4, is authorized for the following contents:

- PWR and BWR fuel assemblies[1];
- MTR fuel assemblies and plates;
- DIDO fuel assemblies;
- metallic fuel rods;
- 25 high burnup PWR and BWR fuel rods (including up to 14 fuel rods classified as damaged)[2];
- 16 PWR MOX fuel rods (or mixed load of up to 16 PWR MOX and UO_2 PWR fuel rods) and up to 9 burnable poison rods (BPRs);
- TRIGA fuel elements and TRIGA fuel cluster rods;
- General Atomics (GA) High-Temperature Gas-Cooled Reactor (HTGR) and Reduced-Enrichment Research and Test Reactor (RERTR) Irradiated Fuel Materials (IFM);
- up to 700 PULSTAR fuel elements;
- spiral fuel assemblies;
- MOATA plate bundles; and,
- up to eight (8) SLOWPOKE Fuel Canisters.

The authorized contents previously listed include both irradiated and unirradiated forms of the materials.

Irradiated hardware is also authorized to be shipped in the NAC-LWT cask by truck, ISO container, and/or by railcar, as a Type B(U)F-96 package, as defined in 10 CFR 71.4. Irradiated hardware is defined as solid, irradiated and contaminated fuel assembly structural or reactor internal component hardware, which may include fissile material, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c).

Shipment of the NAC-LWT cask by truck, ISO container, and/or by railcar, as a Type B(M)-96 package, as defined in 10 CFR 71.4, is also authorized for the following contents:

- up to 300 Tritium Producing Burnable Absorber Rods (TPBARs), of which two can be prefailed; and

[1] NAC-LWT casks containing PWR and BWR fuel assemblies are to be transported on an open trailer with a personnel barrier.

[2] PWR and BWR fuel rods may be transported in either a fuel assembly lattice (skeleton) or in a fuel rod insert. The fuel rod insert may contain PWR instrument/guide tubes and BWR water/inert rods in addition to the fuel rods.

- up to 55 TPBARs segmented during post-irradiation examination (PIE), including segmentation debris.

In accordance with 10 CFR 71.59, the NAC-LWT cask is assigned a Criticality Safety Index (CSI) for criticality control of the approved contents as follows:

Approved Contents	CSI
PWR fuel assemblies	100
BWR fuel assemblies	5.0
MTR fuel elements	0.0
Metallic fuel rods	0.0
TRIGA fuel elements (in poisoned TRIGA fuel baskets)	0.0
TRIGA fuel elements (in nonpoisoned TRIGA fuel baskets)	12.5
TRIGA fuel cluster rods	0.0
High burnup PWR rods	0.0
High burnup BWR rods	0.0
PWR MOX rods	0.0
DIDO fuel elements	12.5
General Atomic Irradiated Fuel Material (GA IFM)	0.0
TPBARS and segmented TPBARS	0.0
Intact (uncanned) PULSTAR fuel	0.0
Canned PULSTAR fuel	33.4
ANSTO fuel	0.0
Solid irradiated hardware	0.0
ANSTO-DIDO fuel combination	0.0
SLOWPOKE Fuel Rods in Fuel Canisters	0.0

TPBARs do not contain fissile material and criticality assessments are not required. Solid, irradiated and contaminated hardware contents could include fissile material not exceeding a Type A quantity and the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c). A CSI of 0 is assigned for these contents for documentation purposes.

The estimated Transport Index (TI) for shielding for the prior listed contents is shown in Table 5.1.1-1. The actual TI for individual shipments will be determined in accordance with 10 CFR 71.4 by the licensee.

Table 1.2.1-1 Terminology and Notation

Cask Model	NAC-LWT
Package	The Packaging with its radioactive contents (payload), as presented for transportation (10 CFR 71.4). Within this report, the Package is denoted as the NAC-LWT cask or simply as the cask.
Packaging	The assembly of components necessary to ensure compliance with packaging requirements (10 CFR 71.4). Within this report, the Packaging is denoted as the NAC-LWT cask.
NAC-LWT Cask	This packaging consists of a spent-fuel shipping cask body and closure lid with energy absorbing impact limiters.
Contents (Payload)	• 1 PWR assembly • up to 2 BWR assemblies • up to 25 PWR or BWR rods (including high burnup fuel rods and up to 14 fuel rods classified as damaged)[1] • up to 16 PWR MOX fuel rods (or mixed contents of up to 16 PWR MOX and UO_2 PWR fuel rods) and up to 9 BPRs • up to 42 MTR fuel elements (including plates) • up to 42 DIDO fuel assemblies • up to 7 degraded clad DIDO fuel assemblies in damaged fuel cans (DFCs) in ANSTO top basket module • up to 15 sound (cladding intact) metallic fuel rods • up to 9 damaged metallic fuel rods or 3 severely damaged metallic fuel rods in filters • up to 140 intact or damaged TRIGA fuel elements/debris • up to 560 intact or damaged TRIGA fuel cluster rods • 2 GA IFM packages • up to 300 TPBARs (including up to 2 prefailed TPBARs) • up to 55 TPBARs segmented into individual segments and segmentation debris • up to 700 intact or damaged PULSTAR fuel elements in either assembly or element form, including fuel debris • up to 42 intact spiral fuel assemblies (also referred to as Mark III spiral fuel), including up to 7 degraded clad spiral fuel assemblies in DFCs. Spiral fuel assemblies may be cropped. • up to 42 intact MOATA plate bundles, including up to 7 MOATA plate bundles in DFCs

[1] PWR and BWR fuel rods may be transported in either a fuel assembly lattice (skeleton) or in a fuel rod insert. The fuel rod insert may contain PWR instrument/guide tubes and BWR water/inert rods in addition to the fuel rods.

Table 1.1-1 Terminology and Notation (cont'd)

	• any combination of individual ANSTO basket modules containing either spiral fuel assemblies or MOATA plate bundles up to a total of 42 assemblies/bundles, including up to 7 degraded clad DIDO, spiral or MOATA elements/ bundles in DFCs placed in an ANSTO top basket module • combination ANSTO-DIDO basket assembly (one ANSTO top module and five DIDO intermediate and base basket modules) containing up to 42 DIDO, spiral or MOATA elements/bundles with up to 7 degraded clad elements/bundles in the ANSTO top module in DFCs • up to eight (8) SLOWPOKE Fuel Canisters each containing up to 100 undamaged and/or damaged SLOWPOKE fuel rods • up to 4,000 lbs of solid, irradiated and contaminated hardware, which may include fissile material less than a Type A quantity and meeting the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c). Total allowed mass includes the weight of spacers, shoring and dunnage.
Impact Limiters	Aluminum honeycomb energy absorbers located at the ends of the cask.
Intact LWR Fuel (Assembly or Rod)	Spent nuclear fuel that is not Damaged LWR Fuel, as defined herein. To be classified as intact, fuel must meet the criteria for both intact cladding and structural integrity. An intact fuel assembly can be handled using normal handling methods, and any missing fuel rods have been replaced by solid filler rods that displace a volume equal to, or greater than, that of the original fuel rod.
Damaged LWR Fuel (Assembly or Rod)	Spent nuclear fuel that includes any of the following conditions that result in either compromise of cladding confinement integrity or recognition of fuel assembly geometry. 1. The fuel contains known or suspected cladding defects greater than a pinhole leak or a hairline crack that have the potential for release of significant amounts of fuel particles. 2. The fuel assembly: i. is damaged in such a manner as to impair its structural integrity; ii. has missing or displaced structural components such as grid spacers;

1.1 Introduction

The NAC-LWT spent-fuel shipping cask has been developed by NAC International (NAC) as a safe means of transporting radioactive materials authorized as approved contents. The cask design is optimized for legal weight over the road transport, with a gross weight of less than 80,000 pounds. The cask provides maximum safety during the loading, transport, and unloading operations required for spent-fuel shipment. The NAC-LWT cask assembly is composed of a package that provides a containment vessel that prevents the release of radioactive material. The actual containment boundary provided by the package consists of a 4.0-inch thick bottom plate, a 0.75-inch thick, 13.375-inch inner diameter shell, an upper ring forging, and an 11.3-inch thick closure lid. The cask lid closure is accomplished using twelve, 1-inch diameter bolts. The cask has an outer shell, 1.20 inches thick, to protect the containment shell and also to enclose the 5.75-inch thick lead gamma shield. Neutron shielding is provided by a 5.0-inch thick neutron shield tank with a 0.24-inch (6mm) thick outer wall, containing a water/ethylene glycol mixture and 1.0 minimum weight percent (wt %) boron (58 wt % ethylene glycol; 39 wt % demineralized water; 3 wt % potassium tetraborate [$K_2B_4O_7$]). The neutron shield tank system includes an expansion tank to permit the expansion and contraction of the shield tank liquid without compromising the shielding or overstressing the shield tank structure. Aluminum honeycomb impact limiters are attached to each end of the cask to absorb kinetic energy developed during a cask drop, and limit the consequences of normal operations and hypothetical accident events.

The NAC-LWT is a legal weight truck cask designed to transport the following contents:

- 1 PWR assembly;
- up to 2 BWR assemblies;
- up to 15 sound metallic fuel rods;
- up to 42 MTR fuel elements;
- up to 42 DIDO fuel assemblies;
- up to 25 high burnup PWR fuel rods (including up to 14 rods classified as damaged)[1];
- up to 25 high burnup BWR fuel rods (including up to 14 rods classified as damaged)[1];
- up to 16 PWR MOX fuel rods (or a combination of 16 PWR MOX and UO_2 PWR rods) and up to 9 BPRs
- up to 9 damaged metallic fuel rods;
- up to 3 severely damaged metallic fuel rods in filters;
- up to 140 TRIGA intact or damaged fuel elements/fuel debris ("TRIGA" is a Trademark of General Atomics);
- up to 560 TRIGA intact or damaged fuel cluster rods/fuel debris;
- 2 GA IFM packages;
- up to 300 TPBARs (of which two can be prefailed) in a consolidation canister;

[1] PWR and BWR fuel rods may be transported in either a fuel assembly lattice (skeleton) or in a fuel rod insert. The fuel rod insert may contain PWR instrument/guide tubes and BWR water/inert rods in addition to the fuel rods.

- up to 25 TPBARs (of which two can be prefailed) in a rod holder;
- up to 55 TPBARs segmented during post-irradiation examination (PIE), including segmentation debris;
- up to 700 PULSTAR fuel elements (intact or damaged);
- up to 42 spiral fuel assemblies;
- up to 42 MOATA plate bundles;
- up to 800 SLOWPOKE undamaged and/or damaged fuel rods contained in up to eight (8) SLOWPOKE fuel canisters (up to 100 fuel rods each); or,
- up to 4,000 lbs of solid, irradiated and contaminated hardware, which may include fissile material less than a Type A quantity and meeting the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c). Total allowed mass includes the weight of spacers, shoring and dunnage.

PWR or BWR fuel rods may be placed in a fuel rod insert (also referred to as a rod holder) or in a fuel assembly lattice. The fuel rod holder is composed of a 4×4 or a 5×5 rod array. An alternate 5×5 rod holder is designed to contain an oversize nonfuel-bearing component (e.g., CE guide tube or BWR water rod). The alternative configuration reduces fuel-bearing capacity to a maximum of 21 fuel rods. The lattice may be irradiated or unirradiated. Up to 14 of the fuel rods may be classified as damaged. Damaged fuel rods must be placed in a rod holder. Damaged fuel rods or rod sections may be encapsulated to facilitate handling prior to placement in the rod holder. PWR rods may include Integral Fuel Burnable Absorber (IFBA) rods.

PWR MOX fuel rods (or a combination of PWR MOX and UO_2 PWR fuel rods) are required to be loaded in a screened or free flow PWR/BWR Rod Transport Canister with a 5×5 insert. PWR MOX/UO_2 rods may include Integral Fuel Burnable Absorber (IFBA) rods.

Damaged TRIGA fuel elements, cluster rods and fuel debris are required to be loaded in a sealed damaged fuel canister (DFC).

PULSTAR fuel elements may be configured as intact fuel assemblies, may be placed into a fuel rod insert, i.e., a 4×4 rod holder (intact elements only), or may be loaded into one of two can designs, designated as the PULSTAR screened fuel can or the PULSTAR failed fuel can. Damaged PULSTAR fuel elements and nonfuel components of PULSTAR fuel assemblies must be loaded into cans. PULSTAR fuel cans may only be loaded into the top or base module of the 28 MTR basket assembly. Intact PULSTAR fuel assemblies and intact PULSTAR fuel elements in a TRIGA fuel rod insert may be loaded in any basket module.

SLOWPOKE undamaged or damaged fuel rods will be loaded into 5 x 5 or 4 x 4 tube arrays with four (4) tube arrays placed in each screened SLOWPOKE Canister. A maximum of up to eight (8) SLOWPOKE Canisters may be loaded in two MTR-28 basket modules (top and top intermediate) with three (3) center row cells fitted with cell block spacers. Damaged fuel rods

that cannot be accommodated in the 5 x 5 tube array will be loaded into the larger diameter 4 x 4 tube array. Empty lower intermediate and bottom MTR-28 basket modules will be installed as axial spacers for the two loaded basket modules.

Irradiated hardware may be loaded directly into the NAC-LWT cavity or preloaded into a canister or cage. Stainless steel dunnage may be used to limit the movement of the irradiated hardware within the cask cavity. The maximum gamma source term of the irradiated hardware shall be limited to that defined for the authorized PWR content condition as described in Chapter 5.

The NAC-LWT cask provides a testable containment for the contents during both normal operations and hypothetical accident conditions, satisfying the requirements of 10 CFR 71.51. Any number of NAC-LWT casks may be shipped at one time, each on its own vehicle.

The NAC-LWT has two leaktight configurations as defined by ANSI N14.5. The standard configuration is provided by a closure lid with a metal containment seal and alternate vent and drain port covers provided with Viton® containment O-rings. The second configuration is provided by a closure lid with a metal containment seal and Alternate B vent and drain port covers provided with metal seals. The metal port cover seal containment configuration is required to be utilized for all TPBAR contents and may be used for other contents. The NAC-LWT standard, Viton® O-ring containment configuration is not authorized for TPBAR contents.

NAC-LWT casks may be shipped in a closed International Shipping Organization (ISO) container when containing all fuel contents other than PWR and BWR fuel assemblies. NAC-LWT casks containing PWR and BWR fuel assemblies are to be transported on an open trailer with a personnel barrier.

The terminology of MTR, DIDO and TRIGA fuel elements will be used independent of whether the element contains low, medium or high enriched uranium (i.e., LEU, MEU or HEU), except when required for analysis or loading purposes.

TPBAR contents may be placed into a consolidation canister, waste container or 5×5 rod insert (also referred to as a rod holder) in a PWR/BWR Rod Transport Canister. Segmented TPBARs are only permitted within the waste container. The three TPBAR shipping configurations are individually placed within one of the two TPBAR basket assemblies (one with a 7-inch bottom spacer and the alternative TPBAR basket assembly with a 6.5-inch alternative bottom spacer), depending on container configuration.

required to be configured with Alternate B drain and vent port covers incorporating metallic seals. For all other contents, the leaktight capable (i.e., no credible leakage) alternate port covers incorporating Viton O-ring seals can be used. The transport arrangement drawings for approved contents are presented in Section 1.4.

An alternative drain tube, including a drain tube alignment ring, is required to be installed and utilized when loading and transporting modular fuel baskets (i.e., not full length) and canisters.

The impact limiters and the personnel barrier are designed to be removed and installed without the aid of supplemental lifting gear or fixtures. All approved content may be transported in an International Shipping Organization (ISO) container, except for PWR and BWR fuel assemblies. All operational features are readily apparent from the drawings provided in Section 1.4. Operational procedures are delineated in Chapter 7.

1.2.3 Contents of Packaging

The NAC-LWT cask is analyzed, as presented in this SAR, for the transport of the contents listed in Table 1.1-1 and Section 1.1.

Shipments in the NAC-LWT package shall not exceed the following limits:

1. The maximum contents weight shall not exceed 4,000 pounds.
2. The limits specified in Table 1.2-1 through Table 1.2-13 for the fuel and other radioactive contents shall not be exceeded.
3. Any number of casks may be shipped at one time, one cask per tractor/trailer vehicle.
4. The maximum decay heat shall not exceed the following: 2.5 kW for PWR fuel assemblies; 2.2 kW for BWR fuel assemblies; 2.3 kW for 25 high burnup PWR fuel rods; 2.1 kW for 25 high burnup BWR fuel rods; 2.3 kW for 16 PWR MOX/UO_2 fuel rods; 1.26 kW for MTR fuel; 1.05 kW for DIDO fuel assemblies with top spacer and 0.756 kW without top spacer; 1.05 kW for TRIGA fuel elements or fuel cluster rods; 13.05 W for GA IFM packages; 0.693 kW for 300 TPBARs; 0.127 kW for TPBAR segments; 0.058 kW for 25 TPBARs; 0.84 kW for the PULSTAR fuel contents; 0.659 kW for spiral fuel assemblies (0.109 kW per basket); 0.126 kW for MOATA plate bundles (21 W per basket); 5.0 W for SLOWPOKE fuel rods; and 1.26 kW for solid, nonfissile, irradiated hardware.
5. Radiation levels shall meet the requirements delineated in 10 CFR 71.47 or 49 CFR 173.441. The neutron shield tank may be drained for shipment of metallic fuel rods.
6. Surface contamination levels shall meet the requirements of 10 CFR 71.87(i) or 49 CFR 173.443.
7. Damaged TRIGA fuel elements and fuel debris (up to two equivalent elements) will be shipped in a sealed damaged fuel canister.
8. Damaged TRIGA cluster rod and fuel debris will be transported in a sealed damaged fuel canister (maximum of up to six equivalent fuel cluster rods).

9. MTR fuel elements may consist of any combination of intact or damaged highly enriched uranium (HEU), medium enriched uranium (MEU) or low enriched uranium (LEU) fuel elements that are enveloped by the parameters listed in Table 1.2-4 as supported by information presented in Table 5.1.1-2, Table 6.4.3-21, Table 6.4.3-22, Table 6.4.3-25 and Table 6.4.3-28. MTR fuel elements will be transported in a leaktight configuration NAC-LWT cask.
10. High burnup PWR fuel rods will be shipped in either a sealed, free flow or screened can.
11. High burnup BWR fuel rods will be shipped in either a sealed, free flow or screened can.
12. Up to 25 high burnup PWR or BWR fuel rods in a fuel assembly lattice or rod holder. Up to 14 of the fuel rods in a rod holder may be classified as damaged. Damaged fuel rods or rod sections may be placed into fuel rod capsules prior to placing them in the fuel rod holder. Typical failed fuel rod capsule configuration is shown in Figure 1.2-11.
13. Production TPBARs will either be shipped in an open top consolidation canister as shown in Figure 1.2.3-10 and assembled in the cask as shown in Figure 1.2.3-12, or shipped in a PWR/BWR Rod Transport Canister in accordance with License Drawing No. 315-40-104.
14. Intact PULSTAR fuel elements may be loaded into a fuel rod insert or the PULSTAR screened or failed fuel can.
15. Damaged PULSTAR fuel elements and nonfuel components of PULSTAR fuel assemblies shall be loaded into either a PULSTAR failed fuel or screened fuel can, and placed into the top or base module of the 28 MTR fuel basket. Damaged fuel, including fuel debris, may be placed in an encapsulating rod prior to loading in a PULSTAR can.
16. Any combination of spiral fuel assemblies or MOATA plate bundles, each loaded into separate ANSTO basket modules containing up to a total of 42 assemblies/bundles.
17. Segmented TPBARs will be shipped in a sealed, dry Waste Container as shown in Figure 1.2.3-16 and assembled in the cask as shown in Figure 1.2.3-17.
18. Solid, irradiated and contaminated hardware containing less than a Type A quantity of fissile material and meeting the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c), loaded directly into the cask or contained in a secondary container or basket. The irradiated hardware spacer will be installed to limit the axial movement of the hardware above the lead shielded region of the cask body. As needed, additional secondary containers, dunnage and shoring may be used to limit the movement of the contents during normal and accident conditions of transport.
19. PWR MOX fuel rods (or a combination of PWR MOX and UO_2 PWR fuel rods) are required to be loaded in a screened or free flow PWR/BWR Rod Transport Canister provided with a 5 × 5 insert.
20. Any combination of up to 7 degraded clad DIDO, spiral or MOATA plate elements/ bundles loaded into an aluminum screened DFC as shown Figure 1.2.3-18 placed in an ANSTO top basket module, with remainder of either ANSTO basket modules containing MOATA plate bundles or spiral fuel elements or ANSTO-DIDO combination basket containing DIDO elements. Degraded aluminum-clad DIDO, spiral or MOATA plate elements/bundles will be transported in a leaktight configuration NAC-LWT cask.

21. Any combination of undamaged or damaged SLOWPOKE Fuel Rods contained in 5 x 5 or 4 x 4 rod insert assemblies loaded into screened aluminum Fuel Canisters (four rod insert assemblies per fuel canister) with a maximum of four (4) SLOWPOKE Canisters per MTR-28 top and upper intermediate basket module (maximum of eight canisters per NAC-LWT cask). Cell block spacers will be installed in the center three fuel cells for the loaded basket modules.

1.2.3.1 TRIGA Fuel and Basket Description

Two basic types of TRIGA fuel are to be transported in the NAC-LWT cask: TRIGA fuel elements and smaller fuel rods from TRIGA fuel cluster assemblies. TRIGA fuel elements are approximately 1-1/2 inches in diameter and are described in Section 1.2.3.1.1. TRIGA fuel cluster rods are smaller; approximately 1/2-inch in diameter and are also described in Section 1.2.3.1.1.

Up to 140 TRIGA fuel elements in the form of: a) standard fuel elements – either aluminum clad or stainless steel clad; b) instrumented fuel elements – similar to standard fuel elements (aluminum clad or stainless steel clad), but containing thermocouple instrumentation; and c) fuel follower control rod elements (aluminum or stainless steel clad) – poison rods with a fuel follower in a single tube may be shipped in the NAC-LWT cask. Up to 560 TRIGA fuel cluster rods may be shipped.

Up to six equivalent TRIGA fuel cluster rods may be loaded and transported in a sealed damaged fuel can (DFC). Up to the equivalent of two TRIGA damaged fuel elements and debris may be loaded and shipped in a sealed DFC. The TRIGA transport baskets and DFCs are described in Section 1.2.3.1.2.

1.2.3.1.1 TRIGA Fuel

TRIGA Fuel Elements

The characteristics of the design basis TRIGA fuel element are presented in Table 1.2-4 and in Table 1.2-1 for the poisoned basket and in Table 1.2-2 for the nonpoisoned basket.

The fuel material in a TRIGA fuel element is a solid, homogeneous mixture of uranium-zirconium hydride alloy, i.e., a metal alloy fuel. Both the aluminum-clad and the stainless steel-clad TRIGA fuel elements are approximately 1.5-inch diameter rods by approximately 30 inches long. The fuel follower control rod elements range in length from 45 inches to 66.5 inches and are cut, as required, to fit the basket length. Instrumented fuel elements are identical to standard fuel elements with the exception of thermocouples and wires and lead-out tubing. The lead-out tubing needs to be detached prior to shipment in order for the instrumented fuel elements to fit into the standard element height envelope. The aluminum-clad TRIGA fuel element and instrumented fuel element, the stainless steel-clad TRIGA fuel element and instrumented fuel element, and the standard fuel follower control rod element are shown in Figure 1.2.3-1 through Figure 1.2.3-5, respectively.

TRIGA Fuel Cluster Rods

The fuel material in TRIGA fuel cluster rods is a solid, homogeneous mixture of uranium-zirconium-erbium hydride alloy, i.e., a metal alloy fuel. Erbium is a burnable neutron poison that is used in the fuel to enhance the flux profile along the length of the fuel rod, and conservatively ignored in the nuclear evaluations. The rods have a nominal diameter of 0.54 inch and are approximately 31 inches long. The rod cladding is Incoloy 800 material and is 0.015-inch thick, minimum. Instrumented rods are identical to the standard rods, with the exception of thermocouples and wires. A diagram of the TRIGA fuel cluster rods, and the individual fuel pin (cluster rod) making up the cluster, is shown in Figure 1.2.3-6.

The active fuel region of a TRIGA fuel cluster rod is a maximum of 0.53 inch in diameter, 22.5 inches in length, and has an initial uranium enrichment of up to 95 percent for HEU material and 20percent for LEU material. A compression spring is utilized to fill the space in the plenum region of the rod, and top and bottom plugs are used to seal the fuel within the rod. The design-basis TRIGA fuel cluster rod characteristics are summarized in Table 1.2-3, Table 1.2-4, and Tables 5.1.1-1, 5.1.1-2, 6.2.6-1 and 6.2.6-2.

Axial fuel spacers, as shown on Drawing 315-40-085, may be used to axially position the TRIGA fuel elements, fuel inserts and DFCs. The axial spacers do not provide a safety function and are dunnage used to position the fuel elements to facilitate fuel handling. The total weight per basket module cell for the TRIGA fuel elements or cluster rods, inserts, spacer(s) and fuel cans, as applicable, shall be limited to a maximum of 80 pounds.

TRIGA Fuel Classification

The TRIGA fuel contents are divided into three categories based on fuel condition for evaluation, loading configuration and transport in the NAC-LWT:

1. Intact fuel (i.e., no cladding breach) is loaded directly into the TRIGA fuel basket modules (Section 1.2.3.1.2) with a maximum of four TRIGA fuel elements per loading position. Certain high ^{235}U content, LEU and HEU intact stainless steel TRIGA fuel elements, as defined in Table 1.2-2, are restricted to a maximum loading of three fuel elements per basket module cell in a top or bottom basket module only. To ensure that four fuel elements are not loaded into a cell containing high ^{235}U content fuel elements, a dummy TRIGA spacer tube is preinstalled in the basket prior to loading. Up to 16 intact cluster rods are loaded into fuel rod inserts (Drawing 315-40-096) that are inserted into the TRIGA fuel basket module cell openings. Intact TRIGA fuel elements and cluster rods may be loaded into a sealed DFC, if length permits.
2. Damaged TRIGA fuel elements and TRIGA fuel debris (up to the equivalent of two fuel elements) shall be loaded into a sealed DFC (Section 1.2.3.1.2), and then loaded into a top or base basket module.

plates. Two thick (0.635 cm) aluminum nonfuel side plates support the fuel plate stack from two sides, making a possible total of 16 plates per bundle. At each axial end, the plates in the stack are connected by a pin. Spacing between plates is maintained by disk spacers placed onto the top and bottom pins between each fuel plate and the aluminum side plates. A sketch of a typical MOATA plate bundle is provided in Figure 1.2.3-15.

1.2.3.10 Solid, Irradiated and Contaminated Hardware

The design basis characteristics of the solid, irradiated and contaminated hardware are provided in Table 1.2.3-13. As described in the content definition, the solid, irradiated and contaminated hardware may contain small quantities of fissile materials. Fissile materials in the irradiated hardware contents are acceptable if the quantity of fissile material does not exceed a Type A quantity and does not exceed the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c).

The irradiated hardware may be directly loaded into the NAC-LWT cask cavity, or may be contained in a secondary container or basket. As needed, appropriate component spacers, dunnage and shoring may be used to limit the movement of the contents during normal and accident conditions of transport.

To ensure that the movement of the irradiated hardware contents above the lead shielded length of the NAC-LWT cask body (i.e., the approximately upper 6.25 inches of the cavity length) is precluded, an Irradiated Hardware Lid Spacer as shown on Drawing No. 315-40-145 shall be installed for all irradiated hardware content configurations. The total installed height of the spacer is 6.5 inches. Therefore, the available cavity length for the irradiated hardware is approximately 171 inches. The NAC-LWT cask shall be assembled for transport as shown on NAC Drawing No. 315-40-01 with the irradiated hardware spacer installed on the lid.

A comparative shielding evaluation for a conservatively selected irradiated hardware transport configuration (i.e., a single line source with no self-shielding) or consideration of the additional shielding provided by additional spacers, dunnage, inserts or secondary containers is presented in Chapter 5. The evaluations show that the regulatory dose rate requirements per 10 CFR 71.47 for normal conditions of transport, or 10 CFR 71.51(b) under hypothetical accident conditions, are not exceeded.

1.2.3.11 PWR MOX Fuel Rods

The NAC-LWT cask is analyzed and evaluated for the transport of up to 16 PWR MOX fuel rods (or a combination of up to 16 PWR MOX and UO_2 fuel rods) loaded into a 5 × 5 insert placed in a screened or free flow PWR/BWR Rod Transport Canister. The authorized characteristics of

the evaluated PWR MOX fuel rods are provided in Table 1.2-4. For mixed PWR MOX and UO_2 PWR fuel rod combinations, the UO_2 PWR fuel rods may have the identical heat load, burnup and cool time characteristics as the PWR MOX fuel rods.

In addition to the 16 PWR MOX fuel rods (or a combination of PWR MOX and UO_2 PWR fuel rods), up to 9 burnable poison rods (BPRs) may be loaded in the remaining openings in the 5 × 5 insert in the PWR/BWR Rod Transport Canister.

1.2.3.12 SLOWPOKE Fuel Rods

SLOWPOKE fuel rods are transported in the NAC-LWT in the 28 MTR fuel basket assembly, which consists of four modules with seven cells per module. The basket assembly is composed of a top module, a base module, and two intermediate modules. Fuel load is limited to a maximum of four loaded cells per basket module, with fuel only loaded in the top and top intermediate modules. The lower intermediate and bottom modules are used as axial spacers and are not loaded. The center row of three cells within the basket modules containing fuel is not loaded and contains a blocking device in each opening to prevent inadvertent loading. Therefore, a cask load for SLOWPOKE fuel rods is limited to eight loaded cells per cask.

SLOWPOKE fuel rods must be loaded into a canister. The canister is a screened boundary providing gross particle control for damaged fuel material. Damaged and undamaged fuel may be mixed when loaded into the canister. Canister content is composed of 4x4 or 5x5 aluminum tube arrays that are stacked four high within the canister. Mixed load of 4x4 and 5x5 tube arrays are permitted in a canister. Based on the 5x5 tube arrays and four tube arrays per canister, the maximum content per canister is 100 SLOWPOKE fuel rods (or the equivalent quantity of damaged material).

SLOWPOKE fuel rods are composed of highly enriched (> 90 wt %) uranium-aluminum alloy fuel meat within aluminum cladding. During reactor operation ~300 rods form a reactor core. Criticality in a SLOWPOKE core is achieved by the use of a thick beryllium neutron reflector surrounding the core. A sketch of a SLOWPOKE fuel rod is provided in Figure 1.2.3-19. Key physical, radiation protection and thermal characteristics of the SLOWPOKE fuel rods are listed in Table 1.2.3-14.

The SLOWPOKE canister is constructed primarily of aluminum. A limited quantity of stainless steel is located within the canister lid structure. The canister is designed to: a) minimize the dispersal of gross fuel particles that may escape from damaged fuel rod cladding and/or fuel debris (note that metallic fuel is not expected to release significant gross particulate even with severe clad damaged); b) facilitate retrieval of the contents from the transportation cask; and, c) confine damaged fuel and/or debris within a known volume to facilitate criticality control,

maintain dose limits, and control thermal loads within the cask. SLOWPOKE fuel pellets, pieces, and debris may be placed into an aluminum tube structure located within the canister. The aluminum tubes provide structural support for individual fuel rods/pieces during transport in the NAC-LWT but are not required within the analysis to maintain safety limits.

Figure 1.2.3-1 Aluminum Clad TRIGA Fuel Element

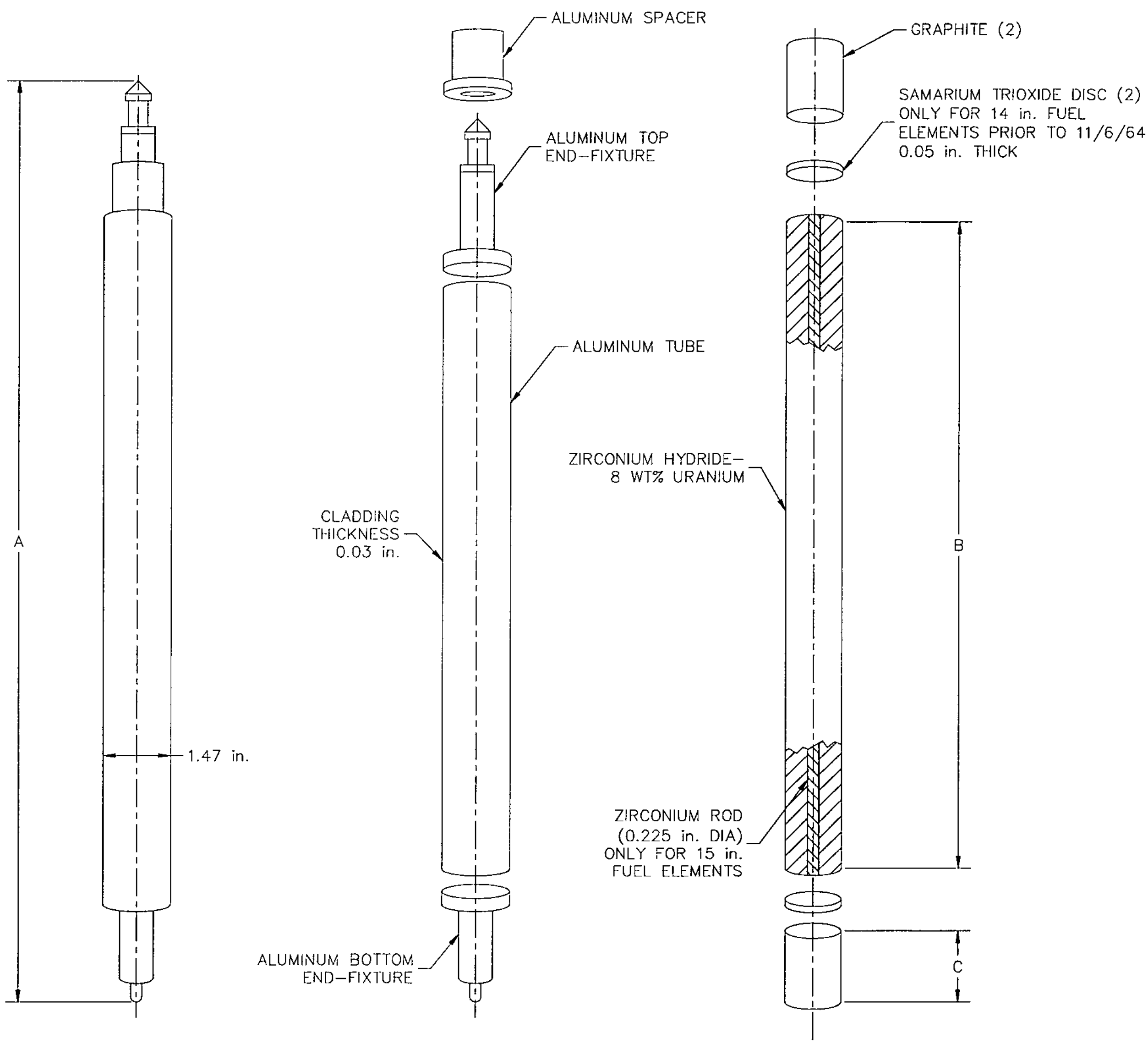


Figure 1.2.3-2 Aluminum Clad Instrumented Fuel Element

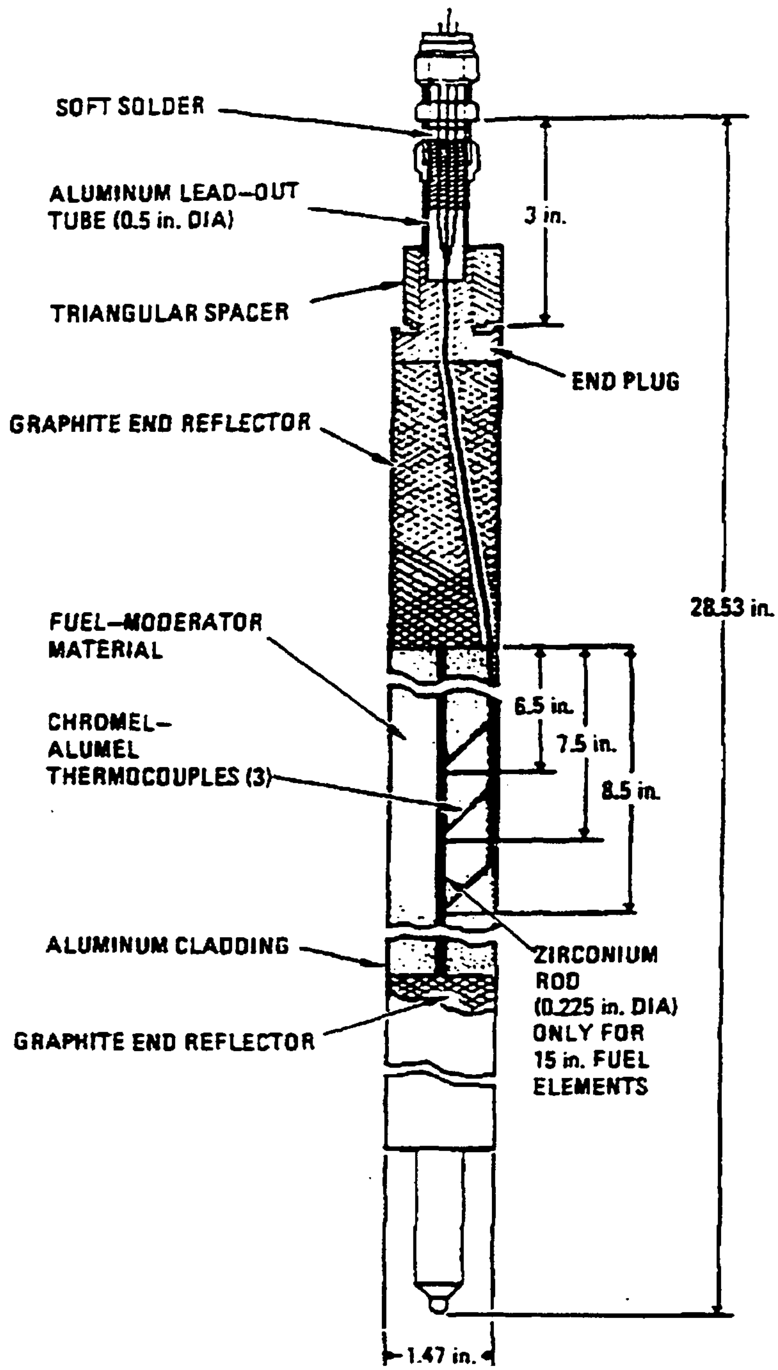


Figure 1.2.3-3 Stainless Steel Clad TRIGA Fuel Element

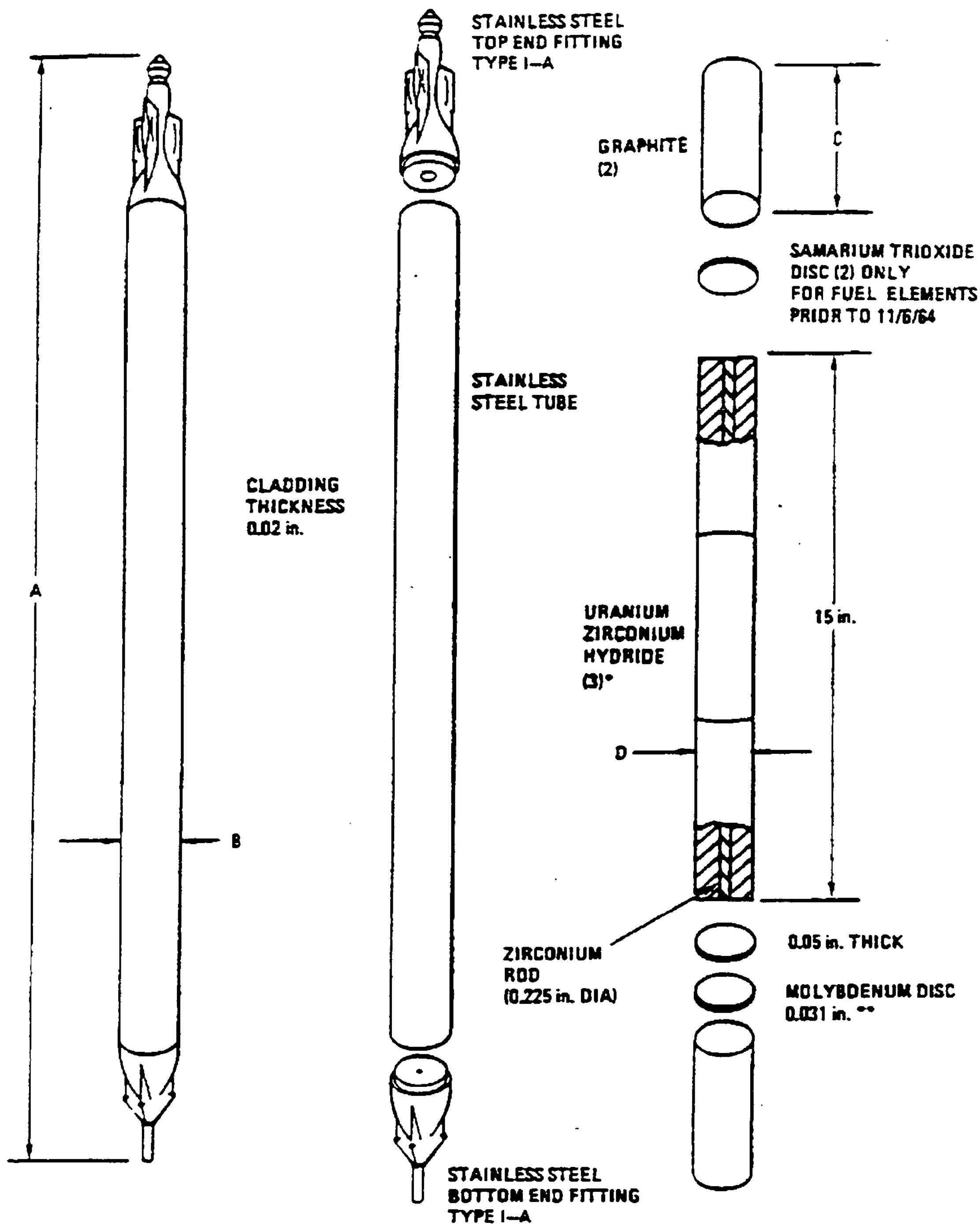


Figure 1.2.3-4 Stainless Steel Clad Instrumented Fuel Element

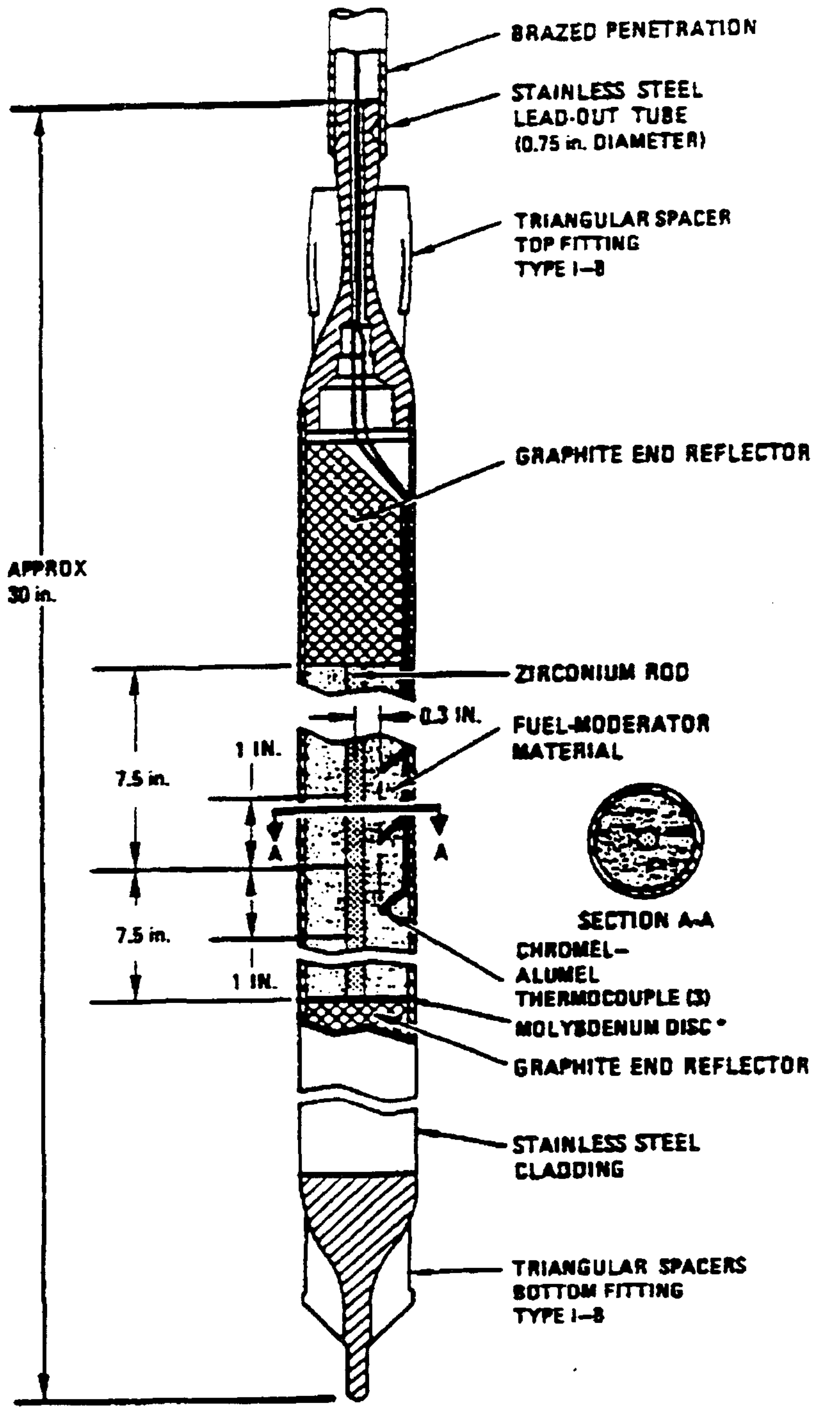


Figure 1.2.3-5 Standard Fuel Follower Control Rod Element

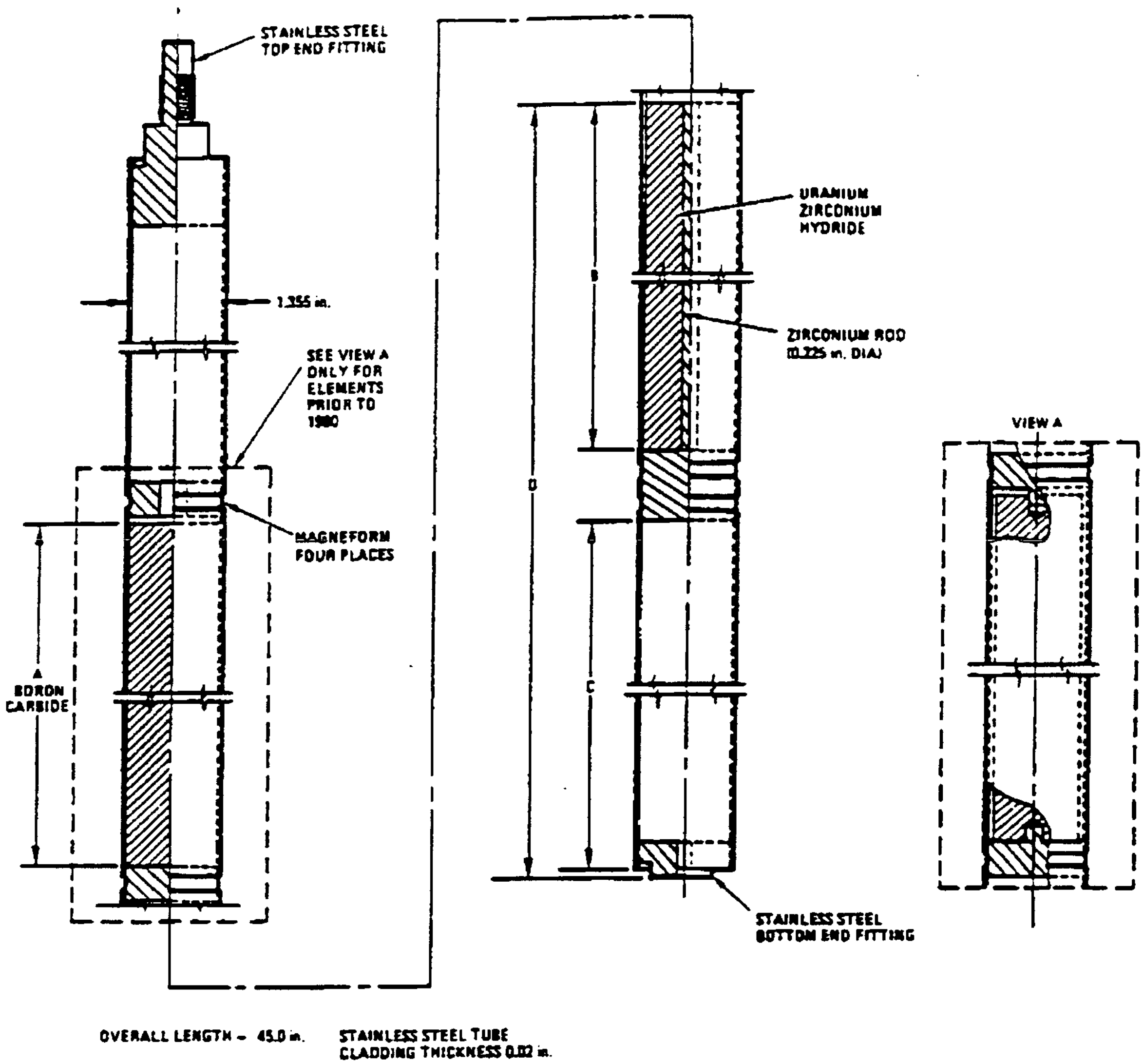
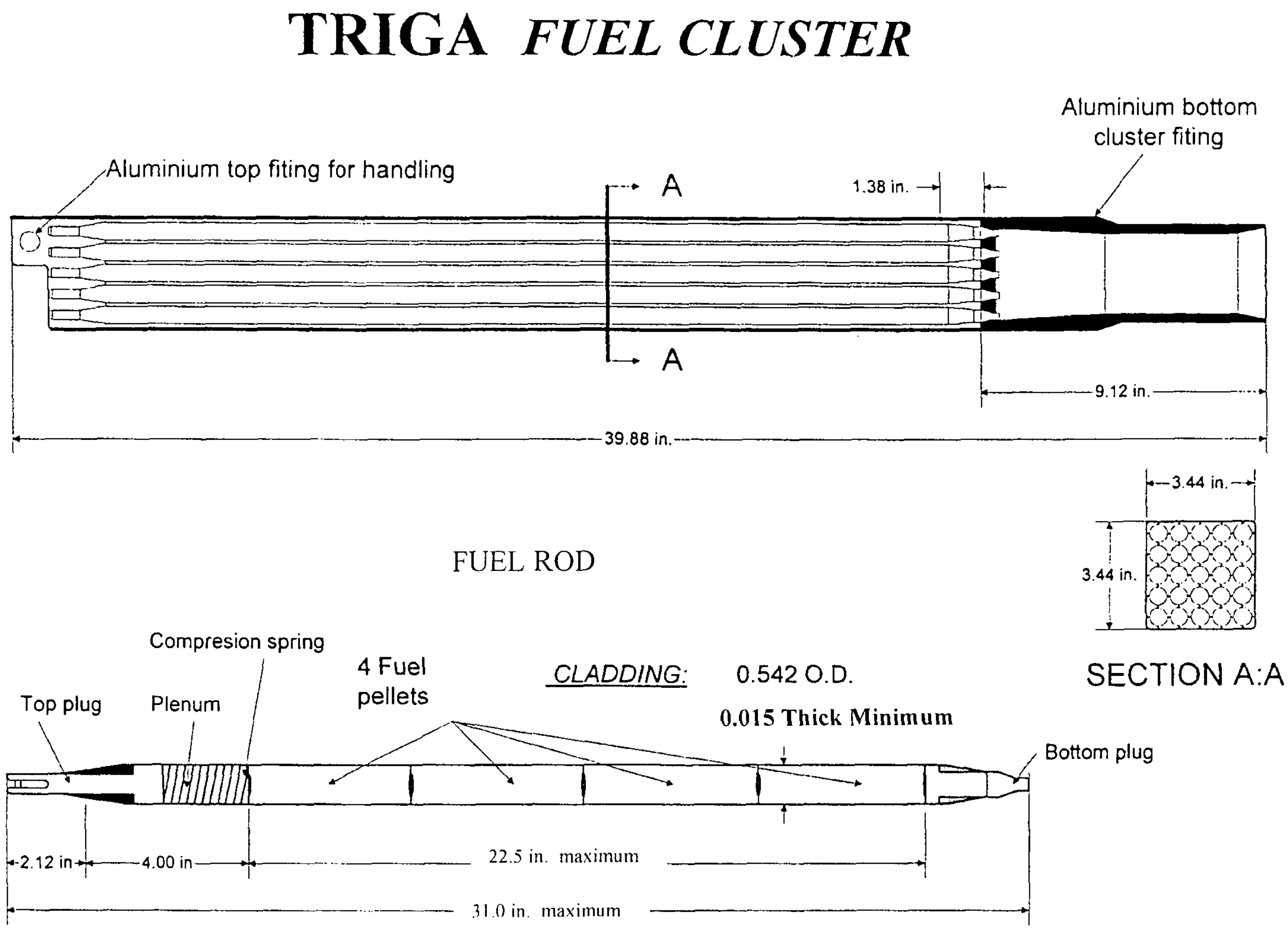


Figure 1.2.3-6 TRIGA Fuel Cluster and Rod Details



Fuel bundle and fuel element

Figure 1.2.3-7 HTGR Fuel Handling Unit

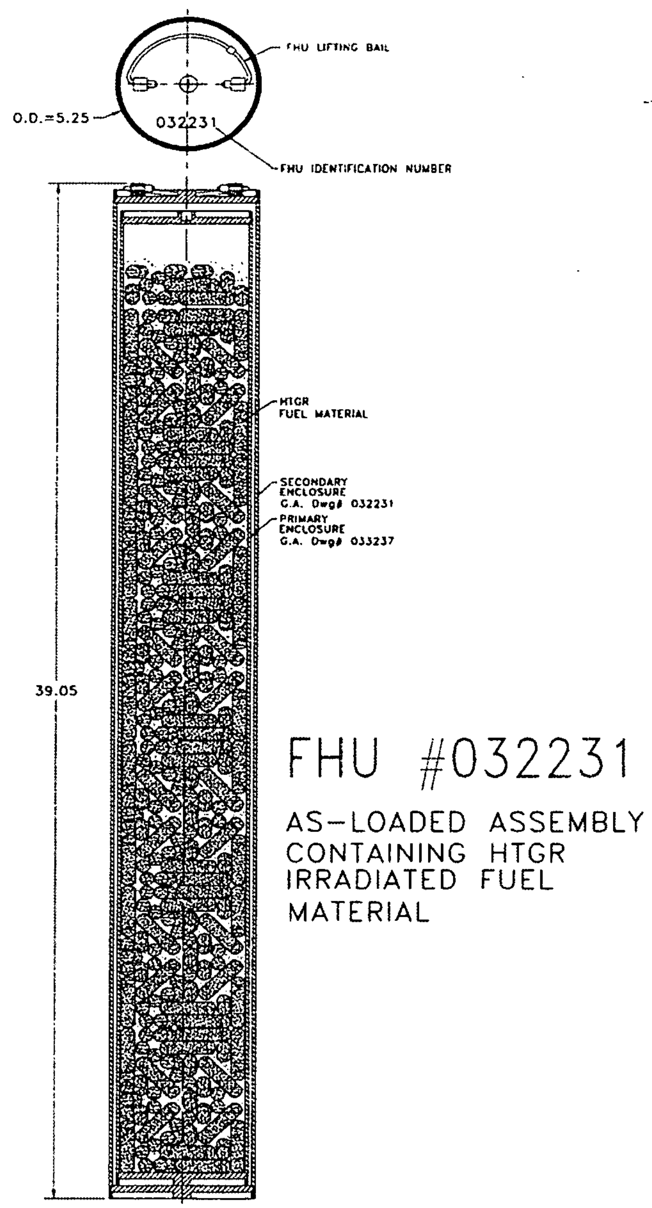


Figure 1.2.3-8 RERTR Fuel Handling Unit

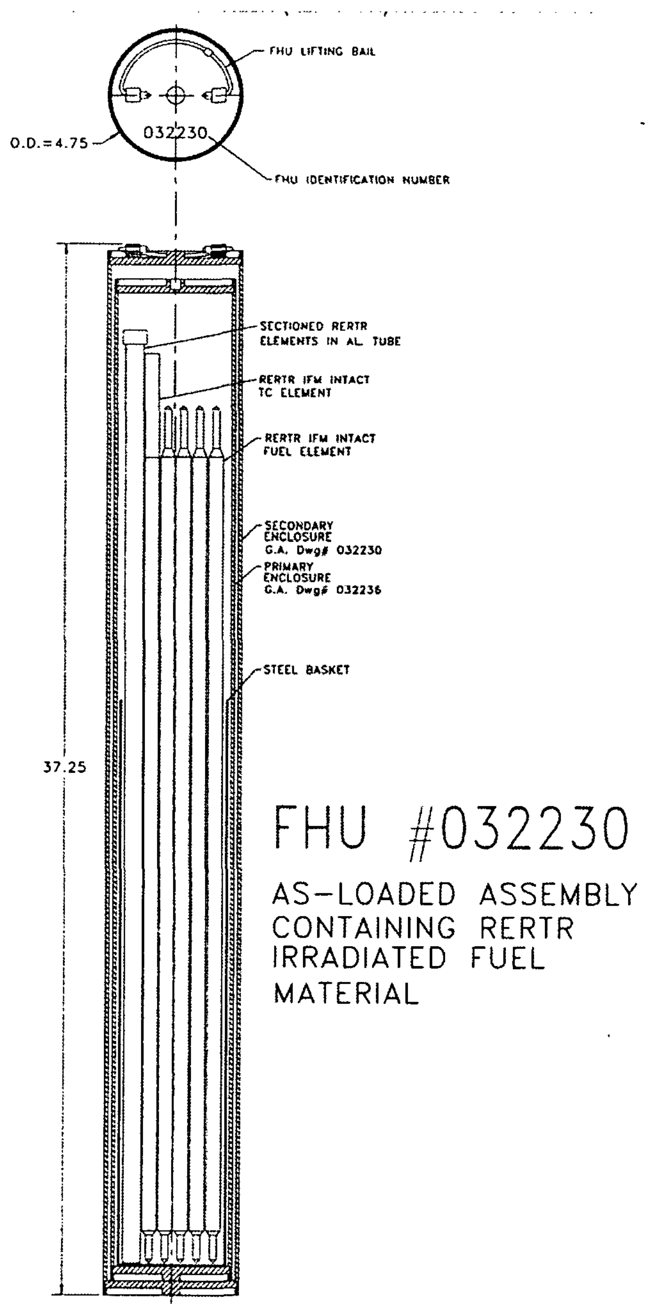


Figure 1.2.3-9 Typical TPBAR Assembly

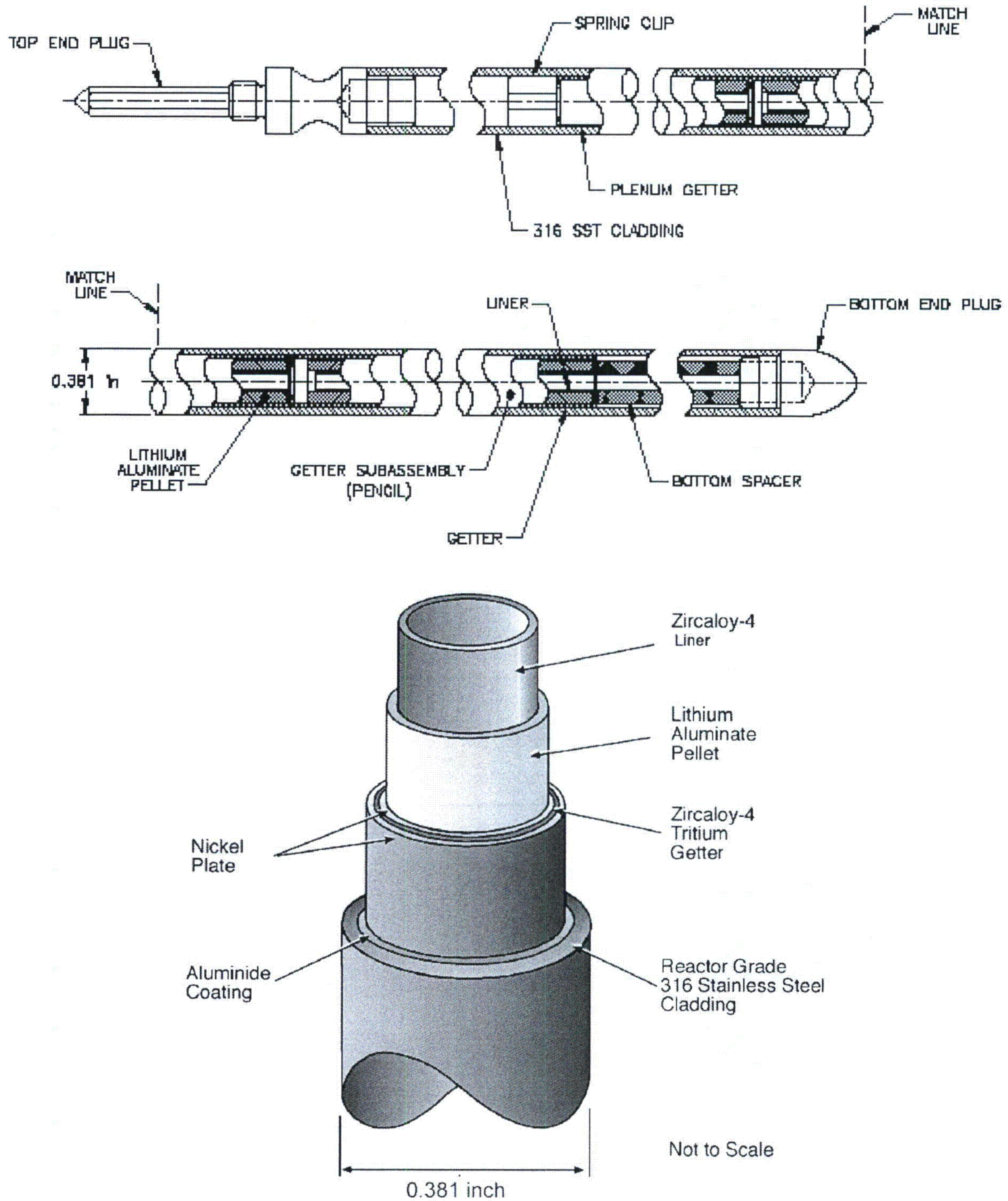
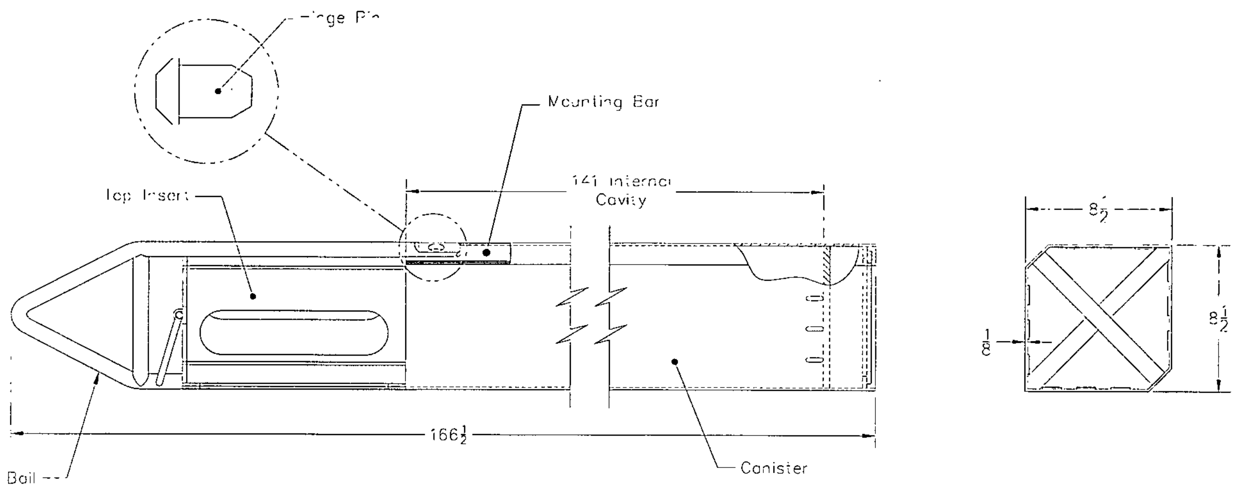


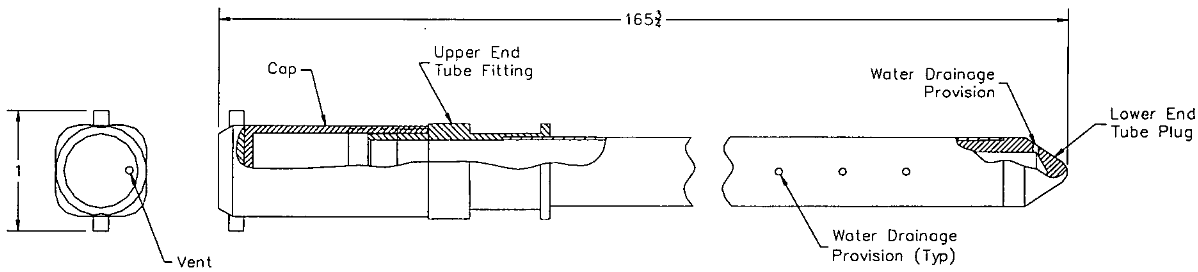
Figure 1.2.3-10 TPBAR Consolidation Canister Sketch



Conceptual Layout with Approximate Dimensions

Note: Material of construction is stainless steel.

Figure 1.2.3-11 Failed PWR/BWR Fuel Rod Capsule



Failed Fuel Rod Capsule Conceptual Layout

All Dimensions Approximate

Note: Material of construction is stainless steel.

Figure 1.2.3-12 NAC-LWT with TPBAR Consolidation Canister Payload

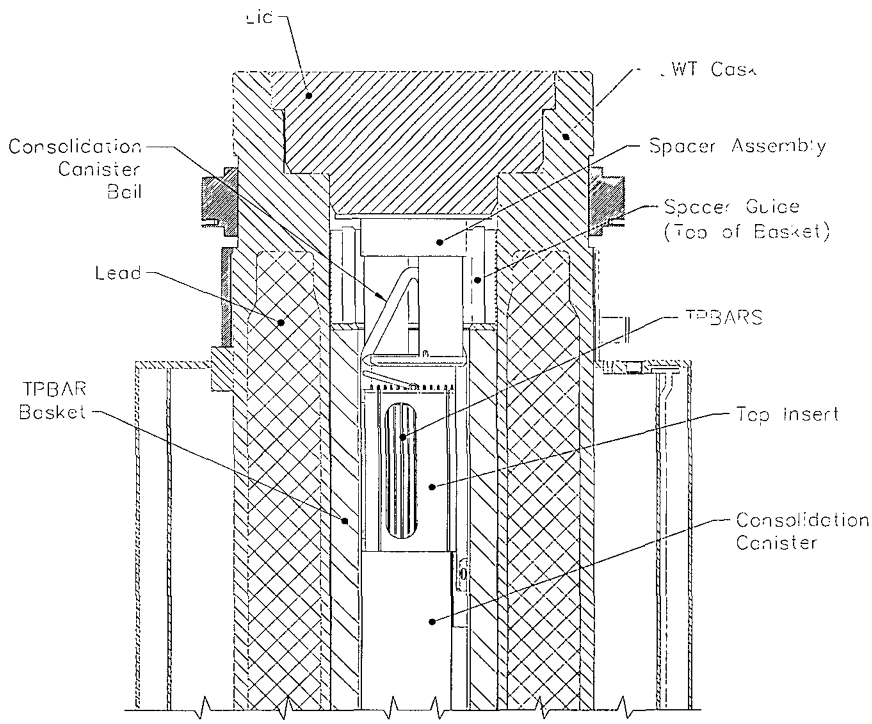


Figure 1.2.3-13 PULSTAR Fuel Assembly

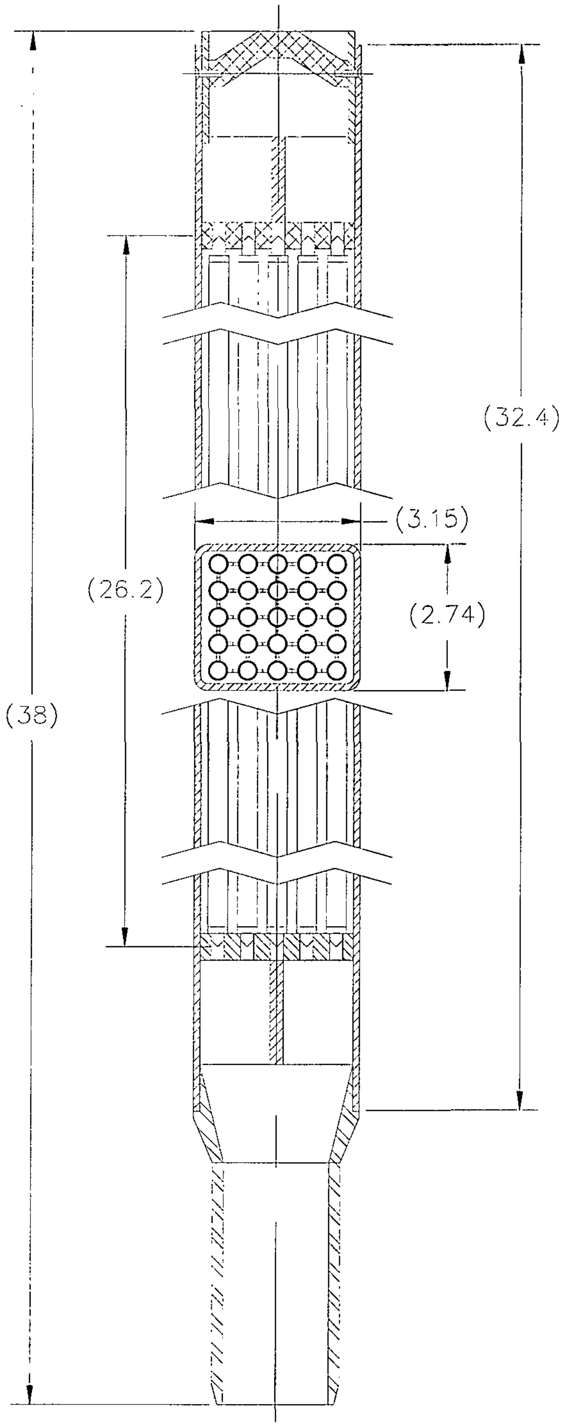
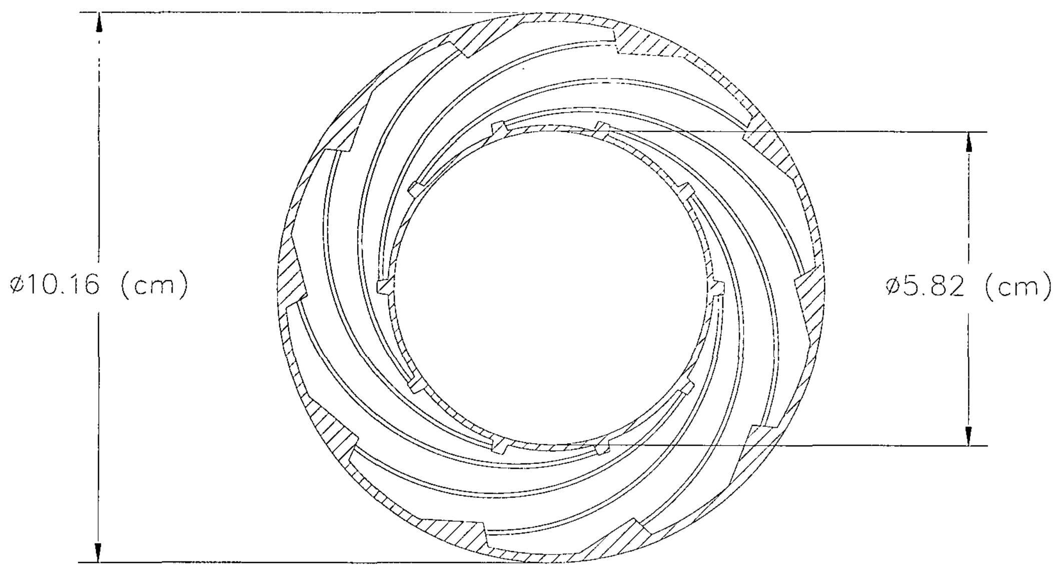
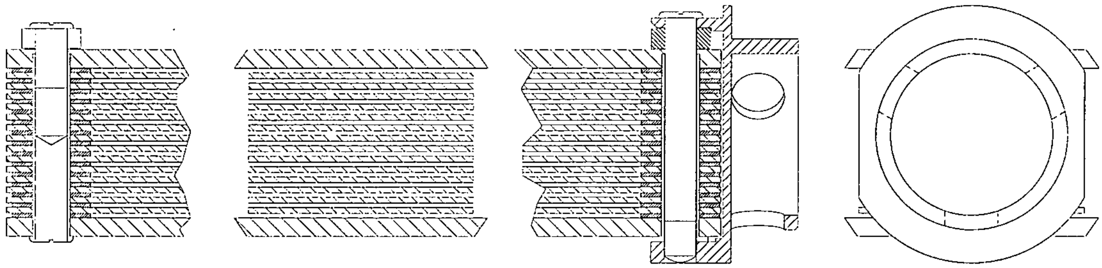


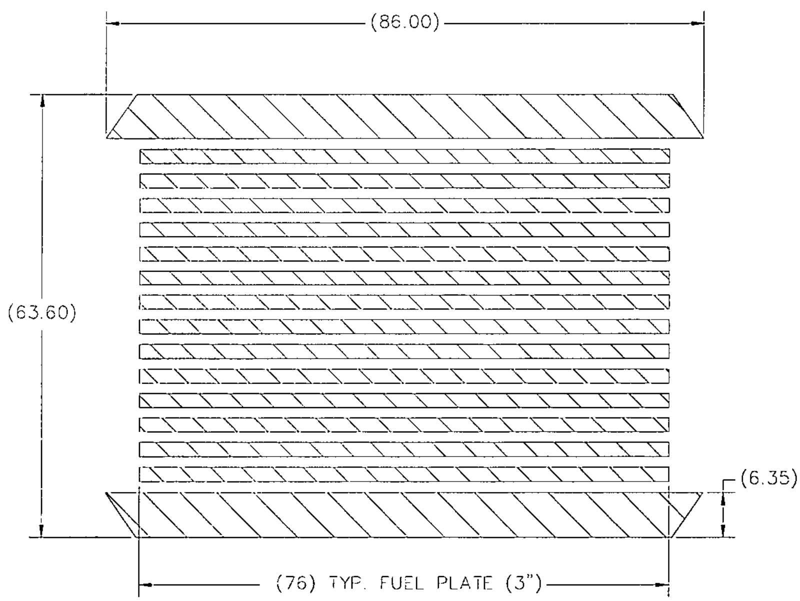
Figure 1.2.3-14 Spiral Fuel Assembly Cross-Section Sketch



Note: Nominal dimensions

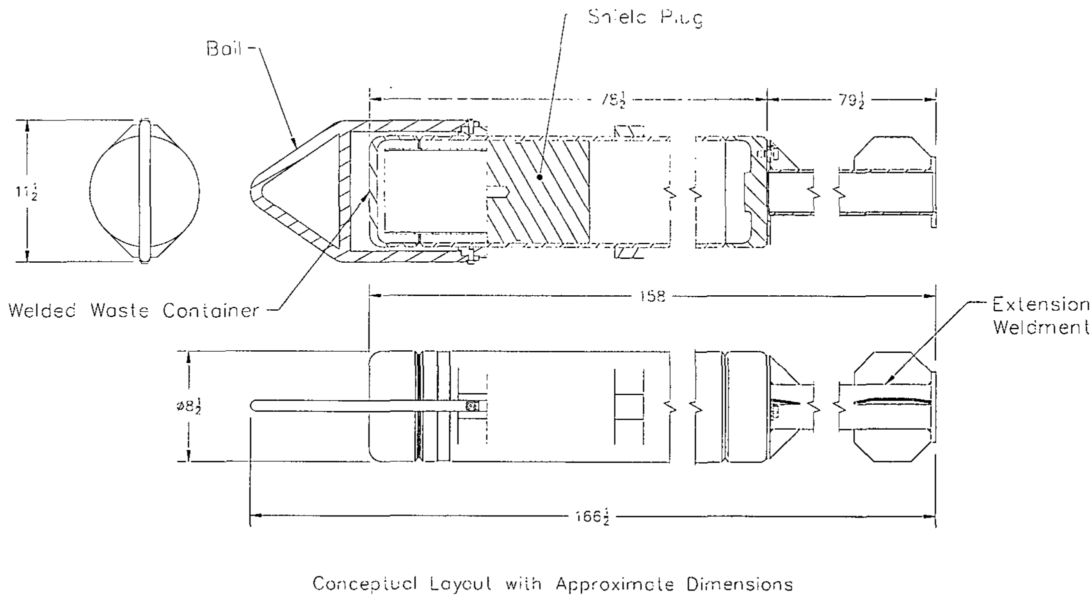
Figure 1.2.3-15 MOATA Plate Bundle Sketches





Note: 14-plate bundle configuration. Dimensions are reference values. Bundles with a reduced number of plates retain the plate pitch and compensate by wider side plates and outside spacers to retain overall bundle dimensions.

Figure 1.2.3-16 TPBAR Waste Container and Extension Weldment Sketch



Note: Material of construction is stainless steel.

Figure 1.2.3-17 NAC-LWT with TPBAR Waste Container Payload

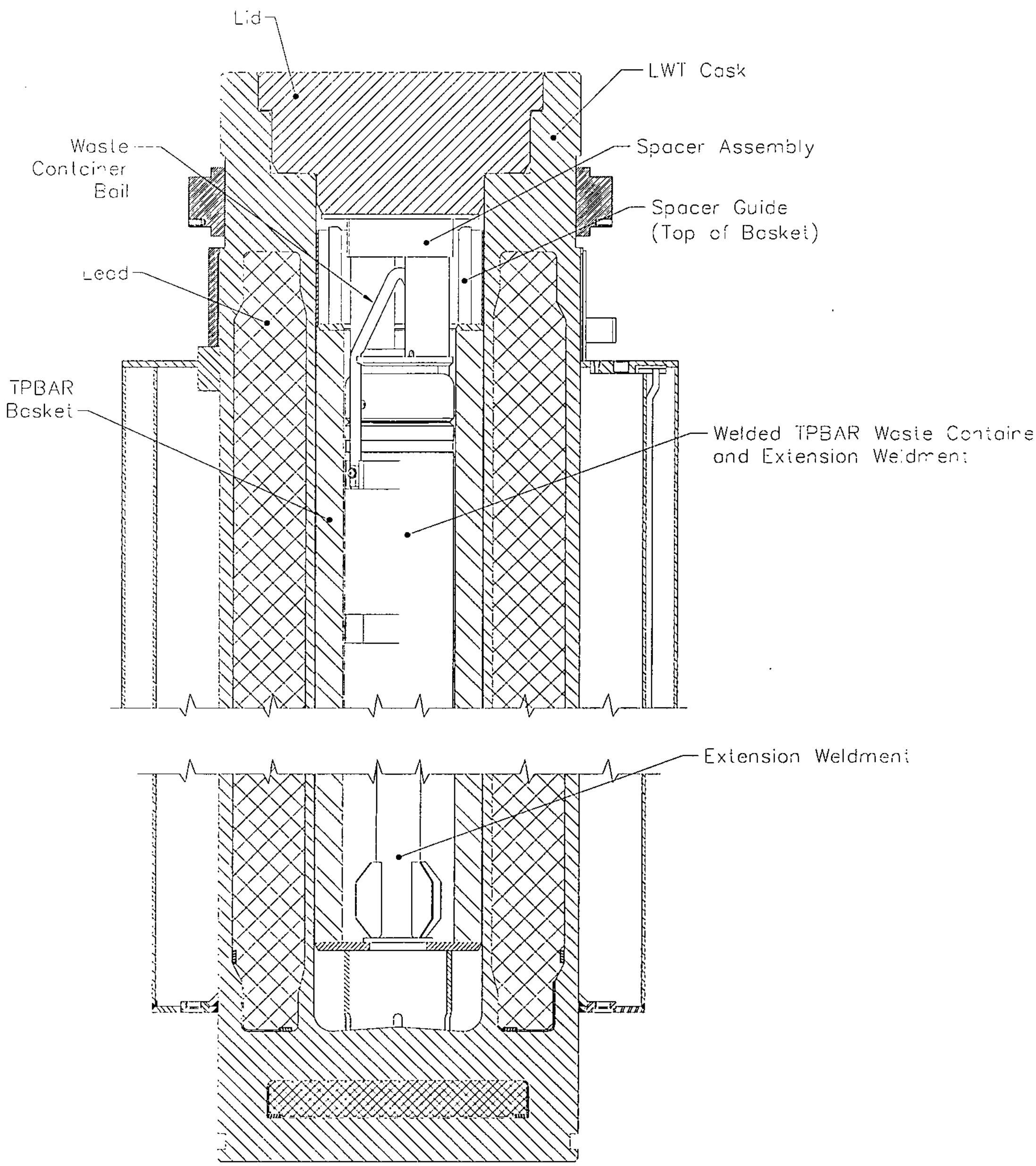
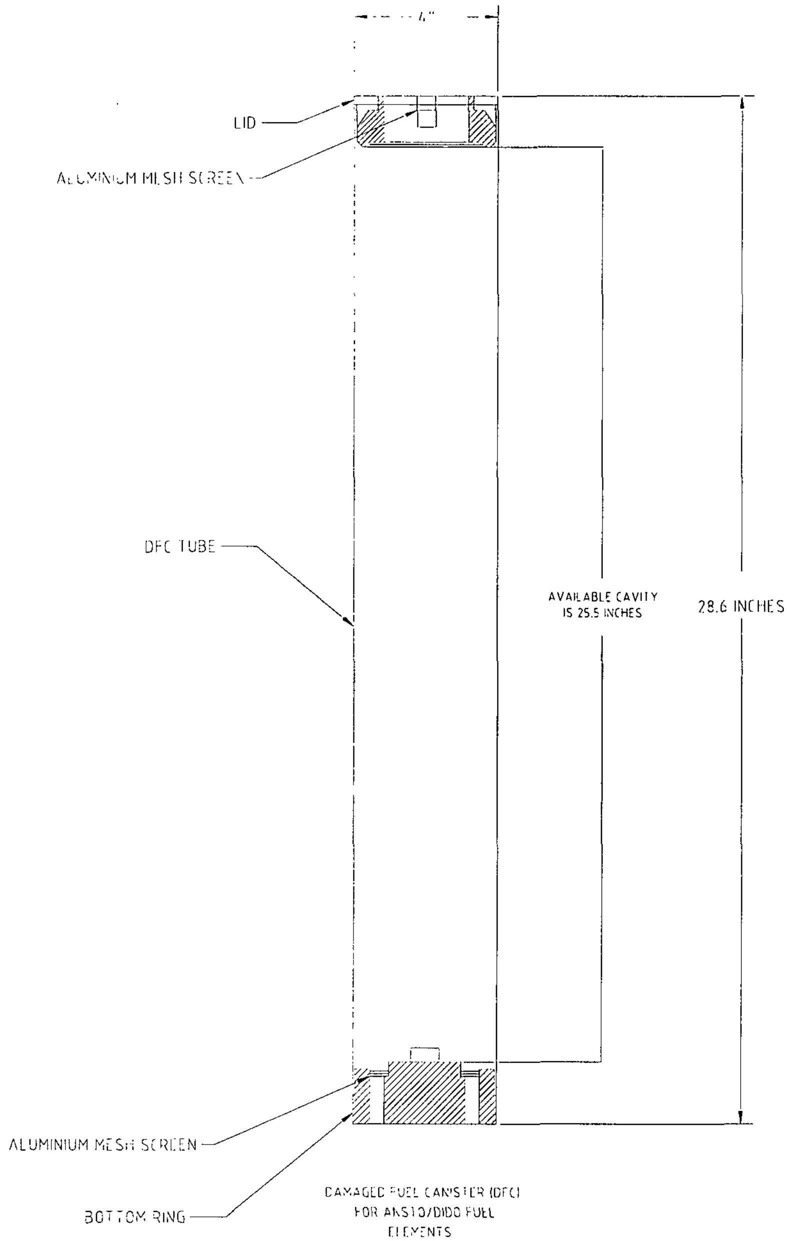
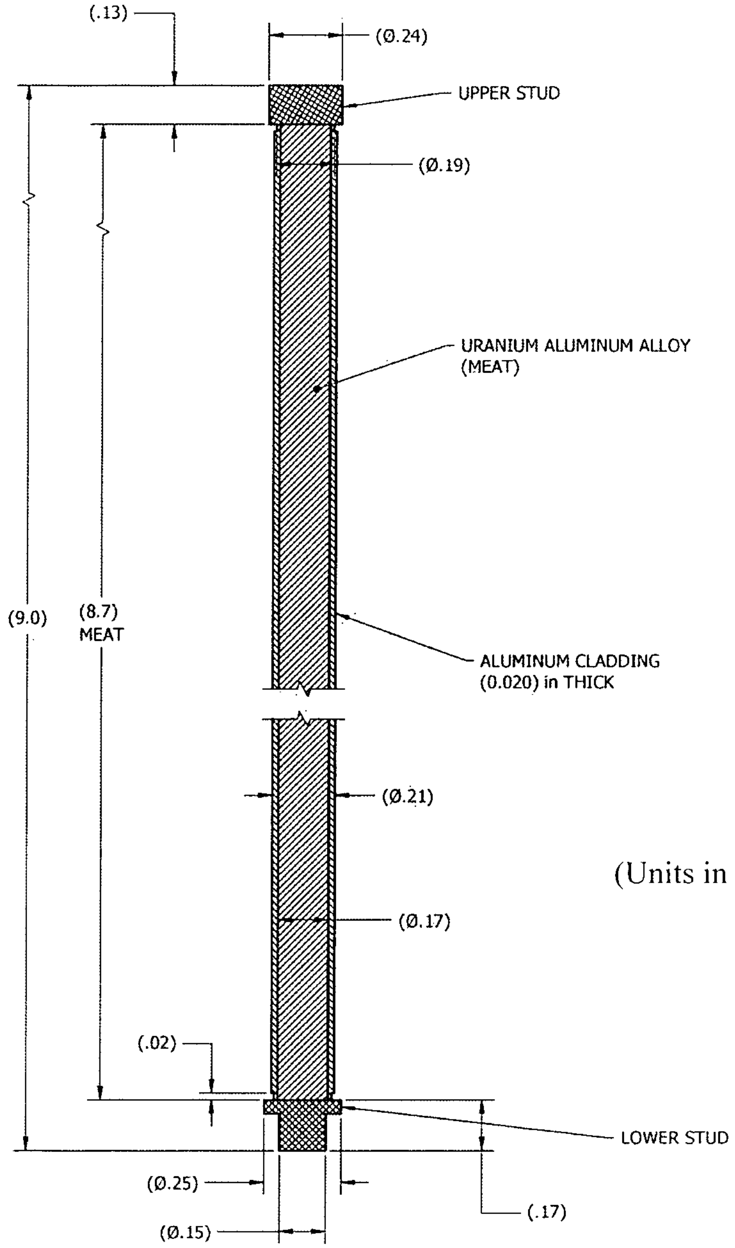


Figure 1.2.3-18 ANSTO Damaged Fuel Can (DFC)



NOTES
1. DFC MATERIAL - 6061 ALUMINIUM
2. DIMENSIONS NOMINAL

Figure 1.2.3-19 SLOWPOKE Fuel Element



(Units in inches)

Table 1.2.3-1 Characteristics of Design Basis TRIGA Fuel Elements Acceptable for Loading in the Poisoned TRIGA Basket

	TRIGA HEU (Notes 1, 2, 6 & 7)	TRIGA LEU (Notes 1, 2, 6 & 7)	TRIGA LEU (Notes 1, 2, 6 & 7)
Fuel Form	Clad U-ZrH rod	Clad U-ZrH rod	Clad U-ZrH rod
Maximum Element Weight, lbs	13.2	13.2	13.2
Maximum Element Length, in	47.74	47.74	47.74
Element Cladding	Stainless Steel	Stainless Steel	Aluminum
Clad Thickness, in	0.02	0.02	0.03
Active Fuel Length, in	15	15	14-15 (Note 4)
Element Diameter, in	1.478 max.	1.478 max.	1.47 max.
Fuel Diameter, in	1.435 max.	1.435 max.	1.41 max.
Maximum Initial U Content/Element, kilograms	0.196	0.845	0.205
Maximum Initial ^{235}U Mass, grams	137	169	41
Maximum Initial ^{235}U Enrichment, weight percent	70	20	20
Zirconium Mass, grams (Note 5)	2060	1886-2300	2300
Hydrogen to Zirconium Ratio, max. (Note 5)	1.6	1.7	1.0
Maximum Average Burnup, MWd/MTU	460,000 (80% ^{235}U)	151,100 (80% ^{235}U)	151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)	90 days (Note 3)	90 days (Note 3)

Notes:

1. Mixed TRIGA LEU and HEU contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Aluminum clad fuel with 14-inch active fuel is solid and has no central hole with a zirconium rod.
5. Zirconium mass and H/Zr ratio apply to the fuel material (U-Zr-H_x) and do not include the center zirconium rod.
6. Listed TRIGA fuel elements have a 0.225-inch diameter zirconium rod in the center.
7. Dimensions listed are as-fabricated (unirradiated) nominal values.

Table 1.2.3-2 Characteristics of Design Basis TRIGA Fuel Elements Acceptable for Loading in the Nonpoisoned TRIGA Basket

	TRIGA HEU (Notes 1, 2, 6)		TRIGA LEU (Notes 1, 2, 6)		TRIGA LEU (Notes 1, 2, 6)
Fuel Form	Clad U-ZrH rod (Note 4)		Clad U-ZrH rod (Note 4)		Clad U-ZrH rod (Note 4)
Maximum Element Weight, lbs	13.2		13.2		13.2
Maximum Element Length, in	47.74		47.74		47.74
Element Cladding	Stainless Steel		Stainless Steel		Aluminum
Minimum Clad Thickness, in	0.01		0.01		0.01
Active Fuel Length, in	(Note 5)		(Note 5)		(Note 5)
Maximum Element Diameter, in	1.5 max.		1.5 max.		1.5 max.
Fuel Diameter, in	(Note 5)		(Note 5)		(Note 5)
Maximum Initial U Content/Element, kilograms	0.198	0.186	0.845	1.447	0.205
Maximum Initial ^{235}U Mass, grams	138	175 (Notes 7, 8)	169	275 (Notes 7, 8)	41
Maximum Initial ^{235}U Enrichment, weight percent	71	95 (Notes 7, 8)	25	25 (Notes7, 8)	25
Zirconium Mass, grams	(Note 5)		(Note 5)		(Note 5)
Hydrogen to Zirconium Ratio, max.	(Note 5)		(Note 5)		(Note 5)
Maximum Average Burnup, MWd/MTU	460,000 (80% ^{235}U)	583,000	151,100 (80% ^{235}U)		151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)		90 days (Note 3)		90 days (Note 3)

Notes:

1. Mixed LEU and HEU TRIGA fuel element, and LEU and HEU TRIGA fuel cluster rod, as defined in Table 1.2-3, contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Element may contain zirconium rod in the center.
5. See criticality analyses in Chapter 6, Section 6.4.5.6, for the evaluations determining critical fuel characteristics.
6. Dimensions listed are as-fabricated (unirradiated) nominal values.
7. Elements limited to loading in top and bottom basket module only.
8. Elements limited to a maximum of three per basket module cell.

Table 1.2.3-3 Characteristics of Design Basis TRIGA Fuel Cluster Rods

Element Type	TRIGA Fuel Cluster Rod
Max. Rod Length (in)	31.0
Max. Active Length (in)	22.5
Clad Material	Incoloy 800
Min. Clad Thickness (in)	0.015
Fuel Material	U-ZrH
Max. Pellet Diameter (in)	0.53
Max. Rod Weight (kg)	0.65
Min. U in U-ZrH (wt %)	43.0 (LEU) or 9.5 (HEU)[1]
Max. ^{235}U in U (wt %)	19.9 to 93.3
^{235}U Mass (g)	55.0 (LEU) or 46.5 (HEU)
Max. H to Zr Ratio	1.7

[1] Equivalent to a maximum zirconium mass of 357 g for LEU fuel and 457 g for HEU fuel material. Lower weight percents are permitted, provided the maximum zirconium mass limits are not exceeded.

Table 1.2.3-4 Fuel Characteristics

Parameter	PWR Fuel Assembly	BWR Fuel Assembly	PWR Rods	High Burnup PWR Rods	PWR MOX Fuel Rods[6]	High Burnup BWR Rods 7 × 7	High Burnup BWR Rods[1] 8 × 8[2]
Maximum Number of Assemblies, Elements or Rods	1	2	25 rods	25 rods	16 rods	25 rods	25 rods
Maximum Overall Weight, lbs	1650	750	N/A	N/A	N/A	N/A	N/A
Maximum Overall Length, in	178.25	176.1	162	162	162	176.1	176.1
Maximum Active Fuel Length, in	150	150	150	150	153.5	150	150
Fuel Rod Cladding	Zirc	Zirc	Zirc	Zirc	Zirc	Zirc	Zirc
Maximum Uranium, kg U	475	198	58.2	65.6	41.6[7]	198	198
Maximum Initial ^{235}U, wt %	*See below*[3]	4.0	5.0	5.0	7.0 max/2.0 min, fissile Pu[8]	5.0	5.0
Maximum Burnup, MWd/MTU	35,000	30,000	60,000[4]	80,000	62,500	60,000 - 80,000	80,000
Maximum Unit Decay Heat, kW	2.5	1.1	0.564	0.92	0.143	0.84	0.84
Maximum Cask Decay Heat, kW	2.5	2.2	1.41	2.3	2.3	2.1	2.1
Minimum Cool Time, yr	2	2	150 days	150 days	90 days	210 - 270 days[5]	150 days

1 High burnup rods are loaded in a fuel assembly lattice or rod holder. Up to 14 rods, loaded in a rod holder, may be classified as damaged. The lattice may be irradiated.
2 Includes rods from all larger BWR assembly arrays (e.g., 9×9, 10×10).
3 See Table 1.2.3-5 for maximum PWR fuel enrichment by fuel type.
4 Up to 2 of the 25 PWR rods may have a maximum burnup of 65,000 MWd//MTU.
5 Minimum cool time for high burnup BWR 7×7 rods is determined by extent of burnup. See Section 5.3.8 and Table 5.3.8-23.
6 Up to 16 PWR MOX fuel rods or a combination of up to 16 MOX PWR and UO_2 PWR fuel rods can be loaded.
7 Maximum fuel mass is 2.6 kg HM/rod.
8 Maximum 5.0 wt % ^{235}U for UO_2 rods.

Table 1.2-4 Fuel Characteristics (Continued)

Parameter	Metallic Fuel	Metallic Fuel	Metallic Fuel	MTR HEU	MTR MEU	MTR LEU	TRIGA LEU Element	TRIGA HEU Element	TRIGA Cluster Rod
Maximum Number of Assemblies, Elements or Rods	15 rods (sound)	9 rods (failed)	3 rods (severely failed in filters)	42[1]	42	42[2]	140	140	560
Maximum Overall Weight, lbs	1805	1805	1805	30 (max)[3]	30 (max)[3]	30 (max)[3]	13.2 (max)[3]	8.82 (nom.) 13.2 (max)[3]	1.5[3]
Maximum Overall Length, in	120.5	120.5	120.5	25.4[4]	26.1[4]	26.1[4]	47.74[5]	47.74[5]	31.0
Maximum Active Fuel Length, in	120.0	120.0	120.0	24.8	25.6	25.6	15	15	22.5
Fuel Rod Cladding	Al	Al	Al	Al	Al	Al	Al or SS	Al or SS	Incoloy 800
Maximum Uranium, kg U	54.5	54.5	54.5	0.422 0.511	0.950	2.474 3.368[2]	0.824	0.196	0.0505 (HEU) 0.2894 (LEU)
Maximum Initial ^{235}U, wt %	Natural	Natural	Natural	94	94[6]	25	20	70	95 (HEU)/20 (LEU)
Maximum Burnup, MWd/MTU	1,600	1,600	1,600	Variable up to 660,000[7]	Variable up to 293,300	Variable up to 139,300	151,100 (80% ^{235}U)	460,000 (80% ^{235}U)	600,000 (HEU)/ 140,000 (LEU) (80% ^{235}U)
Maximum Unit Decay Heat, kW	0.036	0.036	0.036	Variable[8]	0.030[8]	0.030[8]	0.0075	0.0075	0.001875
Maximum Cask Decay Heat, kW	0.54	0.54	0.54	1.26	1.26	1.26	1.05	1.05	1.05
Minimum Cool Time, yr	1	1	1	Variable[8]	Variable[8]	Variable[8]	Variable[9]	Variable[9]	Variable[9]

1 For NISTR fuel, 42 assemblies may be cut in half, producing 84 fuel-bearing pieces. Each fuel-bearing piece may contain up to 0.211 kgU.

2 MTR fuel elements having ^{235}U content >470 g (>22 g per plate) are limited to a total of 4 elements in a 7-element basket. Basket openings 1, 2 and 3 shall be blocked by cell block spacers to ensure that MTR elements are not loaded in these openings. Therefore, depending on the number of such 4-element baskets, the maximum number of elements per cask will be reduced accordingly.

3 Maximum weight of fuel element(s), spacer(s) and fuel can, as applicable, per basket module cell shall be 80 pounds.

4 For MTR fuel elements, which are cut to remove nonfuel-bearing hardware prior to transport, a nominal 0.28 inch of nonfuel or spacer hardware will remain above and below the active fuel region to allow for fuel handling operations. The HFBR element, with an element length of 57.24 inches, must be cut prior to shipment. For HEU MTR elements having >380 g ^{235}U but less than 460 g ^{235}U, a minimum of 2.0 cm (0.8 inch) of nonfuel hardware and/or spacers/plates shall be provided at the ends of the element.

5 Permissible fuel element length is limited to basket cavity length, which is a minimum 47.74 inches for the basket top module, 30.94 inches for the intermediate modules, and 32.64 inches for thebottom module.

6 Typical MEU enrichment is 45 wt% ^{235}U. Criticality analysis supports up to 94 wt% under the MEU fuel definition.

7 Maximum burnup is 660,000 MWd/MTU for 380g ^{235}U and 577,500 MWd/MTU for 460g ^{235}U.

8 Minimum cool times for MTR fuel, down to 30 days, shall be determined using the procedure presented in Section 7.1.5.

9 Minimum cool times for TRIGA fuel elements and fuel cluster rods, down to 90 days, are determined so that the maximum decay heat of any element to be shipped is ≤ 7.5 watts and any fuel cluster rod is ≤ 1.875 watts.

Table 1.2.3-4 Fuel Characteristics (Continued)

Parameter	DIDO HEU	DIDO MEU	DIDO LEU
Number of Fuel Cylinders per Assembly	4	4	4
Maximum Overall Weight (lb)[1]	15	15	15
Minimum Plate Thickness, in	0.051	0.051	0.051
Minimum Clad Thickness (Al), in	0.00984	0.00984	0.00984
Maximum ^{235}U per Element, g	190	190	190
Maximum Initial ^{235}U, wt %	94	94	94
Minimum Initial ^{235}U, wt %	90	40	19
Maximum Uranium, kg U	0.2111	0.4750	1.0000
Minimum Active Fuel Height, in	23.13	23.13	23.13
Minimum Element Height[2], in	24.21	24.21	24.21
Maximum Burnup, MWd/MTU	577,460	256,650	121,910
Maximum Unit Decay Heat[3], kW	0.025	0.025	0.025
Maximum Cask Decay Heat, kW	1.05	1.05	1.05
Minimum Cool Time[4], yr	Variable	Variable	Variable

1 Maximum weight of fuel element(s), spacer(s) and fuel can, as applicable, per basket module cell shall be 80 pounds.

2 Element height provides for spacing of fissile material. An optional spacer may be used to maintain spacing if the element is cut shorter than 24.21 inches.

3 Maximum unit decay heat of 0.025 kW allowed only in conjunction with spacers for top basket (see Section 7.1.4). The per element heat load is limited to 0.018 kW with no top basket spacer. For DIDO fuel elements loaded into a top ANSTO basket module, the maximum decay heat load is limited to 0.010 kW per element (with or without DFC).

4 Minimum cool times for DIDO fuel assemblies, down to 180 days, shall be determined using the procedure presented in Section 7.1.4.

Table 1.2.3-5 PWR Fuel Characteristics

Fuel Type	No. of Fuel Rods	Max. Assembly Length (in.)	Max. Assembly Weight (lb)	Max. Enrich. (wt %)	Max. MTU	Pitch (in.)	Rod Dia. (in.)	Clad Thick. (in.)	Pellet Dia.(in.)	Max. Active Length (in.)
B&W 15 × 15	208	165.63	1515	3.5	0.4750	0.5680	0.430	0.0265	0.3686	144.0
B&W 17 × 17	264	165.72	1505	3.5	0.4658	0.5020	0.379	0.0240	0.3232	143.0
CE 14 × 14	176	157.00	1270	3.7	0.4037	0.5800	0.440	0.0280	0.3765	137.0
CE 16 × 16	236	178.25	1430	3.7	0.4417	0.5060	0.382	0.0250	0.3250	150.0
WE 14 × 14 Std	179	159.71	1302	3.7	0.4144	0.5560	0.422	0.0225	0.3674	145.2
WE 14 × 14 OFA	179	159.71	1177	3.7	0.3612	0.5560	0.400	0.0243	0.3444	144.0
WE 15 × 15	204	159.71	1472	3.5	0.4646	0.5630	0.422	0.0242	0.3659	144.0
WE 17 × 17 Std	264	159.77	1482	3.5	0.4671	0.4960	0.374	0.0225	0.3225	144.0
WE 17 × 17 OFA	264	160.10	1373	3.5	0.4282	0.4960	0.360	0.0225	0.3088	144.0
Ex/ANF 14 × 14 WE	179	160.13	1271	3.7	0.3741	0.5560	0.424	0.0300	0.3505	144.0
Ex/ANF 14 × 14 CE	176	157.24	1292	3.7	0.3814	0.5800	0.440	0.0310	0.3700	134.0
Ex/ANF 15 × 15 WE	204	159.70	1433	3.7	0.4410	0.5630	0.424	0.0300	0.3565	144.0
Ex/ANF 17 × 17 WE	264	159.71	1348	3.5	0.4123	0.4960	0.360	0.0250	0.3030	144.0

Table 1.2.3-6 BWR Fuel Characteristics

Fuel Type	No. of Fuel Rods	No. of Water Rods	Max. Assembly Length (in.)	Max. Assembly Weight (lb)	Max. MTU	Pitch (in.)	Rod Dia. (in.)	Clad Thick. (in.)	Pellet Dia. (in.)	Max. Active Length (in.)
GE 7 x 7	49	0	175.9	678.9	0.1923	0.738	0.563	0.037	0.477	146
GE 8 x 8-1	63	1	175.9	681.0	0.1880	0.640	0.493	0.034	0.416	146
GE 8 x 8-2	62	2	175.9	681.0	0.1847	0.640	0.483	0.032	0.410	150[1]
GE 8 x 8-4	60	4	176.1	665.0	0.1787	0.640	0.484	0.032	0.410	150[1,2]
GE 9 x 9	74	2[3]	176.1	646.0	0.1854	0.566	0.441	0.028	0.376	150[1,4]
	79	2	176.1	646.0	0.1979	0.566	0.441	0.028	0.376	150[1,4]
Ex/ANF 7 x 7	49	0	171.3	619.1	0.1960	0.738	0.570	0.036	0.490	144
Ex/ANF 8 x 8-1	63	1	171.3	562.3	0.1764	0.641	0.484	0.036	0.4045	145.2
Ex/ANF 8 x 8-2	62	2	176.1	587.8	0.1793	0.641	0.484	0.036	0.4045	150
Ex/ANF 9 x 9	79	2	176.1	575.3	0.1779	0.572	0.424	0.03	0.3565	150
	74	2[3]	176.1	575.3	0.1666	0.572	0.424	0.03	0.3565	150

[1] 6" natural uranium blankets on top and bottom.
[2] May have 1 large water hole - 3.2 cm ID, 0.1 cm thickness.
[3] 2 large water holes occupying 7 fuel rod locations - 2.5 cm ID, 0.07 cm thickness.
[4] Shortened active fuel length in some rods.

Table 1.2.3-7 Characteristics of General Atomics Irradiated Fuel Material (GA IFM)

Parameter	RERTR	HTGR
Maximum Number of Assemblies, Elements or Rods	13 intact; 7 sectioned	N/A
Maximum Loaded Enclosure Weight, lbs	76.0	71.5
Maximum Fuel Weight, lbs	23.73	23.52
Maximum Overall Length, in	29.92	N/A
Maximum Active Fuel Length, in	22.05	N/A
Fuel Material	U-ZrH	UC_2, UCO, UO_2, $(Th,U)C_2$, $(Th,U)O_2$
Fuel Rod Cladding	Incoloy 800	N/A
Maximum Uranium, kg U	3.86	0.21
Maximum Initial ^{235}U, wt %	19.7	93.15
Maximum Burnup, MWd/MTU	N/A	N/A
Maximum Unit Decay Heat, W	11.0	2.05
Maximum Cask Decay Heat, W	13.05	13.05
Earliest Shipment Date	1/1/96	1/1/96
Maximum Activity, Ci	2920	483

Table 1.2.3-8 Typical Production TPBAR Characteristics[1]

Parameter Description	Value
Maximum Number of TPBARs per Consolidation Canister	300
Number of Consolidation Canisters per Cask	1
TPBAR Clad Material	316 L Stainless Steel
Rod Length[2], in	153.04
Rod Diameter[2], in	0.381
Maximum Rod Heat Load, W	2.31
Maximum Cask Heat Load, kW	0.693
Maximum Tritium Content per Rod, gram	1.2
Maximum Activity per Cask[3], Ci	3.84×10^6
Loaded TPBAR Consolidation Canister Maximum Weight, pounds[4]	1,000
Maximum Event Failed Tritium Release (Ci/rod)	<55
Minimum Cooling Time, days	30

1 Refer to Section 1.5, Chapter 1 Appendices, Unclassified DOE Reference Documents and Drawings.
2 Beginning of life, nominal, unirradiated dimensions.
3 Primary dose contribution: 1.1×10^4 Ci ^{60}Co/cask
4 The bounding weight employed in the structural analysis.

Table 1.2.3-9 PULSTAR Fuel Characteristics

Description	Value
Maximum Pellet Diameter (inch)	0.423
Minimum Element (Rod) Cladding Thickness (inch)	0.0185
Minimum Element (Rod) Diameter (inch)	0.470
Maximum Active Fuel Height (inch)	24.1
Element (Rod) Length (inch)	26.2
Rod Pitch (inch)	0.525×0.607
Assembly Length (inch)	38
Box Outside Width (inch)	2.745×3.155
Box Thickness (inch)	0.06
Maximum Assembly or Loaded Can Weight (lb)[1]	80
Maximum PULSTAR Can Content Weight (lb)[2]	39.6
Maximum Enrichment (wt % ^{235}U)	6.5
Maximum ^{235}U Content per Element (g)	33
No. of Elements (Rods) per Assembly	25
No. of Elements (Rods) per Can[2]	25
Maximum Depletion (%^{235}U)	45
Minimum Cool Time (yrs)	1.5
Maximum Heat Load per Assembly (W)	30
Maximum Heat Load per Element (W)	1.2

[1] Listed weight is the maximum weight evaluated for the structural calculation to bound all payload configurations, including loaded cans, and spacers. Nominal PULSTAR assembly weight is 45 pounds.

[2] The contents of a PULSTAR can are restricted to the equivalent of the fuel material in 25 PULSTAR fuel elements and of the displaced volume of 25 intact PULSTAR fuel elements. Fuel material may be in damaged form including fuel debris. The listed weight represents the can content limit established by the structural analyses.

Table 1.2.3-10 Spiral Fuel Assembly Characteristics

Parameter	Value
Number of elements per assembly	10
Fuel element type	Curved plate
Nominal dimensions of element (cm)	0.147 × 7.33 × 63.5 (individual plate)
Chemical form of fuel meat	U-Al_x-alloy
Cladding material	Aluminum
Nominal over-all dimensions (cm)	63.818 (height) × 10.16 diameter[1]
Max total weight of ^{235}U (g)	160 (total per assembly)
Maximum enrichment (wt % ^{235}U)	95
Side plate material	Aluminum (inner and outer tubes)
Nominal side plate – dimensions (cm)	Inner 6.045 OD, 5.82 ID × 63.818 Outer 10.16 OD, 9.85 ID × 63.818[2]
Max. assembly weight (lb)	18[3]
Assembly maximum heat load (W)	15.7[4]
Burnup/cool time limit	Variable[5]

1 Cropped to fit within ANSTO fuel basket module nominal height of 28.3 inches.

2 Criticality evaluations reduced inner and outer shell thickness to 0.01 cm to provide additional moderator within the assembly.

3 Typical assembly weight is 7.9 pounds. Bounding structural analysis weight is listed. Bounding weight includes DFC and fuel plates.

4 Thermal and shielding evaluation employed 18 W per element. Based on cool time constraint, 15.7 W represents maximum heat load. Spiral fuel elements with degraded cladding loaded into aluminum DFCs shall be limited to a maximum decay heat of 10 W per element.

5 Spiral fuel is constrained to DIDO MEU cool time limits as a function of burnup. Minimum cool times for the spiral assembly, down to 270 days, shall be determined using the procedure presented in Section 7.1.4 for 18 W DIDO MEU fuel.

Table 1.2.3-11 MOATA Plate Bundle Characteristics

Parameter	Value
Maximum number of elements per assembly	14
Nominal dimensions of element (cm)	66 cm long, 7.6 cm wide and 0.203 cm thick
Nominal dimensions of fuel meat (cm)	58.4 cm long, 6.99 cm wide and 0.1016 cm thick (bounding active fuel width evaluated to a maximum of 7.32 cm)
Chemical form of fuel meat	U-Al_x-alloy
Cladding material	Aluminum
Nominal clad thickness (cm)	0.05 cm (evaluated to 0.01 cm minimum)
Plate spacer thickness (cm)	0.147 min, 0.152 max (evaluated to 0.18 maximum)
Maximum weight of ^{235}U (g) per plate	22.3
Maximum enrichment (wt % ^{235}U)	92
Nominal side plate thickness (cm) Max. assembly weight (lb)	0.635 (bounding evaluation replaced by cavity moderator) 18[1]
Maximum heat load per assembly (W)[2]	3 (total for 14 fuel plates)
Maximum burnup	30,000 MWd/MTU or 4.1 % depletion ^{235}U
Minimum cool time (years)	10

1 Typical assembly weight is 13.6 pounds. Bounding structural analysis weight is listed. Bounding weight includes DFC and fuel plates.

2 Actual heat load at limiting burnup and cool time < 1 Watt. Thermal evaluations at 3 Watt per bundle.

Table 1.2.3-12 Typical TPBAR Segment Characteristics in Waste Container

Parameter/Description	Value
Maximum Number of TPBAR Segments and Debris per Waste Container, equivalent number of TPBARs	55
Number of Waste Containers per Cask	1
Waste Container Material	316L Stainless Steel
Maximum Tritium Content per TPBAR equivalent, gram	1.2
Maximum Activity per Cask, Ci	6.66 ×10+5
Maximum Heat Load per Waste Container, watts Maximum Loaded Waste Container Weight, pounds	127 700[1]
Minimum Cooling Time, years	90

[1] Design basis weight of a loaded waste container is 700 pounds. Applying a maximum payload of 55 TPBARs, with storage canister, yields a maximum weight of 662 pounds. Use of shrouds to contain segments and/or TPBAR debris reduces overall waste container weight due to a reduction in TPBAR payload capacity resulting from the reduced container free volume.

Table 1.2.3-13 Solid, Irradiated Hardware Characteristics[1]

Parameter	Value
Maximum Content Weight Maximum Content Length	4,000 pounds [2] 171.5 inches [3]
Hardware Material	Solid, irradiated and contaminated fuel assembly structural or reactor internal component hardware [4]
Maximum Cask Heat Load	1.0 KW
Maximum Activity per Cask, Ci	6.0 x 10E+6
Maximum Source Term, gamma/sec	6.0 x 10E+15
Maximum Source Term, MeV/sec	1.0 x 10E+15

[1] Maximum content weight includes any spacers, containers or dunnage loaded in the cavity with the irradiated hardware.

[2] Length of cavity is limited to 171.5 inches by the installation and use of an irradiated hardware spacer bolted to the underside of the closure lid.

[3] Appropriate secondary containers will be used to prevent any contact and cross-contamination between the carbon steel contents and the stainless steel internals of the cask cavity.

[4] The irradiated hardware contents may contain fissile material, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53.

Table 1.2.3-14 SLOWPOKE Fuel Rods

Parameter	Value
Maximum Cask Heat Load	5 W
Maximum Canister Heat Load	0.625W
Payload Limit (lb/canister)	23.1
Maximum ^{235}U per rod (g)	2.800
Maximum U per rod (g)	3.111
Minimum Cool Time	14 yr
Maximum Burnup (GWd/MTU or wt% ^{235}U Depletion)	30 GWd/MTU 4.5 wt% ^{235}U

Notes:

1.) Heat load limit established by thermal analysis.

2.) Fissile material (^{235}U) mass limit established by criticality analysis.

3.) Fuel (U) mass, cool time, burnup/depletion limit established by shielding analysis.

4.) Payload weight limit established by structural analysis and includes both fuel and canister weight.

R-217263

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK LID ASSY
SAFETY ANALYSIS
REPORT

PROJECT	DRAWING	REV
315-40	04	12
SCALE N.T.S. (12) EST. WT. (12)	SH 1 OF 1	

1K-21727'

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER LID ASSEMBLY,
SEALED FAILED FUEL CAN,
TRIGA FUEL

PROJECT 315-40		DRAWING 087	REV 7
SCALE FULL	WEIGHT 5#	SH 1 OF 1	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL			
CANISTER BODY ASSEMBLY, SEALED FAILED FUEL CAN, TRIGA FUEL			
PROJECT 315-40		DRAWING 088	REV 3
SCALE N.T.S. ⟨3⟩		SH 1 OF 1	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL			
LEGAL WEIGHT TRUCK TRANSPORT CASK ASSY. PWR/BWR ROD TRANSPORT CANISTER			
PROJECT 315-40		DRAWING 104	REV 6
SCALE	WEIGHT	SH 1 OF 3	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK ASSY
PWR/BWR ROD
TRANSPORT CANISTER

PROJECT 315-40	DRAWING 104	REV 6
	SH 2 OF 3	

A

1

E-213,234

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK ASSY
PWR/BWR ROD
TRANSPORT CANISTER

PROJECT 315-40	DRAWING 104	REV 6
	SH 3 OF 3	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK ASSY
TPBAR SHIPMENT
SAFETY ANALYSIS REPORT

PROJECT 315-40		DRAWING 128	REV 4
SCALE N.T.S. 4		SH 1 OF 2	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK ASSY
TPBAR SHIPMENT
SAFETY ANALYSIS REPORT

PROJECT 315-40		DRAWING 128	REV 4
SCALE N.T.S.	ESTIMATE	SH 2 OF 2	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER BODY ASSEMBLY, SEALED FAILED FUEL CAN, PULSTAR

(2)

A

PROJECT 315-40	DRAWING 129	REV 2	
SCALE N.T.S. (2)	SH 1 OF 1		

1

Kr 2 1 7 2 7

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
SEALED FAILED FUEL CAN,
2 PULSTAR

PROJECT 315-40		DRAWING 130	REV 2
SCALE N.T.S. 2		SH 1 OF 1	

K-121727

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TRANSPORT CASK ASSEMBLY,
PULSTAR SHIPMENT,
LWT CASK

PROJECT 315-40		DRAWING 133	REV 2
SCALE N.T.S.	2	SH 1 OF 2	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TRANSPORT CASK ASSEMBLY,
PULSTAR SHIPMENT,
LWT CASK

PROJECT 315-40		DRAWING 133	REV 2
SCALE N.T.S.(2)	EST. WT.	SH 2 OF 2	

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BODY WELDMENT,
SCREENED FUEL CAN,
PULSTAR FUEL

PROJECT 315-40		DRAWING 134	REV 2
SCALE N.T.S.	2	SH 1 OF 1	

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER ASSEMBLY
SLOWPOKE FUEL

PROJECT 315-40		DRAWING 156	REV 1
SCALE N.T.S	WEIGHT N/A	SH 1 OF 4	1/26/2012

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER ASSEMBLY
SLOWPOKE FUEL

PROJECT 315-40		DRAWING 156	REV 1
SCALE N. T. S.	WEIGHT N/A	SH 2 OF 4	1/29/2012

A

1

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER ASSEMBLY
SLOWPOKE FUEL

PROJECT 315-40		DRAWING 156		REV 1
SCALE N. T. S.	WEIGHT N/A	SH 3 OF 4		1/26/2012

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CANISTER ASSEMBLY
SLOWPOKE FUEL

PROJECT 315-40		DRAWING 156	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 4 OF 4	1/26/2012

A

1

R1-217042

Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LEGAL WEIGHT TRUCK
TRANSPORT CASK ASSY
SLOWPOKE FUEL

PROJECT 315-40		DRAWING 158	REV 0
SCALE N.T.S	WEIGHT N/A	SH 1 OF 1	12/12/2011

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2.2 Weights and Centers of Gravity

2.2.1 Major Component Statistics

The weights of the major components of the NAC-LWT cask and their respective centers of gravity are presented in Table 2.2.1-1. The axial location of the center of gravity is measured from the bottom surface of the cask body. The center of gravity is always on the longitudinal centerline of the cask because the cask is essentially axisymmetric about that axis. The center of gravity location of the fuel is representative of typical fuel configurations.

The weights and centers of gravity of the cask package in eight different shipping configurations are presented in Table 2.2.1-2. In each case, the center of gravity is measured from the bottom surface of the cask body. The term "loaded" refers to the presence of fuel or other radioactive materials in the cask cavity; the term "empty" implies the absence of any fuel or other radioactive materials in the cask cavity. However, the fuel basket does remain in the cask cavity for the "empty" configuration. The weight of a lifting yoke is not included in the tabulated package weights.

All of the values tabulated in Table 2.2.1-1 and Table 2.2.1-2 are calculated to the nearest pound to obtain an accurate cask weight and center of gravity. The cask package weight and center of gravity used in the analyses of this report are the design values - 52,000 pounds and 98.93 inches. A design value of 4,000 pounds is conservatively used for the total weight of the cask contents (including the appropriate basket).

Table 2.2.1-1 Weights of the NAC-LWT Cask Major Components

Component	Weight (pounds)	Axial Center of Gravity Location (inches)
Cask Body	39,906	96.46
Closure Lid and Bolts	941	195.11
Impact Limiters Top Bottom	 1,535 1,320	 202.98 -3.18
Shield Tank Fluid	3,506	96.26
PWR Fuel Basket and Spacer	874	100.98
PWR High Burnup Rod Payload	1,620	95.33
PWR Fuel Payload (Maximum)	3,126	96.63
BWR Fuel Basket	1,124	97.88
BWR Fuel Payload	1,500	97.88
Metallic Fuel Basket	128	96.40
Metallic Fuel Payload	2,080	96.40
MTR Four Unit Basket	982	96.20
MTR Four Unit Fuel Payload	840[1]	96.20
MTR Four Unit PULSTAR Fuel Payload	2,240[2]	96.20
MTR Five Unit Basket	1,015	96.20
MTR Five Unit Fuel Payload	1,050[1]	96.20
MTR Six Unit Basket	1,002	96.20
MTR Six Unit Fuel Payload	1,260[1]	96.20
GA IFM Basket and Spacer	818	98.06
GA IFM Fuel Payload	148	167.34
TPBAR Basket and Spacer	675	110.40
TPBAR Payload	978[3]	96.00
ANSTO Basket	911	100.95
ANSTO Payload	756	100.95
TPBAR Basket	575	97.43
TPBAR with Rod Transport Canister Payload	1,326[4]	102.26

1 For conservatism, a design basis MTR fuel weight of 30 lbs/assy is used in the structural analysis. The maximum MTR element weight is 13.2 lbs for an intact element and 9.7 lbs for the cut elements in the 42-element configuration. The maximum weight for the SLOWPOKE canister is 23 pounds.

2 For conservatism, a bounding weight of 80 pounds is considered for each of the 28 fuel cells for PULSTAR fuel

3 TPBAR payload represents the combined weight of the TPBAR and consolidation canister. A conservative 1,000 lb weight is applied in the structural analysis.

4 TPBAR with Rod Transport Canister payload represents the combined weight of the 25 TPBARs, the PWR /BWR Rod Transport Canister and the PWR insert.

$$M.S. = \frac{S_y}{S_{c_{max}}} - 1 = \underline{+0.22}$$

2.6.12.5.3 Compressive Stress Calculation

For the same reasons that are stated in Section 2.6.12.3.2, the metallic fuel basket needs only to be self-supporting. The metallic fuel basket weighs 128 pounds, which during a normal operating conditions 1-foot fall, is decelerated at 15.8 g. The total compressive load acting over the three fuel tubes is P_F = 2,022 pounds (128 lb × 15.8 g). The total cross sectional area of the three aluminum fuel tubes is 6.48 square inches, resulting in a normal operating conditions compressive stress ($S_{c_{fuel}}$) of 312 psi (2,022 lb/6.48 in^2). The spacer tube must support the basket and its contents during a bottom end drop. The total weight that the spacer tube supports is 2,208 pounds, resulting in a normal operating conditions 1-foot bottom end drop compressive load of P_c = 34,886 pounds. The cross sectional area of the 9.0-inch outer diameter aluminum spacer tube is 6.87 square inches, resulting in a compressive stress ($S_{c_{spacer}}$) of 5078 psi. Assuming that the impacting end is fixed and the other end is free, the critical buckling stresses for each tube column (Shigley, page 116) is calculated:

Fuel Tubes (3)

$$P_{cr} = \frac{n\pi^2 EI}{L^2}$$

$$= 33,550 \text{ psi}$$

where:

n = 0.25, end fixity coefficient
$E_{Al250°F}$ = 9 x 4 10^6 psi
$I_{basket\ body}$ = 46 in^4
L = 145.25 in, fuel tube length
MS = $\underline{P_{cr}}$ / P_F - 1 = $\underline{+Large}$

Spacer Tube

$$P_{cr} = \frac{n\pi^2 EI}{L^2}$$

$$= 1,754,000 \text{ psi}$$

n = 0.25, end fixity coefficient
$E_{Al250°F}$ = 9.4 x 10^6 psi
$I_{spacer\ tube}$ = 66 in^4
L = 29.5 in, spacer tube length
MS = $\underline{P_{cr}}$ / P_c - 1 = $\underline{+Large}$

2.6.12.6 MTR Fuel Basket Construction

The MTR modular basket assembly has five configurations. One configuration is for 28 uncut (intact) MTR fuel assemblies (28 MTR – 4 unit basket); the second is for 35 partially cut MTR elements that have had portions of the upper and lower end fittings removed (35 MTR – 5 unit basket). The third configuration is for 42 MTR fuel assemblies (42 MTR – 6 unit basket) with the upper and lower end fittings removed; and the fourth configuration is for up to 700

PULSTAR fuel elements loaded in the 28 MTR basket. The PULSTAR fuel may be intact fuel assemblies, intact fuel elements (rods) loaded in a fuel rod insert or in fuel cans, or damaged fuel elements, fuel debris, and nonfuel components of fuel assemblies loaded in fuel cans. The fifth configuration is for up to 800 SLOWPOKE intact and/or damaged fuel rods contained in canisters. Each MTR basket configuration consists of one base module, one top module, and two, three or four intermediate modules for the 28, 35 and 42 element configurations, respectively. Each MTR basket module is designed to hold up to seven MTR or PULSTAR fuel assemblies. The modules are not interchangeable between basket configurations. Up to four (4) SLOWPOKE canisters may be loaded in each of the top and the upper intermediate MTR-28 basket modules with the center fuel cells blocked. The lower intermediate and bottom basket modules are installed as axial spacers. The structural analysis is not affected by the specific fuel element design or enrichment as long as the fuel characteristics are in compliance with the fuel characteristics listed in Table 1.2-4 or Table 1.2.3-14.

Axial fuel and plate spacers may be used to axially position the MTR fuel assemblies in the basket modules. Cell block spacers are used to prevent the loading of fuel assemblies in basket module positions 1, 2 and 3 when LEU MTR fuel elements having >470 g ^{235}U per element (>22 g ^{235}U per plate) are loaded. The presence and/or use of spacers, fuel plate canisters or fuel cans does not affect the structural integrity of the MTR fuel baskets as the total weight of fuel element, spacer and fuel plate canister or fuel can is limited to the evaluated load of 80 pounds/cell. The axial fuel and cell block spacers perform no safety function and are considered dunnage. Plate spacers are used, if required, to ensure that the criticality evaluation required minimum nonfuel hardware is provided.

Each module, fabricated from Type 304 stainless steel, is a weldment made up of two 1/2-inch thick, 13.265-inch diameter, circular plates at each end of the longitudinal divider plates creating seven MTR fuel assembly cavities. The outside wall of the four symmetric outermost fuel compartments is fabricated from 11-gage Type 304 stainless steel sheet. The 1/2-inch thick plate at the top end of the MTR fuel basket module is welded to the exterior surfaces of the fuel tube weldment with a 1/8-inch continuous weld on the under side of the top plate and with a continuous fillet seal weld on the top side. The 1/2-inch thick baseplate is continuously welded to the 1/4-inch thick divider plates and the 5/16-inch thick web plates. The 11-gage sheet metal and the 5/16-inch intermediate webs are discontinued at 1/4 inch from the surface of the baseplate to provide for compartment drainage. The 5/16-inch plate material may be machined to a minimum thickness of 0.28 inch. In addition to the drainage path at the base of each assembly cavity, a 1-inch diameter hole is located at the center of each of the compartments in the module. Each MTR basket base module sits on a 1.5-inch long, 10-inch schedule 80S pipe welded to the 1/2-inch thick baseplate. The 10-inch schedule 80S pipe carries the total weight of the MTR basket assembly and bears directly on the bottom forging of the cask.

The MTR fuel basket base module and intermediate modules have guide pins fixed to the surface of the top support plate. The guide pins fit into holes in the base plate of the top and intermediate modules and provide controlled alignment of the basket assembly. A groove slot on the outside of

Transport Canister (66 lbs). Therefore, no further analysis is required for the 4×4 and 5×5 inserts.

2.6.12.13 SLOWPOKE Fuel Canister Assembly

Evaluation of the SLOWPOKE fuel canister assembly components for normal conditions of transport includes longitudinal (end) and lateral (side) 1-foot drop evaluation. The 'g' loads of 20 g's and 25 g's (Table 2.6.7-34) are conservatively assumed for a 1-foot end drop and a 1-foot side drop, respectively. Evaluation of all the assembly components for the normal condition of transport is given below. Table 2.6.12-3 gives the summary of weights and other dimensions of the SLOWPOKE fuel canister assembly components.

Table 2.6.12-3 Summary of SLOWPOKE Fuel Canister Assembly Component Dimensions and other Inputs

Component	Canister Weldment	Canister Insert	Fuel
Weight, W, lbs (6)	11.3+1.6=12.9(1)	6.8(4)	3.445(5)
Length, L, in	39.81	9.25×4=37(2)	8.66×4=34.64(2)
Outer Dimension/Diameter, D, in	3.3	0.50	0.206
Inner Dimension/Diameter, d, in	2.8	0.402	0.166
Number of Component, n	1	25	25
Area of Each Tube, A, in^2	2.8(3)	6.943×10^{-2}	1.169×10^{-2}
Stiffness K= (EA/L), lbs/in	670,987	447,517	80,464
Material	6061-T6 Aluminum	6061-T6 Aluminum	6061-T6 Aluminum
Elastic Modulus, $\times10^6$ psi (7)	9.54	9.54	9.54

Note:
1) The weight of the canister weldment is conservatively assumed as11.3 lbs instead of 11 lbs.
2) Total Length of four sets arranged in the canister assembly
3) Maximum canister weldment cross sectional area
4) Weight of four 5×5 inserts
5) Weight of 100 fuel rods and each fuel rod weighs 15.659 grams (Reference 7)
6) The total weight of the SLOWPOKE fuel canister assembly is 23.145 lbs (12.9+6.8+3.445)
7) Elastic modulus is taken at 200°F

Canister Weldment

Canister weldment is a square tube made of four quarter inch aluminum plates welded along the length of the plate. The canister weldment is evaluated for longitudinal (end) and lateral (side) drop for normal condition of transport. For stress analysis, the area of the canister weldment is calculated near the lid where the cross sectional area is minimum. The detailed analysis of the canister weldment is given below.

Evaluation of canister weldment for the 1-foot end drop:

The compressive stress on the canister weldment is calculated as:

$$\sigma_c = \frac{WG_{E1}}{2(A1 + A2)} = \frac{11.3 \times 20}{2(0.5225 + 0.4975)} = 111\, psi$$

where:

W = 11.3 lbs, Weight of the canister weldment
G_{E1} = 20 g, 1-foot end drop 'g' load
A1 = (2.8-0.71) ×0.25=0.5225 in^2, Cross sectional area of Side wall-A
A2 = (2.8-0.81) ×0.25=0.4975 in^2, Cross sectional area of Side wall-B

Margin of safety is:

$$MS = \frac{Membrane.\ Allowable}{Compressive\ Stress} - 1 = \frac{10500}{111} - 1 = +\text{Large}$$

Buckling analysis of canister weldment:

Column slenderness ratio separating elastic and inelastic buckling:

$$C_C = \sqrt{\frac{2\pi^2 E}{F_y}} = \sqrt{\frac{2\pi^2 \times 9.54 \times 10^6}{3.12 \times 10^4}} = 77.65$$

Slenderness ratio Kl/r:

$$kl/r = \frac{1.0 \times l}{\sqrt{I/A}} = \frac{1.0 \times 39.44}{\sqrt{3.66/2.04}} = 29.42$$

where:

E = 9.54×10^6 psi, Elastic modulus of aluminum at 200°F
F_y = 3.12×10^4 psi, Yield strength of aluminum at 200°F
l = 39.44 in, Length of canister weldment wall
I = $2 \times \left(\frac{2.8 \times 0.25^3}{12} + 2.8 \times 0.25 \times \left(2.8/2\right)^2 \right) + 2 \times \left(\frac{0.25 \times 2.8^3}{12} \right) = 3.66\ in^4$, Moment of inertia

A = $2 \times (0.523 + 0.498) = 2.04\ in^2$, Cross sectional area of weldment

If Kl/r < C_C, then

Allowable compressive stress on the tube:

$$F_a = \frac{\left[1 - \frac{(kl/r)^2}{2C_C^2}\right] F_y}{\frac{5}{3} + \frac{3(kl/r)}{8C_C} - \frac{(kl/r)^3}{8C_C^3}} = \frac{\left[1 - \frac{(29.42)^2}{2 \times 77.65^2}\right] 3.12 \times 10^4}{\frac{5}{3} + \frac{3(29.42)}{8 \times 77.65} - \frac{(29.42)^3}{8 \times 77.65^3}} = 16{,}089$$

Margin of safety is:

$$MS = \frac{Allowable\,Comp.\,Stress}{Maximum\,Comp.\,Stress} - 1 = \frac{16089}{111} - 1 = +\text{Large}$$

For the 1-foot side drop, the weldment wall is assumed to be fixed at all the corners.

The bending stress on the canister weldment wall is calculated as (Roark's, 7th Edition, Table 11.4):

$$\sigma_b = \frac{-\beta_1 q b^2}{t^2} = \frac{-(0.5 \times 0.640 \times 2.8^2)}{0.25^2} = -40\ psi$$

where:

β_1 = 0.5, Parameter for a/b = ∞

q = $\frac{W}{4} \times G_{S1} \times \frac{1}{A} = \frac{11.3}{4} \times 25 \times \frac{1}{39.44 \times 2.8} = 0.64\ psi$, Uniformly distributed load

W = 11.3 lbs, Weight of the canister weldment

G_{S1} = 25 g, 1-foot side drop 'g' load

A = 39.44×2.8 = 110.43in^2, Area of the weldment wall

Margin of safety:

$$MS = \frac{Bending\ Allowable}{Compressive\ Stress} - 1 = \frac{15750}{40} - 1 = +\text{Large}$$

Canister Lid

Canister lid is a one-inch aluminum plate which covers the top end of the canister weldment. The lid assembly includes lid plate, handle, housing, latch, and handle cap screw. For normal condition of transport analysis purpose, the lid is assumed to be simply supported at the outer edges and the total weight (inertial load) is assumed to act at the central rectangular area. The total weight of SLOWPOKE fuel canister assembly with fuel is 23.14 pounds.

For the 1-foot end drop:

The bending stress on the canister lid is calculated by assuming the inertial load acting at the center of the lid and with simply supported at all edges (Roark's, 7th Edition, Table 11.4).

$$\sigma_{b-Lid} = \frac{\beta W}{t^2} = \frac{1.28 \times W_t \times G_{E1}}{t^2} = \frac{1.28 \times 23.14 \times 20}{1.0^2} = 593\ psi$$

where:

β = 1.28, Parameter for $a_1/b = a/b_1 = 0.2$
W_t = 23.14 lbs, Weight of the canister weldment assembly with fuel
G_{E1} = 20 g, 1-foot end drop 'g' load
t = 1.0 in^2, Thickness of the lid

Margin of safety is:

$$MS = \frac{Bending\ Allowable}{Bending\ Stress} - 1 = \frac{15750}{593} - 1 = +Large$$

During the 1-foot end drop, the lid handle can shear or puncture the canister lid. The shear stress on the lid due to the handle is:

$$\tau_{l-h} = \frac{W_t G_{E1}}{A_1} = \frac{W_t G_{E1}}{2(w_1 + t_1) \times 2} = \frac{23.14 \times 20}{2(1.0 + 0.65) \times 2} = 70\ psi$$

where:

w_1 = 1.0 in, Width of the handle
t_1 = 0.65 in, Thickness of the handle

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{15615}{70} - 1 = +Large$$

During the 1-foot end drop, the canister weldment can shear the lid. The shear stress on the lid due to canister weldment is:

$$\tau_{l-w} = \frac{W_t G_{E1}}{A_2} = \frac{W_t G_{E1}}{4 w_2 t_2} = \frac{23.14 \times 20}{(4 \times 2.75 \times (1 - 0.38))} = 68\ psi$$

where:

w_2 = 2.75 in, Width of the lid at the canister weldment seating
t_2 = 1-0.38=0.62 in, Thickness of the lid at the canister weldment seating

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{15615}{68} - 1 = +Large$$

Lid Handle

The aluminum lid handle is 0.65" thick and 2.5" tall and is fastened to the lid plate using two socket head cap screws. For the end drop, the entire end drop load is transferred through the lid handle.

For 1-foot end drop:

The membrane or compressive stress on the lid handle is calculated as:

$$\sigma_{mh} = \frac{W_t G_{E1}}{A_3} = \frac{W_t G_{E1}}{2 \times w_3 t_3} = \frac{23.14 \times 20}{2 \times 1.0 \times 0.65} = 356\, psi$$

where:

W_t = 23.14 lbs, Weight of the canister weldment assembly with fuel
G_{E1} = 20 g, 1-foot end drop 'g' load
w_3 = 1.0 in, Width of the lid handle
t_3 = 0.65 in, Thickness of the lid handle

Margin of safety is:

$$MS = \frac{Membrane\, Allowable}{Membrane\, Stress} - 1 = \frac{10500}{356} - 1 = +Large$$

The bearing stress on the lid handle is during end drop is calculated as:

$$\sigma_{bH} = \frac{W_t G_{E1}}{A_4} = \frac{W_t G_{E1}}{L_h \times t_3} = \frac{23.14 \times 20}{3.75 \times 0.65} = 190\, psi$$

where:

L_h = 3.75 in, Length of the lid handle
t_3 = 0.65 in, Thickness of the lid handle

Margin of safety is:

$$MS = \frac{Bearing\, Allowable}{Bearing\, Stress} - 1 = \frac{31230}{190} - 1 = +Large$$

Housing

The housing supports the latch and the spring mechanism and these three components maintain the position of the lid. Figure 2.6.12-9 shows the housing with dimensions and critical locations for evaluation. On the housing, compressive or membrane stresses are evaluated at Location MN, bearing stresses are evaluated at Location RS, bending stresses are evaluated at Location A and shear stresses are evaluated at Locations A and B.

For the end drop, the canister insert along with fuel contacts the housing at Location RS (Figure 2.6.12-9). The weight of the insert and the fuel transferred to the housing is 10.245 pounds.

The membrane stress on the housing Location MN (Figure 2.6.12-9) is calculated as:

$$\sigma_{mHo} = \frac{W_t G_{E1}}{A_5} = \frac{W_t G_{E1}}{2(w_5 - d)t_5} = \frac{(6.8 + 3.445) \times 20}{2(0.75 - 0.38)0.75} = 369\,psi$$

where:

W_t = 6.8+3.445 = 10.245 lbs, Weight of the canister insert and fuel

G_{E1} = 20 g, 1-foot end drop ‘g’ load

W_5-d = (0.75-0.38) = 0.37 in, Width of the housing at Location MN

t_5 = 0.75 in, Thickness of the housing at Location MN

Margin of safety is:

$$MS = \frac{Membrane\ Allowable}{Membrane\ Stress} - 1 = \frac{10500}{369} - 1 = +Large$$

The bearing stress on the housing Location RS (Figure 2.6.12-9) is calculated as:

$$\sigma_{bHo} = \frac{W_t \times G_{E1}}{A_6} = \frac{W_t \times G_{E1}}{2 \times w_5 \times t_5} = \frac{(6.8 + 3.445) \times 20}{2 \times 0.75 \times 0.75} = 182\,psi$$

where:

W_5 = 0.75 in, Width of the housing at Location RS

t_5 = 0.75 in, Thickness of the housing at Location RS

Margin of safety is:

$$MS = \frac{Bearing\,Allowable}{Bearing\ Stress} - 1 = \frac{31230}{182} - 1 = +Large$$

The shearing stress on the housing Location A (Figure 2.6.12-9) is:

$$\tau_A = \frac{W_t G_{E1}(R)}{A_7} = \frac{W_t G_{E1}(R)}{2h_1 \times 2t} = \frac{(6.8 + 3.445) \times 20 \times \left(0.38/0.75\right)}{2 \times 0.25 \times 2 \times 0.75} = 138\,psi$$

where:

R = Ratio of load on the curved section

$$= \frac{\text{Thickness of curved section (Length AA)}}{\text{Thickness of housing (Length RS)}} = \frac{0.38}{0.75} = 0.51$$

(Note: R = 0.51, Half of the load on the housing is conservatively applied on the curved section of the housing)

h_1 = 0.25 in, Height of the housing at Location A

t = 0.75 in, Thickness of the housing at Location A

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{15615}{138} - 1 = +Large$$

The bending stress on the housing Location A is (assumed as beam fixed at two ends):

$$\sigma_A = M\frac{Y}{I} = \frac{W_a L^2}{24}\frac{Y}{I} = \frac{W_t G_{E1} R}{L}\frac{L^2}{24}\frac{Y}{I}$$

$$= \frac{(6.8 + 3.445) \times 20\left(0.38/0.75\right)}{0.38}\frac{0.38^2}{24}\frac{0.25/2}{9.77 \times 10^{-4}} = 421\ psi$$

where:

I = $(bh_1^3/12) = (0.75\times0.25^3/12) = 9.77\times10^{-4}$ in^4, Moment of inertia at Location A

Y = 0.25/2 in, Height of the housing from mid point at Location A

L = 0.38 in, Length of the housing (Length AA)

The stress intensity on the housing Location A is:

$$\sigma_S = \sqrt{\sigma_A^2 + 4\tau_A^2} = \sqrt{421^2 + 4 \times 138^2} = 504\ psi$$

Margin of safety is:

$$MS = \frac{Bending\ Allowable}{Stress\ Intensity} - 1 = \frac{15750}{504} - 1 = +Large$$

Location B:

The shearing stress on housing Location B is:

$$\tau_B = \frac{W_t G_{E1}(R)}{A_B} = \frac{W_t G_{E1}(R)}{2(h_2 \times t)} = \frac{(6.8 + 3.445) \times 20 \times \left(0.38/0.75\right)}{2(0.06 \times 0.75)} = 1{,}154\ psi$$

where:

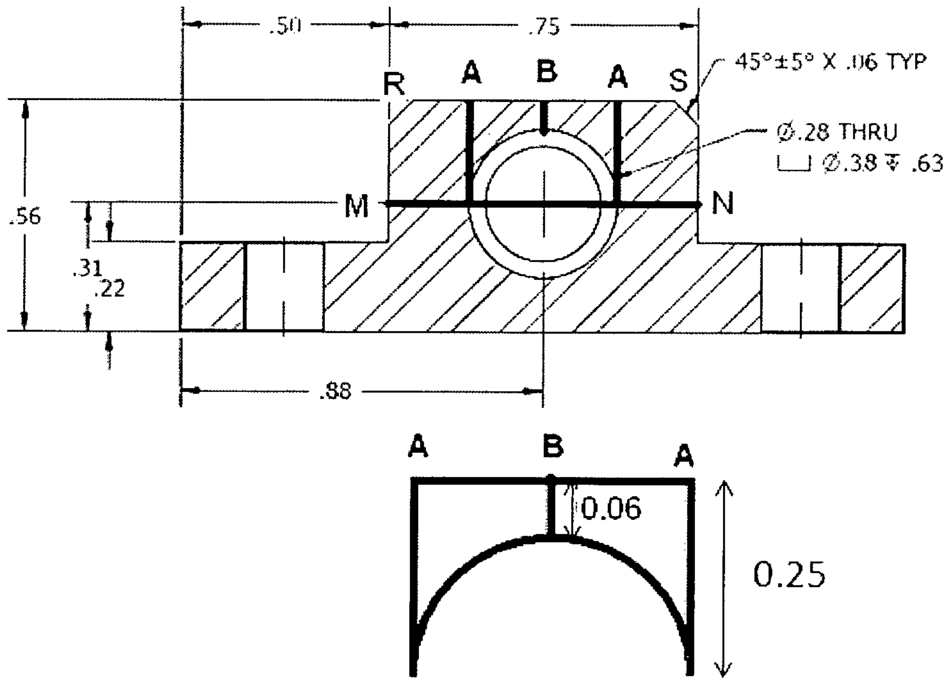
h_2 = 0.06 in, Height of the housing at Location B (Figure 2.6.12-9)

t = 0.75 in, Thickness of the housing at Location B (Figure 2.6.12-9)

Margin of safety is:

$$MS = \frac{Shearing\ Allowable}{Shearing\ Stress} - 1 = \frac{15615}{1154} - 1 = +Large$$

Figure 2.6.12-9 SLOWPOKE Fuel Canister Assembly Housing



Canister Insert

The fuel tubes are placed inside the aluminum canister insert (four inserts per canister) during transportation. The canister insert tubes are arranged in 4×4 or 5×5 matrix on a base plate. The canister insert is evaluated for normal conditions of transport only. Conservatively for analysis purposes, the fuel weight is added to the canister insert weight. Both the 4×4 and 5×5 matrix insert tubes have same thickness (0.1"), but the 5×5matrix insert tube is smaller than the 4×4 matrix insert tube. Therefore, only 5×5 matrix insert are evaluated because 5×5 matrix insert has lower moment of inertia and area and weighs more than the 4×4 matrix insert.

For the 1-foot end side:

The compressive or membrane stress on the canister insert is calculated as:

$$\sigma_{ci} = \frac{W_{if} G_{E1}}{25 \times A} = \frac{W_{if} \times G_{E1}}{25 \times \pi(D^4 - d^4)/4} = \frac{(6.8 + 3.445) \times 20}{25 \times \pi(0.5^2 - 0.402^2)/4} = 118\ psi$$

where:

W_{if} = 6.8+3.445 = 10.245 lbs, Weight of the canister insert and fuel

G_{E1} = 20 g, 1-foot end drop 'g' load

D = 0.5 in, Outer diameter of the 5×5 insert tube

d = 0.402 in, Inner diameter of the 5×5 insert tube

Margin of safety is:

$$MS = \frac{Membrane.\ Allowable}{Compressive\ Stress} - 1 = \frac{10500}{118} - 1 = +Large$$

Buckling Analysis of canister weldment:

Column slenderness ratio separating elastic and inelastic buckling:

$$C_C = \sqrt{\frac{2\pi^2 E}{F_y}} = \sqrt{\frac{2\pi^2 \times 9.54 \times 10^6}{3.12 \times 10^4}} = 77.65$$

Slenderness ratio Kl/r:

$$kl/r = \frac{1.0 \times l}{\sqrt{I/A}} = \frac{1.0 \times 9.25}{\sqrt{\dfrac{\pi(0.5^4 - 0.402^4)/64}{\pi(0.5^2 - 0.402^2)/4}}} = 57.67$$

where:

E = 9.54×10^6 psi, Elastic modulus of aluminum at 200°F

F_y = 3.12×10^4 psi, Yield strength of aluminum at 200°F

l = 9.25 in, Length of canister insert tube

$$I = \pi(0.5^4 - 0.402^4)/64 = 1.79\times10^{-3}\ \text{in}^{-3}\text{, Moment of inertia of insert tube}$$

$$A = \pi(0.5^2 - 0.402^2)/4 = 0.069\ \text{in}^2\text{, Cross sectional area of the insert tube}$$

If Kl/r < C_C, then

Allowable compressive stress on the tube:

$$F_a = \frac{\left[1 - \frac{(kl/r)^2}{2C_C^2}\right]F_y}{\frac{5}{3} + \frac{3(kl/r)}{8C_C} - \frac{(kl/r)^3}{8C_C^3}} = \frac{\left[1 - \frac{(57.67)^2}{2\times 77.65^2}\right]3.12\times10^4}{\frac{5}{3} + \frac{3(57.67)}{8\times 77.65} - \frac{(57.67)^3}{8\times 77.65^3}} = 14{,}770$$

Margin of safety is:

$$MS = \frac{Allowable\,Comp.Stress}{Maximum\,Comp.Stress} - 1 = \frac{14770}{118} - 1 = +Large$$

For the 1-foot side drop, the insert is assumed to be simply supported at ends and the weight is acting along the length of the insert.

The bending stress on the insert is calculated as (Roark's, 7th Edition, Table 8.1):

$$\sigma_{bi} = M\frac{y}{I} = \frac{W_a l^2}{8}\frac{D/2}{I} = \frac{W_{if}G_{S1}}{25\times l}\frac{l^2}{8}\frac{D/2}{I} = \frac{(6.8+3.445)\times 25}{25\times 9.25}\frac{9.25^2}{8}\frac{0.5/2}{1.79\times10^{-3}} = 1{,}658\ psi$$

where:

G_{S1} = 25 g, 1-foot side drop 'g' load

Margin of safety is:

$$MS = \frac{Bending.Allowable}{Bending\,Stress} - 1 = \frac{15750}{1658} - 1 = 8.5$$

Lid Latch and Plunger

For the 1-foot top end drop, the load path through the lid, does not contain the lid latches or the plungers.

For 1-foot side drop analysis of latch and plunger, the accident condition loads (60 g's) are used with normal condition allowable. This bounds both the normal and accident conditions of transport.

During the side drop, the latch experience 60 g's and the maximum shear stress occur on the Location SA.

The shear stress on latch at Location SA (Figure 2.6.12-10):

$$\tau_{SA} = \frac{W_P G_{S1}}{A_{SA}} = \frac{0.05 \times 60}{0.5 \times 0.12} = 50\ psi$$

where:

W_P = Weight of the latch = Volume ×Density of Aluminum
~ 0.5×0.5× (1.13+0.31) ×0.1 = 0.036 ~ 0.05 lbs

G_{S1} = 60 g, 30-foot side drop 'g' load

A_{SA} = 0.5× (0.76-2×0.3) × tan (30) = 0.046 in^2, Area of the latch at Location SA

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{15615}{50} - 1 = +\text{Large}$$

Shear stress on the steel plunger is:

$$\tau_{PL} = \frac{W_P G_{S1}}{A_P} = \frac{W_P G_{S1}}{\pi/4 \times d^2} = \frac{0.05 \times 60}{\pi/4 \times 0.16^2} = 149\ psi$$

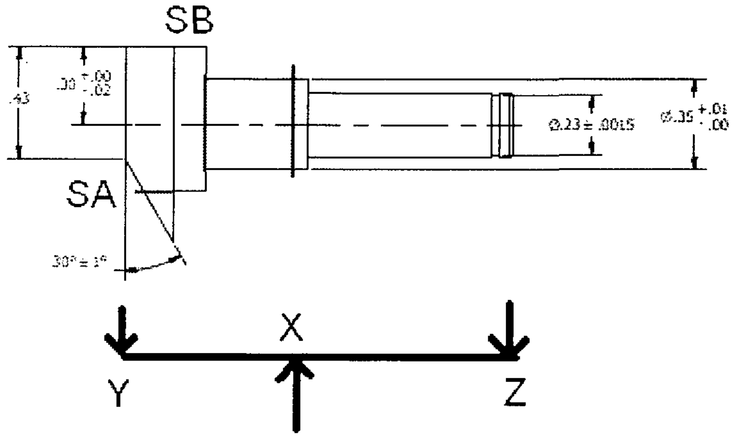
where:

d = 0.16 in, Diameter of the plunger nose

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{6250}{149} - 1 = +Large$$

Figure 2.6.12-10 SLOWPOKE Fuel Canister Assembly Latch and the Free Body Diagram



2.6.12.14 Conclusion

Loads generated during normal operations conditions for each basket assembly design result in total equivalent stresses, which each basket body can adequately sustain. Analyses show that all basket-bearing stresses during a side drop are much less than the material yield strength. Column analyses demonstrate that each basket assembly is self-supporting during an end drop. The minimum Margin of Safety, for all basket designs, is +0.10 as reported in Section 2.6.12.7.4 for the TRIGA basket; +0.003 as shown in Table 2.6.12-2 for the DIDO basket; +0.10 as reported in Section 2.6.12.9.2 for the GA fuel basket; +0.26 as reported in Section 2.6.12.11.1 for the ANSTO basket, and +8.5 as reported in Section 2.6.12.13 for the SLOWPOKE fuel canister assembly. Therefore, it can be concluded that all basket designs have sufficient structural integrity for adequate service during normal conditions of transport.

P_{1g} = 2,208 lb/1.25 in = 1,766 lb/in

$P_{49.7g}$ = (1,766 lb/in)(49.7 g) = 87,790 lb/in

The side drop g load, 49.7 g, is obtained from Table 2.6.7-34.

The allowable bearing stress is the lesser value of the yield strength of aluminum or of stainless steel, which is 23,800 psi; the yield strength (S_y) of Type 304 stainless steel at 250°F. The margin of safety is calculated as:

$$M.S. = \frac{S_y}{S_{c_{max}}} - 1 = \underline{+0.22}$$

2.7.7.5.2 Compressive Buckling Stress Calculations – End Drop

The metallic fuel basket and inner cavity length are designed to minimize the movement of the basket and its contents relative to the cask. The basket length is designed to provide adequate clearance to ensure that differential thermal expansion will not produce longitudinal loads on the basket body. The basket is self-supporting during an end drop accident. Therefore, the basket components are examined for compressive column stability, to ensure that the basket structure is adequate. The fuel tubes are considered to support the weight of the metallic fuel basket (128 lbs) in a top end drop. The spacer tube supports the basket and contents (2,208 lbs) during a bottom end drop. The maximum end drop load results from a 58.2 g cask deceleration. The fuel tubes have a total cross-sectional area of 6.48 square inches; therefore, the accident compressive stress ($S_{c_{fuel}}$) in each tube is (128 lbs)(58.2 g)/6.48 in^2 = 1,150 psi. During a bottom end drop, the spacer tube sustains a compressive stress ($S_{c_{spacer}}$) of (2,208)(58.2)/6.87 = 18,705 psi.

An Euler column analysis is used to determine the critical buckling stress of the basket fuel tubes and the spacer. Assuming that the impacting end is fixed and the other is free, the critical buckling stress (Shigley, page 116) is calculated:

Fuel Tubes (3)

P_{max} = (128)(58.2) = 7450 lbs

Spacer Tube

P_{max} = (2,208)(58.2) = 128,506 lbs

Fuel Tubes (3)

$$P_{cr} = \frac{n\pi^2 EI}{L^2} = 26{,}930 \text{ lbs}$$

Spacer Tube

$$P_{cr} = \frac{n\pi^2 EI}{L^2} = 1{,}754{,}000 \text{ lbs}$$

where:

$n = 0.25$, end fixity coefficient

$E_{AL250°F} = 9.4 \times 10^6$ psi

$I_{fuel\ tubes} = 45.8\ in^4$

$L = 145.3$ in, fuel tube length

$MS = P_{cr} / P_{max} - 1 = \underline{+LARGE}$

$n = 0.25$, end fixity coefficient

$E_{AL250°F} = 9.4 \times 10^6$ psi

$I_{spacer\ tube} = 65.8\ in^4$

$L = 29.5$ in, spacer tube length

$MS = P_{cr} / P_{max} - 1 = \underline{+LARGE}$

2.7.7.6 MTR Fuel Basket Construction

The MTR modular basket assembly has five configurations. One configuration is for intact fuel; the second is for 35 partially cut fuel elements that have had portions of the upper and lower end fittings removed; the third configuration is for 42 MTR fuel assemblies with the end fittings removed; and the fourth is for PULSTAR fuel elements loaded in the 28 MTR basket. The PULSTAR fuel elements may be intact fuel assemblies, intact fuel elements (rods) loaded in a fuel rod insert or a PULSTAR can, or failed fuel elements loaded in a PULSTAR failed fuel can or a PULSTAR screened fuel can. The fifth MTR basket configuration is for SLOWPOKE fuel rods contained in canisters. For SLOWPOKE fuel, up to four (4) canisters (each containing up to 100 SLOWPOKE intact and/or damaged fuel rods) may be loaded in both the top and upper intermediate MTR-28 basket modules. The lower intermediate and bottom MTR-28 basket modules are installed as axial spacers. Each basket configuration consists of one base module, one top module, and either two, three or four intermediate modules for the 28-, 35- and 42-element configurations, respectively. Each 28 MTR basket module is capable of holding up to seven MTR type fuel assemblies or up to seven PULSTAR fuel assemblies.

The hypothetical accident analyses for the MTR fuel baskets are not affected by the specific fuel element design or enrichment as long as the fuel characteristics are in compliance with the fuel characteristics listed in Table 1.2-4.

Each module, fabricated from Type 304 stainless steel, is a weldment made up of two 1/2-inch thick, 13.625-inch diameter, and circular plates at each end of the longitudinal divider plates creating seven MTR fuel assembly compartments. The outside wall of the four symmetric outermost fuel tubes is fabricated from 11- gage Type 304 stainless steel sheet. The 1/2-inch thick plate at the top end of the MTR basket module is welded to all webs and divider plates. The 1/2-inch thick baseplate is continuously welded to the 1/4-inch thick divider plates and the 5/16-inch thick web plates. The 11-gage sheet metal and the 5/16-inch intermediate webs are discontinued at 1/4 inches from the surface of the baseplate to provide full tube drainage. The 5/16-inch plate material may be machined to a minimum thickness of 0.28 inch. In addition, a 1-inch diameter hole is located at the center of each of the compartments in each basket module. The MTR basket base module sits on a 1.5-inch long, 10-inch schedule 80S pipe welded to the

$$= 70 \times \left(\frac{8.87}{2} - \frac{1}{2}\right) \times \frac{1}{2}\left(\frac{8.87}{2} - \frac{1}{2}\right) = 542 \text{ in-lb/in, the maximum bending moment}$$

t = 0.5 in, thickness of the flange

The margin of safety is

$$MS = \frac{S_u}{S_b} - 1 = \frac{31.5}{13} - 1 = +1.42$$

where:

S_u = 31.5 ksi, ultimate strength of 6061-T6 aluminum @300°F

TPBAR Basket – End Drop

In an end drop scenario, the basket is only loaded by its own weight. The TPBAR basket weight is bounded by what was considered for the end drop analysis of the TPBAR basket in Section 2.7.7.12.1. Therefore, the results of the previous analysis are bounding and no further analysis is required.

TPBAR Basket Upper Fitting

The TPBAR basket upper fitting is identical to what was analyzed for the TPBAR basket in Section 2.7.7.12. The TPBAR basket weight and component temperatures considered in Section 2.7.7.12 are conservative. Therefore, the results of the 2.7.7.12 are bounding and no further analysis is required.

TPBAR Basket Lower Fitting

The TPBAR basket lower fitting is identical to the lower fitting of the PWR basket. The weight of the loaded TPBAR basket is less than the weight of the loaded PWR basket. Therefore, no further analysis is required.

PWR Insert

The PWR insert contains the PWR/BWR Rod Transport Canister assembly for insertion into the TPBAR basket. The PWR insert is not a structural component and, therefore, an analysis for accident conditions is not required.

PWR/BWR Rod Transport Canister Assembly Analysis

The PWR/BWR Rod Transport Canister is identical to what was analyzed in Section 2.7.1.7. The fuel weight considered in the Section 2.7.1.7 analysis for the PWR/BWR fuel (350 lbs) bounds the weight of 25 TPBARs in the PWR/BWR Rod Transport Canister (66 lbs). Therefore, no further analysis is required for the PWR/BWR Rod Transport Canister.

Internal Spacer

The internal spacer for the PWR/BWR Rod Transport Canister is identical to what was analyzed in Section 2.7.1.7. The fuel weight considered in the Section 2.7.1.7 analysis for the PWR/BWR fuel (350 lbs) bounds the fuel weight of 25 TPBARs in the PWR/BWR Rod Transport Canister (66 lbs). Therefore, no further analysis is required for the internal spacer.

4×4 and 5×5 Inserts

The 4×4 and 5×5 inserts for the PWR/BWR Rod Transport Canister are identical to what was analyzed in Section 2.7.1.7. The fuel weight considered in the Section 2.7.1.7 analysis for the PWR/BWR fuel (350 lbs) bounds the weight of up to 25 TPBARs in the PWR/BWR Rod Transport Canister (66 lbs max.). Therefore, no further analysis is required for 4×4 and 5×5 inserts.

2.7.7.15 SLOWPOKE Fuel Canister Assembly

Evaluation of the SLOWPOKE fuel canister assembly for accident conditions of transport includes evaluation of canister weldment and lid assembly for both lateral (side) and longitudinal (end) 30-foot drop. Based on canister, basket, and cask drawings, the maximum gap between the canister assembly and the cask lid is 0.71" (0.15+0.56), the maximum gap between the canister insert and the canister lid is 0.5", and the maximum gap between the fuel and the lid is 1.09" (0.5+ 0.59) as shown in Table 2.7.7-3.

The 'g' load on the cask for the 30-foot end drop is calculated using the experimental 30-foot end drop force-deflection curve (Figure 2.10.12-18) of the quarter scale model of the NAC-LWT cask system. The maximum force experienced by the quarter scale NAC-LWT cask system for 30-foot end drop is 138 kips. Therefore the maximum 'g' load on the full scale NAC-LWT cask system is:

$$G = \frac{F_Q \times C}{W} = \frac{138 \times 10^3 \times 16}{48 \times 10^3} = 46 g's$$

where:

F_Q = 138 kips, Maximum force on the quarter scale cask
W = 48 kips, Weight of the full scale NAC LWT cask (Table 2.2.1-1)
C = 16, Conversion factor between quarter and full scale model

The axial loading on SLOWPOKE fuel canister components is due to the inertial load of the components on the cask cavity end in the end drop orientation. For 30-foot side drop, the 'g' load is taken as 60 g's (Table 2.6.7-34).

The relative velocity of the components and the cask end at the time of the contents' contact with the end is determined by performing numerical integration of the accelerations and velocities. The initial position of the fuel, the canister, and lid are assumed to correspond to the maximum gap condition (Table 2.7.7-3).

All components have an initial velocity of 527.5 in/sec and are allowed to free fall under a 1g gravity acceleration. The cask lid also has the same initial velocity, but a deceleration of 46 g's from Figure 2.10.12-18 (factored by 4) is applied to the cask lid to allow the components to contact the cask lid. Contact occurs when the gaps between the components are closed.

Table 2.7.7-4 gives the summary of the initial and cumulative gap and the relative velocity of the canister components. ANSYS is used for computing the force between the SLOWPOKE fuel canister components by modeling them as lumped mass and gap elements (compression only elements). The force obtained from the ANSYS is used in evaluating the SLOWPOKE fuel canister components. The stiffness of the components to be used in the ANSYS finite element model is calculated as follows:

The stiffness of the canister weldment is calculated as:

$$K = \frac{EA}{L} = \frac{9.54 \times 10^6 \times 2.8 \times 0.25 \times 4}{39.81} = 670{,}987\, lbs/in.$$

where:

E = 9.54×10^6 psi, Elastic modulus of the aluminum at 200°F

A = $2.8 \times 0.25 \times 4 = 2.8$ in^2, Cross sectional area of the canister weldment

L = 39.81 in, Length of the canister weldment

The stiffness of the canister insert is calculated as:

$$K = \frac{EA}{L} = \frac{E \times \pi \times {(D^2 - d^2)}/{4} \times n}{L \times m} = \frac{9.54 \times 10^6 \times \pi \times {(0.5^2 - 0.402^2)}/{4} \times 25}{9.25 \times 4} = 447{,}517\, lbs/in.$$

where:

D = 0.5 in, Outer diameter of the canister insert tube

d = 0.402 in, Inner diameter of the canister insert tube

L = 9.25 in, Length of the canister insert tube

n = 25, Number of insert in 5×5 matrix

m = 4, Number of 5×5 matrix insert per canister assembly

The stiffness of the fuel is calculated as:

$$K = \frac{EA}{L} = \frac{E \times \pi \times {(D^2 - d^2)}\big/{4} \times n}{L \times m}$$

$$= \frac{9.54 \times 10^6 \times \pi \times {(0.206^2 - 0.166^2)}\big/{4} \times 25}{8.66 \times 4} = 80{,}464\, lbs/in.$$

where:

D	=	0.206 in, Outer diameter of the SLOWPOKE fuel tube
d	=	0.166 in, Inner diameter of the SLOWPOKE fuel tube
L	=	8.66 in, Length of the SLOWPOKE fuel tube
n	=	25, Number of fuel tubes in 5×5 matrix insert
m	=	4, Number of 5×5 matrix insert per canister assembly

Table 2.7.7-3 Length and Initial Gap of SLOWPOKE Fuel Canister Assembly Components

Component	Length/Gap (inches)
LWT Cask Inner Length	177.65
Length of Four Baskets	44×3+45.5=177.5
Gap between Cask Lid and Basket	177.65-177.5=0.15
Basket Length	43.5
Total Length of Slowpoke Assembly	0.38+39.44+0.62+2.5=42.94
Gap between Basket and Slowpoke Canister Assembly	43.5-42.94=0.56
Total Gap between Slowpoke Canister and Cask Lid	0.15+0.56=0.71
Inside Length of Slowpoke Canister	39.44-0.38-0.56=38.5
Total Length of Four Canister Insert	9.5×4=38
Gap between Insert and Canister Lid	38.5-38=0.5
Length of Canister Insert	9.25
Length of Fuel	8.66
Gap between Insert and Fuel	9.25-8.66=0.59
Maximum Gap between the Fuel and Lid	.05+0.59=1.09

Table 2.7.7-4 Summary of Gap and Relative Velocity between SLOWPOKE Fuel Canister Assembly Components

Components		Gap, in	Cumulative Gap	Relative Velocity, in/s
Lid Handle	Cask Lid	0.71	0.71	160
Canister Insert	Lid Housing	0.50	0.71+0.5=1.21	210
Fuel	Lid Housing	0.59	1.21+0.59=1.8	256

Contact Force on the SLOWPOKE Fuel Canister Assembly Components

The force on the components due to end drop is calculated in two cases. In the first case, the force (F1) on the canister weldment and the canister lid outer edge is calculated and in the second case the total force (F2) on the canister lid handle is calculated. The force on the lid housing is the difference of the above two forces (F3 = F2-F1). In both the cases, the cask weight is taken as 48,000 lbs (Table 2.2.1-1).

In the first case, the force developed in the canister weldment is determined using the model shown in Figure 2.7.7-2. This model contains the mass for the canister weldment assembly and cask body. The initial velocity for the contact associated with the 0.71 inch gap between the canister and cask lid is 160 in/sec from Table 2.7.7-4. The force acting on the weldment is obtained from the transient solution of two lumped mass and a contact element with stiffness K modeled between the lump masses for the canister weldment and the cask lid. The contact element is used to simulate the compression loading of the weldment onto the cask lid. Since the initial velocity of the lump mass for the canister is specified, the contact element is also specified to have a zero gap. The stiffness K for the contact element is 670,987 lb/in. The resulting force time history developed during the contact of the weldment onto the cask is shown in Figure 2.7.7-3 and shows that the peak force is F1=23,910 pounds.

In the second case, the total contact force developed on the lid handle due to the canister weldment, canister insert, and fuel is determined using the model shown in Figure 2.7.7-4 This model contains the lumped mass for the canister weldment, insert, fuel, and cask. This model and the transient solution procedure are similar to the model in the first case except that it has additional components. The initial velocity of the canister contacting the lid (associated with the 0.71 inch gap between the canister and cask lid) is 160 in/sec. Initial gap and velocity for the insert is 1.21 inches and 210 in/sec and for fuel the initial gap and velocity are 1.8 inches and 256 in/sec respectively (Table 2.7.7-4). Gap elements (compression only elements) are modeled between the lumped masses representing the canister components and cask. The stiffness of the canister weldment, insert, and fuel tubes are 670,987 lb/in, 447,517 lb/in, and 80,464 lb/in, respectively. The resulting force time history on the canister lid developed during the contact of the all the canister components on the cask is shown in Figure 2.7.7-5 and shows a peak force of F2 = 30,342 pounds. This peak force is used in evaluating the canister lid handle.

During the end drop, the insert and fuel contact the canister housing, which is mounted on the canister lid. The force on the housing is the difference of the peak forces from the above two cases. Therefore the force on the housing is F3 = F2 - F1 = 30342 – 23910 = 6432 lb. Table 2.7.7.-5 gives the summary of the contact force on the canister components for the 30-foot end drop.

Table 2.7.7-5 End Drop Force on the SLOWPOKE Fuel Canister Assembly Components

Component	Contact Force, lbs
Force on canister lid due to canister weldment	F1 = 23910
Force on handle due to canister weldment, insert and fuel	F2 = 30342
Force on canister housing due to insert and fuel	F3 = F2-F1 = 6432

Figure 2.7.7-2 Mass and Gap Element ANSYS Model of the Cask and Canister Weldment

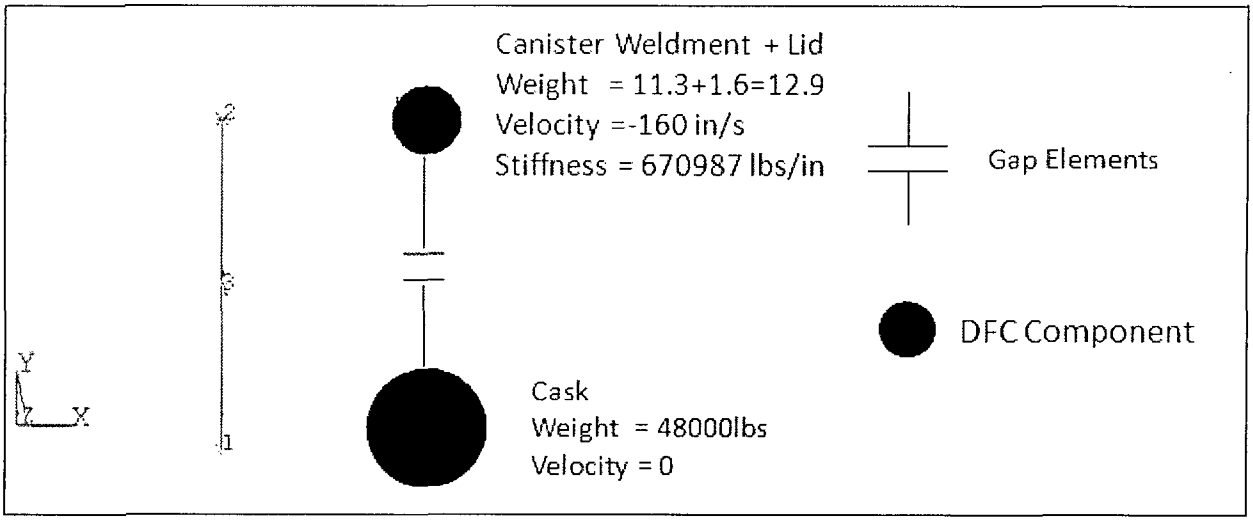


Figure 2.7.7-3 Force on the Canister Weldment for the 30-Foot End Drop

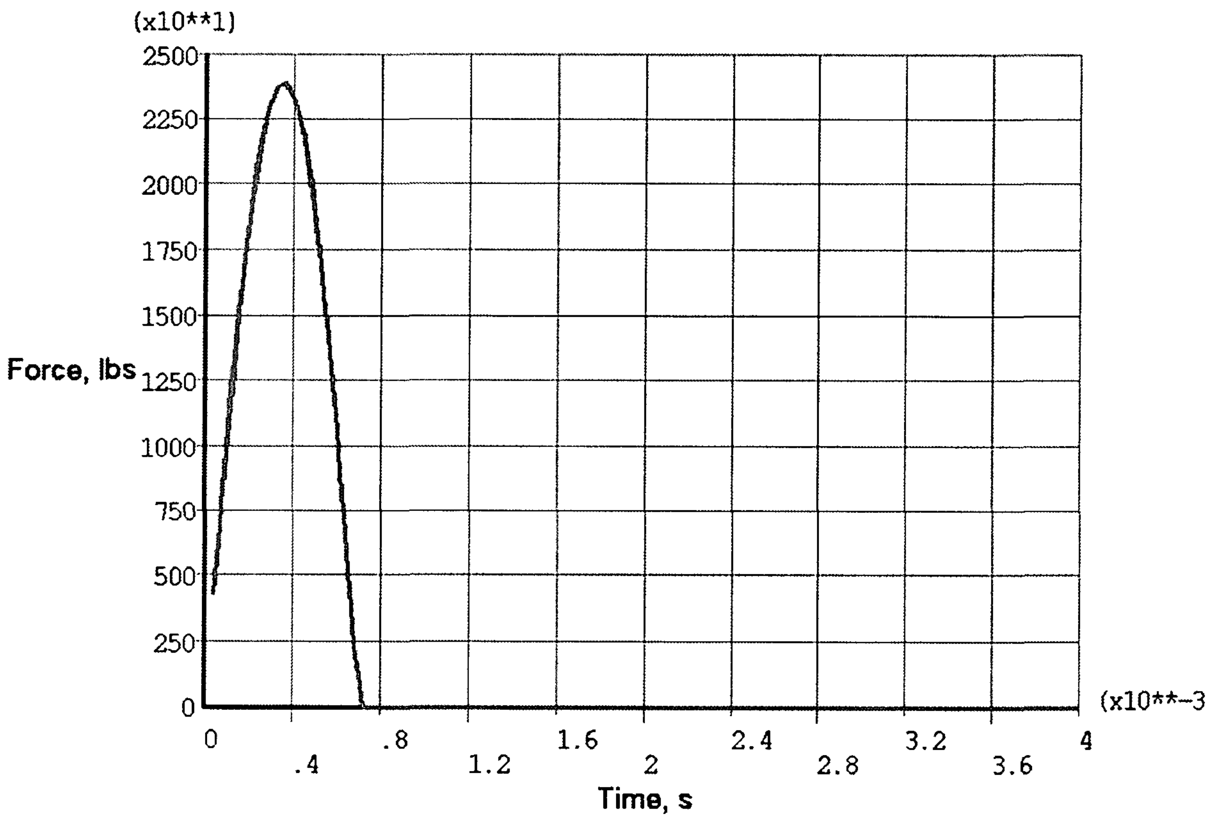


Figure 2.7.7-4 Mass and Gap Element ANSYS Model of the Cask, Canister Weldment, Insert and Fuel

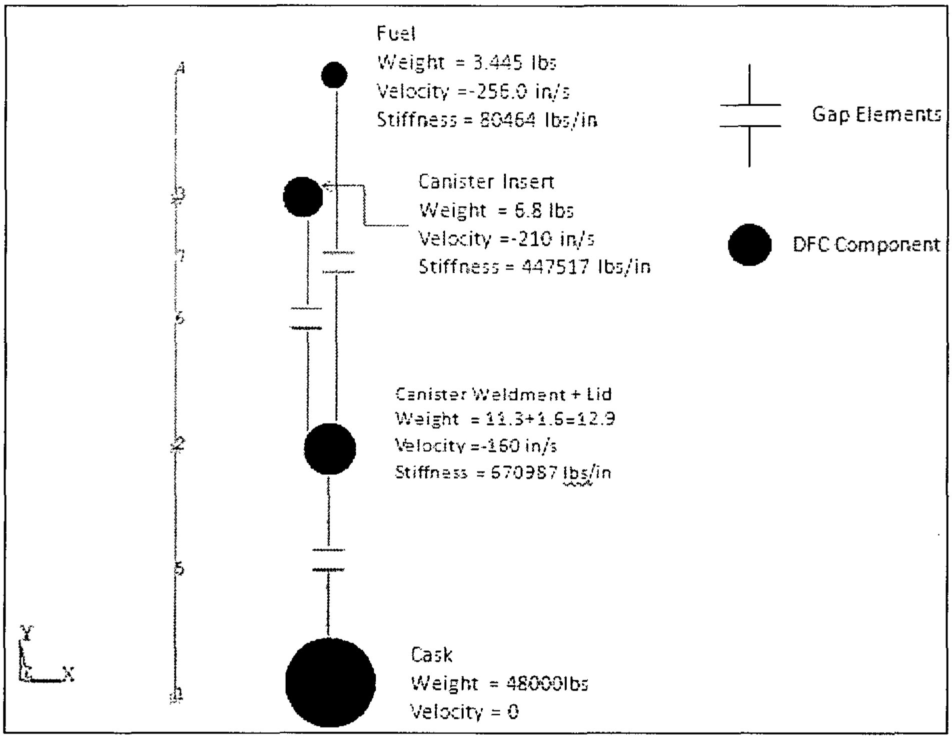
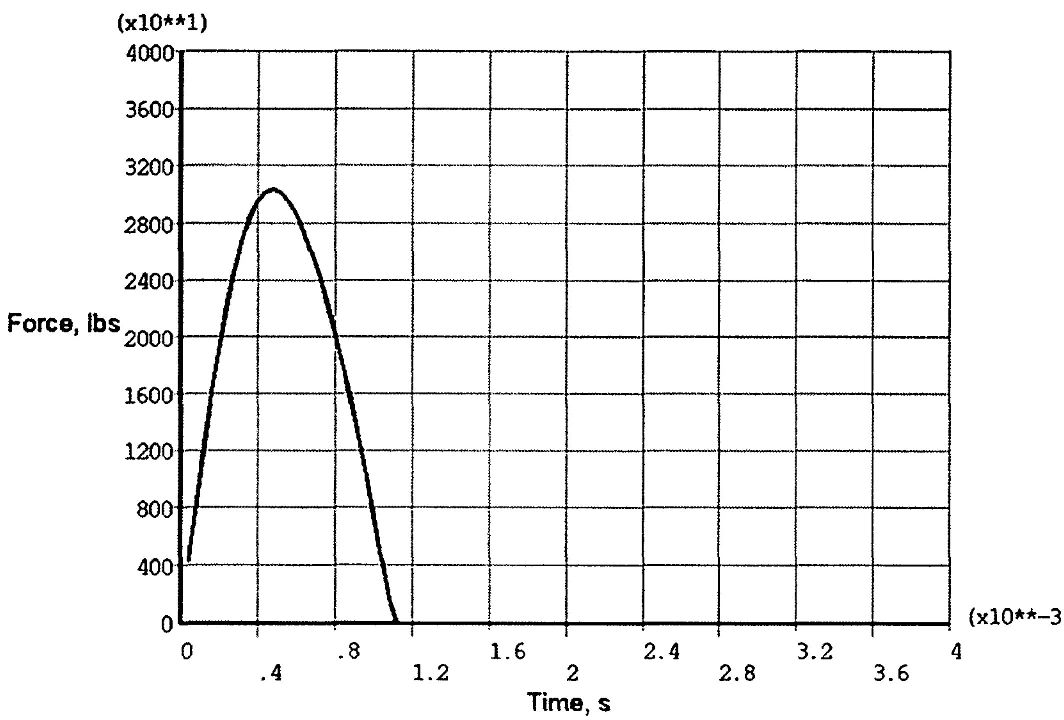


Figure 2.7.7-5 Force on the Canister Lid Handle for the 30-Foot End Drop



Canister Weldment

Canister weldment is a square tube constructed of four quarter inch thick aluminum plates welded along the length of the plate. The canister weldment is evaluated for longitudinal (end) and lateral (side) drops act for accident condition of transport. The area of the canister weldment is calculated near the lid where the cross sectional area is minimum. The detailed analysis of the canister weldment is given below.

Evaluation of canister weldment for end drop:

The compressive stress on the canister weldment is calculated as:

$$\sigma_c = \frac{F1}{2(A1+A2)} = \frac{23910}{2(0.5225+0.4975)} = 11{,}721\ psi$$

where:

F1 = 23910 lbs, force on the canister weldment for 30-foot drop

A1 = (2.8-0.71)×0.25=0.5225 in^2, Cross sectional area of Side wall-A

A2 = (2.8-0.81)×0.25=0.4975 in^2, Cross sectional area of Side wall-B

Margin of safety is:

$$MS = \frac{Membrane.\ Allowable}{Compressive\ Stress} - 1 = \frac{25200}{11721} - 1 = +1.15$$

Buckling Analysis of canister weldment:

Column slenderness ratio separating elastic and inelastic buckling:

$$C_C = \sqrt{\frac{2\pi^2 E}{F_y}} = \sqrt{\frac{2\pi^2 \times 9.54\times10^6}{3.12\times10^4}} = 77.65$$

Slenderness ratio Kl/r:

$$kl/r = \frac{1.0\times l}{\sqrt{I/A}} = \frac{1.0\times 39.44}{\sqrt{3.66/2.04}} = 29.42$$

where:

E = 9.54×10^6 psi, Elastic modulus of aluminum at 200°F

F_y = 3.12×10^4 psi, Yield strength of aluminum at 200°F

l = 39.44 in, Length of canister weldment wall

I = $2\times\left(\frac{2.8\times0.25^3}{12} + 2.8\times0.25\times\left(2.8/2\right)^2\right) + 2\times\left(\frac{0.25\times2.8^3}{12}\right) = 3.66\ in^4$, Moment of inertia

A = $2\times(0.523+0.498) = 2.04\ in^2$, Cross sectional area

If $Kl/r < C_C$, then

Allowable compressive stress on the tube:

$$F_a = \frac{\left[1-\frac{(kl/r)^2}{2C_C^2}\right]F_y}{\frac{5}{3}+\frac{3(kl/r)}{8C_C}-\frac{(kl/r)^3}{8C_C^3}} = \frac{\left[1-\frac{(29.42)^2}{2\times 77.65^2}\right]3.12\times 10^4}{\frac{5}{3}+\frac{3(29.42)}{8\times 77.65}-\frac{(29.42)^3}{8\times 77.65^3}} = 16{,}089$$

Margin of safety is:

$$MS = \frac{Allowable\,Comp.\,Stress}{Maximum\,Comp.\,Stress} - 1 = \frac{16089}{11721} - 1 = +0.37$$

For the 30-foot side drop, the weldment wall is assumed to be fixed on all four edges.

The bending stress on the canister weldment wall is calculated as (Roark's, 7th Edition, Table 11.4):

$$\sigma_b = \frac{-\beta_1 q b^2}{t^2} = \frac{-(0.5\times 1.535\times 2.8^2)}{0.25^2} = -96\ psi$$

where:

β_1 = 0.5, Parameter for a/b = ∞

q = $\frac{W}{4}\times G_{S1}\times\frac{1}{A} = \frac{11.3}{4}\times 60\times\frac{1}{39.44\times 2.8} = 1.535\ psi$, Uniformly distributed load

W = 11.3 lbs, Weight of the canister weldment

G_{S1} = 60 g, 30-foot side drop 'g' load

A = $39.44\times 2.8 = 110.43\text{in}^2$, Area of the weldment wall

Margin of safety is:

$$MS = \frac{Bending\ Allowable}{Compressive\ Stress} - 1 = \frac{36710}{96} - 1 = +Large$$

Canister Lid

The canister lid is a one-inch thick aluminum plate which closes the top end of canister weldment. The lid assembly includes lid plate, handle, housing, latch, and handle cap screw. In the 30-foot end drop, the canister lid experiences bending due to the canister insert inertial loading and shearing or puncturing from the lid handle and also by a shear load applied at the lid outer edges. The bending stress on the canister lid is evaluated using ANSYS for accident condition of transport. Figure 2.7.7-6 shows the quarter scale shell model of the canister lid with loads on each portion of the lid. In the FE model of the lid, the handle load (F2= 30,342 lbs) is applied on the area representing the handle location, the canister weldment load (F1 = 23,910 lbs) is applied on the lid plate edge and the insert and fuel load (F3 = 6,432 lbs) is applied on the

The maximum bending stress intensity on the lid is 23,402 psi. Therefore the margin of safety is +0.57 (36,710/23,402-1).

The detailed evaluation of the lid shearing by the handle and the weldment for the accident condition of transport is given in below.

During the 30-foot end drop, the handle can shear or puncture the canister lid. The shear stress on the lid due to the handle is:

$$\tau_{l-h} = \frac{F2}{A_1} = \frac{F2}{2(w_1 + t_1) \times 2} = \frac{30342}{2(1.0 + 0.65) \times 2} = 4{,}597\ psi$$

where:

F2 = 30342 lbs, Force on the canister handle for 30-foot drop

w_1 = 1.0 in, Width of the handle

t_1 = 0.65 in, Thickness of the handle

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{18355}{4597} - 1 = +2.99$$

During the 30-foot end drop, the canister weldment will apply a shear load to the outer edges of the lid. The shear stress on the lid due to canister weldment is:

$$\tau_{l-w} = \frac{F1}{A_2} = \frac{F1}{4w_2 t_2} = \frac{23910}{(4 \times 2.75 \times (1 - 0.38))} = 3{,}506\ psi$$

where:

F1 = 23910 lbs, Force on the canister weldment for 30-foot drop

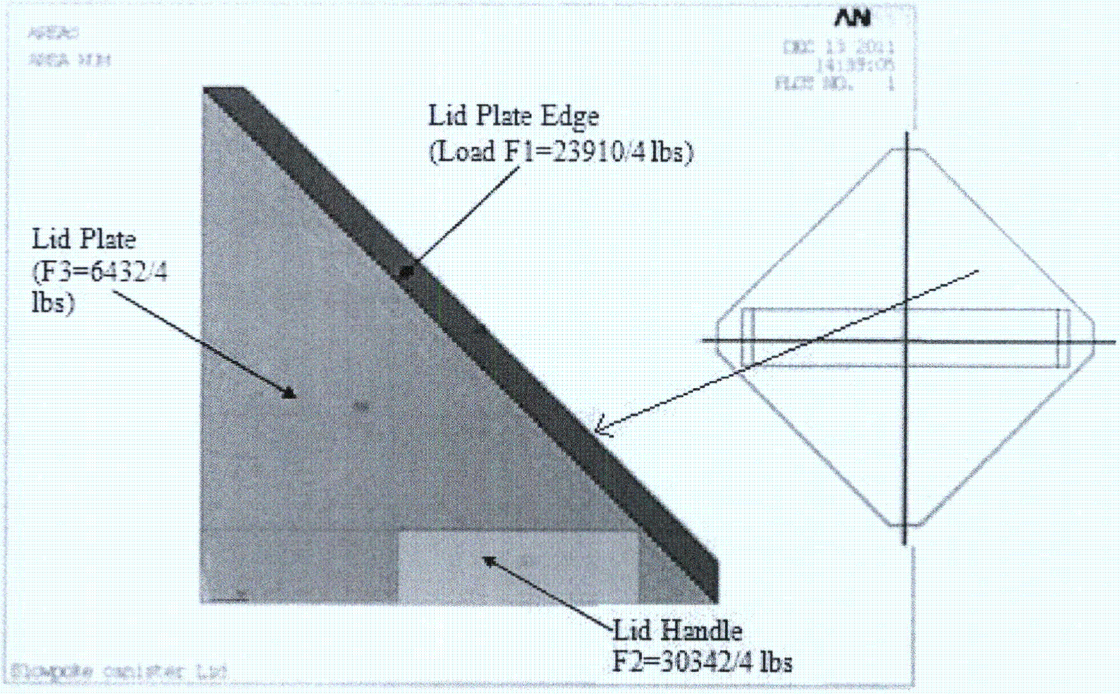
w_2 = 2.75 in, Width of the lid at the canister weldment seating

t_2 = 1-0.38=0.62 in, Thickness of the lid at the canister weldment seating

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{18355}{3506} - 1 = +4.24$$

Figure 2.7.7-6 Quarter Symmetry Model of the Lid



Lid Handle

The aluminum lid handle is 0.65" thick and 2.5" tall and is fastened to the lid plate using two socket head cap screws. For the end drop, the entire inertial load is transferred through the lid handle. The total force on the handle is F2 = 30,342 lbs. The handle evaluation for the accident condition of transport is given below.

For 30-foot end drop:

The membrane or compressive stress on the lid handle is calculated as:

$$\sigma_{mh} = \frac{F2}{A_3} = \frac{F2}{2 \times w_3 t_3} = \frac{30342}{2 \times 1.0 \times 0.65} = 23{,}340\ psi$$

where:

F2 = 30342 lbs, Force on the canister handle for 30-foot drop
w_3 = 1.0 in, Width of the lid handle
t_3 = 0.65 in, Thickness of the lid handle

Margin of safety is:

$$MS = \frac{Membrane\ Allowable}{Membrane\ Stress} - 1 = \frac{25200}{23340} - 1 = +0.08$$

Housing

The housing supports the latch and the spring mechanism and these three components keep the lid securely on the canister weldment. Figure 2.7.7-7 shows the housing with dimensions and critical locations for evaluation. On the housing, compressive or membrane stresses are evaluated at Location MN, the stress intensity is evaluated at Location A.

For the 30-foot end drop, the canister insert along with fuel contacts the housing at Location RS (Figure 2.7.7-7). The load on the housing is F3 = 6,432 lbs.

The membrane stress on the housing Location MN (Figure 2.7.7-7) is calculated as:

$$\sigma_{mHo} = \frac{F3}{A_5} = \frac{F3}{2(w_5 - d)t_5} = \frac{6432}{2(0.75 - 0.38)0.75} = 11{,}589\ psi$$

where:

F3 = 6432 lbs, Load on the canister housing for 30-foot end drop
W_5-d = (0.75-0.38) = 0.37 in, Width of the housing at Location MN
t_5 = 0.75 in, Thickness of the housing at Location MN

Margin of safety is:

$$MS = \frac{Membrane\ Allowable}{Membrane\ Stress} - 1 = \frac{25200}{11589} - 1 = +1.17$$

The shearing stress on the housing Location A (Figure 2.7.7-7) is:

$$\tau_A = \frac{F3\!/\!_2 \times (R)}{A_7} = \frac{F3\!/\!_2 \times (R)}{2h_1 \times 2t} = \frac{6432\!/\!_2 \times \left(0.38\!/\!_{0.75}\right)}{2 \times 0.25 \times 2 \times 0.75} = 2{,}173\ psi$$

where:

F3/2 = 6432/2 = 3216 lbs, Load on the each canister housing for 30-foot end drop

R = Ratio of load on the curved section

$$= \frac{\text{Thickness of curved section (Length AA)}}{\text{Thickness of housing (Length RS)}} = \frac{0.38}{0.75} = +0.51$$

(Note: R = 0.51, Half of the load on the housing is conservatively applied on the curved section of the housing)

h_1 = 0.25 in, Height of the housing at Location A

t = 0.75 in, Thickness of the housing at Location A

Margin of safety is:

$$MS = \frac{Shear\ Allowable}{Shear\ Stress} - 1 = \frac{18355}{2173} - 1 = +7.45$$

The bending stress on the housing Location A is (assumed as beam fixed at two ends):

$$\sigma_A = M\frac{Y}{I} = \frac{W_a L^2}{12}\frac{Y}{I} = \frac{F3\!/\!_2 \times R}{L}\frac{L^2}{12}\frac{Y}{I}$$
$$= \frac{6432\!/\!_2 \left(0.38\!/\!_{0.75}\right)}{0.38}\frac{0.38^2}{12}\frac{0.25\!/\!_2}{9.77 \times 10^{-4}} = 6{,}605\ psi$$

where:

I = $(bh_1^3/12) = (0.75 \times 0.25^3/12) = 9.77 \times 10^{-4}\ in^4$, Moment of inertia at Location A

Y = 0.25/2 in, Height of the housing from mid point at Location A

L = 0.38 in, Length of the curved section of the housing (Length AA)

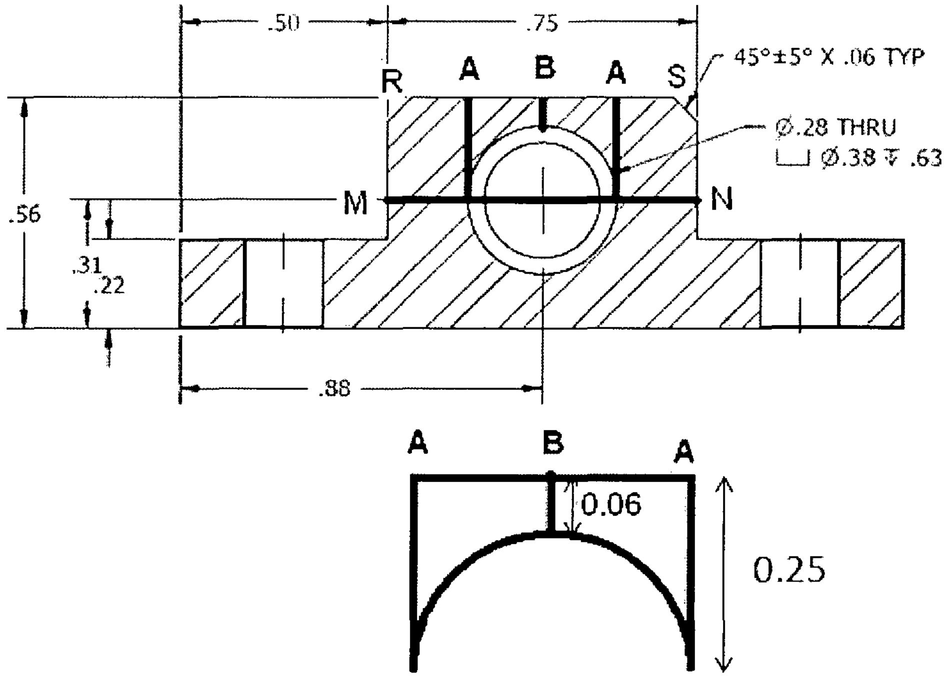
The stress intensity on the housing Location A is:

$$\sigma_S = \sqrt{\sigma_A^2 + 4\tau_A^2} = \sqrt{6605^2 + 4 \times 2173^2} = 7{,}906\ psi$$

Margin of safety is:

$$MS = \frac{Bending\ \ Allowable}{Stress\ Intensity} - 1 = \frac{36710}{7906} - 1 = +3.64$$

Figure 2.7.7-7 SLOWPOKE Fuel Canister Assembly Housing



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3 THERMAL EVALUATION

3.1 Discussion

This chapter summarizes the thermal analyses, which are performed to demonstrate fulfillment of the thermal capability requirements established in 10 CFR 71.

The NAC-LWT cask is designed to safely contain irradiated nuclear fuel and other radioactive materials under a variety of normal transport conditions (as described in 10 CFR 71.71) and accident conditions (as described in 10 CFR 71.73). In order to verify the adequacy of the design, detailed analyses of a reference design PWR shipment are performed considering extreme normal transport and hypothetical accident conditions. The NAC-LWT cask is designed to transport one intact PWR fuel assembly; up to 2 intact BWR fuel assemblies; up to 25 individual PWR or BWR rods (including up to 14 fuel rods classified as damaged); up to 16 PWR MOX fuel rods (or a combination of PWR MOX and UO_2 PWR fuel rods); up to 42 MTR and DIDO fuel elements; up to 140 TRIGA fuel elements or 560 TRIGA fuel rod clusters; up to 300 TPBARs (of which two can be prefailed), up to 55 segmented TPBARs; or up to 700 PULSTAR fuel elements (intact or damaged); and metallic fuel. The PULSTAR fuel will be loaded in the 28 MTR basket and consist of intact fuel assemblies, intact fuel rods loaded in fuel rod inserts or fuel cans, or intact or damaged fuel and nonfuel components or fuel assemblies loaded in fuel cans. High burnup PWR/BWR fuel rods may be placed in a rod holder or in a fuel assembly lattice. Damaged PWR/BWR fuel rods must be placed in a rod holder.

High burnup PWR/BWR/PWR MOX fuel rods are placed in a rod holder (a rod holder is the term generally used in this chapter to describe a PWR/BWR Rod Transport Canister with a 4 × 4 or a 5 × 5 insert as presented on the drawings provided in Section 1.4). The high burnup PWR and BWR rods may also be placed in a fuel assembly lattice. Damaged PWR/BWR fuel rods must be placed in a rod holder. The 16 PWR MOX fuel rods are required to be placed in a rod holder with a 5 × 5 insert. Along with the maximum 16 PWR MOX rod contents (or combination of PWR MOX and UO_2 PWR fuel rods), the remaining tubes may be loaded with burnable poison rods or other intact components with negligible heat loads (total additional heat load of less than 10 watts). Up to four (4) SLOWPOKE fuel canisters each containing up to 100 SLOWPOKE fuel rods with a maximum decay heat load of 0.625 Watts/canister can be loaded in a MTR basket module. For SLOWPOKE fuel contents, only the top and upper intermediate MTR-28 modules may be loaded. The empty lower intermediate and bottom basket modules are installed as axial spacers. The total package decay heat for SLOWPOKE fuel contents is 5 Watts.

An intact PWR fuel assembly with a maximum decay heat load of 2.5 kW is used in a majority of the thermal analyses. The failed fuel basket analysis in Section 3.6 uses a decay heat load of 30 Watts. The 42 MTR fuel assembly basket in Section 3.4.1.3 uses a decay heat load of 1.26

kW. A decay heat load of 1.05 kW is conservatively used for the TRIGA fuel basket analysis and a decay heat load of 0.693 kW is used for the TPBAR basket analysis. The maximum heat load for the PULSTAR fuel is 0.840 kW per cask. The maximum heat load for the maximum number of 16 PWR MOX fuel rods is 2.3 kW per cask (143 W per PWR MOX rod). As long as the decay heat load is within the design limit of 2.5 kW, any of the fuel types and other radioactive material that the NAC-LWT cask is analyzed to transport are bounded by the cask body thermal analyses of the design basis PWR assembly.

The primary heat rejection design criteria for the NAC-LWT cask are that:

1. Components important to safety shall not be subjected to temperatures outside their safe operating ranges.

2. Thermally induced stresses in the cask containment (in combination with pressure and various load condition stresses) shall not cause degradation of the cask containment capability.

The first criterion is fulfilled by thermal analysis results, which show that components important to safety are maintained within their safe operating ranges. In the event that the temperatures of the components important to safety fall outside the safe operating ranges, it is assumed that the component has failed. Temperatures of components important to safety may not fall outside the safe operating range during normal transport conditions. There are three important safety components that are subject to this thermal criterion – the tetrafluoroethylene (TFE), Viton®, and metallic O-ring seals; the lead gamma shield; and the 56 % ethylene glycol and water neutron shield.

An additional thermal consideration is associated with the liquid neutron shield tank – the reduction in neutron shielding capability caused by thermal contraction. An expansion tank is provided to ensure that the neutron shield tank remains full despite worst case contraction of the liquid in the tank during cooling. The method used by the expansion tank to keep the neutron shield tank full is described in Section 2.6.7.7.1.

The second criterion is fulfilled by the structural analysis of Chapter 2, which shows that combined load stresses (including thermally induced stresses) are less than the limits stated in Section 2.1.2.

The thermal analyses were performed for a 0.25-inch thick neutron shield tank shell, while the actual fabricated thickness is only 0.24 inches (6mm). The shell thickness difference of 0.01 inches equates to only a 0.009°F ΔT; therefore, the analyses reported in this chapter are valid.

$$\Delta T_a = (671°F - 387°F)\frac{0.1kW \times 1.15}{2.1kW \times 1.22} = 12.75°F$$

$$\Delta T_b = (228°F - 222°F)\frac{0.1kW}{0.69kW} = 0.87°F$$

$$\Delta T_{total} = \Delta T_a + \Delta T_b = 14°F$$

where:

ΔT_a: is the temperature difference between the maximum contents temperature and PWR insert for the TPBARs in the PWR/BWR Rod Transport Canister configuration.

ΔT_b: is the temperature difference across the TPBAR basket to the inner surface of the inner shell for the TPBARs in the PWR/BWR Rod Transport Canister configuration.

ΔT_{total}: is the temperature difference from the inner surface of the inner shell to the maximum contents temperature.

The maximum contents temperature for the TPBARs in the PWR/BWR Rod Transport Canister configuration is calculated as follows for the normal conditions of transport.

$$T_{max\,contents} = Tis_{max} + \Delta T_{total} = 222°F + 14°F = 236°F$$

where:

Tis_{max}: is the assumed maximum inner shell temperature for the TPBARs in the PWR/BWR Rod Transport Canister configuration

The maximum contents temperature for the NAC-LWT transport cask loaded with TPBAR fuel, the PWR/BWR Rod Transport Canisters, the PWR insert and the TPBAR basket is lower than the maximum temperature for the 300 TPBARs in the TPBAR basket analysis of Section 3.4.1.12 (236°F versus 290°F). Therefore, the component temperatures for the 25 TPBARs in the Rod Transport Canister configuration are bounded by the results of Section 3.4.1.12, which are summarized in Table 3.4-16.

3.4.1.18 SLOWPOKE Fuel Analytical Model

Heat transfer analysis of the NAC-LWT containing SLOWPOKE fuel is performed using a two-dimensional planar finite element model and the general purpose ANSYS computer code. The model represents the entire cask and uses the thermal conductivity of helium to represent the contents inside each basket slot and cask cavity gas. This model is identical to the model of MTR fuel (Section 3.4.1.3) for Condition 1, except for the contents inside the basket slots and the cask cavity gas are modeled as helium. The cavity of air shown in Figure 3.4-4 for MTR fuel

is changed to helium. The heat generation rate is applied to the two outer slots representing cavity air in Figure 3.4-4. The three center slots (containing 70W, 120W, and 20W fuel elements in Figure 3.4-4) are modeled as helium without heat generation applied as the slots are blocked for SLOWPOKE fuel contents. This modified model is used to determine the maximum component temperatures throughout the cask. The NAC-LWT is supported in an ISO container with solar insolance applied on the surface of the ISO container, and the NAC-LWT is considered to be insulated from the environment (only for normal conditions of transport steady state conditions). The gas inside the ISO container is air. The cavity of the NAC-LWT is backfilled with helium as required by operational procedures.

This configuration consists of up to four SLOWPOKE canisters with a design basis heat load of 0.625 Watts per canister. Total decay heat for the SLOWPOKE fuel contents is 5.0 Watts and is modeled as uniformly generated within the homogenized fuel/helium regions of the outer four (4) slots for the top and upper intermediate basket modules. The detailed fuel is not modeled while the contents are conservatively modeled as helium that has low thermal conductivities.

The cask model for SLOWPOKE fuel thermal analysis is identical to the one used for the MTR fuel as described in Section 3.4.1.3, except that helium is used for the slot contents and the cask cavity gas. For details of the model, see Section 3.4.1.3.

3.4.1.18.1 SLOWPOKE Fuel Heat Transfer Analyses Results

The thermal analysis is performed to demonstrate that the temperature of the SLOWPOKE fuel is maintained within acceptable limits. Due to the low heat load, maximum temperature of the loaded cask occurs at the top of the outer surface of the ISO container. Maximum temperatures for package components with the NAC-LWT configured for SLOWPOKE fuel are summarized in Table 3.4-27. The reported temperatures are lower at all locations than the corresponding temperatures for the design basis PWR fuel presented in Table 3.4-2.

3.4.2 Maximum Temperatures

Using the models described, temperatures for the cask body and fuel rod cladding are determined for maximum normal operation conditions (2.5 kW decay heat load, 130°F ambient temperature, still air, full insolation). The maximum cask component and fuel rod cladding temperatures for PWR fuel (2.5 kW) are listed in Table 3.4-2. Not all of the cask components are explicitly modeled; their temperatures are obtained by evaluating the analytical model at the component location. Maximum normal operating temperatures for the 1.26 kW MTR fuel and the 1.05 kW TRIGA fuel configurations are shown in Table 3.4-6 and Table 3.4-8, respectively. Maximum normal operating temperatures for high burnup PWR and BWR fuel rods in a rod holder are shown in Table 3.4-10. The maximum component temperatures for DIDO fuel and General Atomics IFM for normal conditions of transport are shown in Table 3.4-12 and Table 3.4-13, respectively. The maximum component temperatures for 25 high burnup PWR and BWR fuel rods in a fuel assembly lattice are shown in Table 3.4-14. The maximum component temperatures for high burnup PWR or BWR fuel with up to 14 damaged fuel rods in a rod holder are shown in Table 3.4-15. Maximum operating component temperatures for the NAC-LWT containing TPBARs are shown in Table 3.4-16. The maximum operating temperatures for the PULSTAR fuel contents in the MTR basket are shown in Table 3.4-17. The maximum component temperatures for the NAC-LWT containing MOATA plate fuel and Mark III spiral fuel are presented in Table 3.4-22. Section 3.4.1.16 confirms that the temperatures corresponding to the ANSTO-DIDO combination basket are bounded by the temperatures of the ANSTO fuel and the DIDO fuel presented in Table 3.4-22 and Table 3.4-12, respectively. The maximum component temperatures for the NAC-LWT containing SLOWPOKE fuel are presented in Table 3.4-27.

3.4.3 Minimum Temperatures

As stated in Section 3.4.1, the minimum temperatures in the cask occur with a 0.0 kW decay heat load and the minimum ambient conditions. Under these conditions, a uniform temperature of -40°F will exist in the cask. The maximum thermal stresses in the cask, during normal operations conditions, occur when the design basis decay heat load of 2.5 kW exists in the cask along with the minimum ambient conditions (-40°F ambient temperature and no insolance). The cask component and fuel rod clad temperatures for the 2.5 kW decay heat load with minimum ambient conditions are listed in Table 3.4-3.

3.4.4 Maximum Internal Pressures

3.4.4.1 Maximum Internal Pressure for Design Basis Fuel in Normal Conditions

The NAC-LWT cask is filled to one atmosphere (14.7 psia) upon loading. It is necessary to evaluate the internal pressure of the cask after thermal equilibrium has been attained. Assuming a maximum normal fuel cladding temperature of 472°F (932°R) from Table 3.4-2, 3 percent fuel rods rupture, and 30 percent of the fission gas and 100 percent of the rod backfill gas escape from the ruptured fuel rods, the cask internal pressure is calculated. Table 3.4-4 reports the fission gas inventories for the design basis PWR fuel assembly. Table 5.1-2 reports the design parameters of the design basis PWR fuel. Using information from Table 5.1-2, the void volume of a single fuel rod is calculated as 2.43 in^3 (39.82 cm^3) by subtracting the volume of the fuel pellets from the volume of an empty fuel rod (the plenum spring volume is disregarded). The total fuel assembly void volume is calculated as 495.16 in^3 (8,123.28 cm^3) by multiplying the single fuel rod volume by 204, the number of fuel rods in the fuel assembly. The total fuel assembly void volume, the fission gas inventory information in Table 3.4-4, and the maximum normal transport temperature (472°F) are applied to the ideal gas law (PV = nRT) to obtain the pressure in the unruptured fuel rods due to the fission gases. This fission gas pressure, 1,771.5 psia, is also reported in Table 3.4-4, based on 100% availability of fission gases, later adjusted to 30%. The releasable fission gas pressure and rod backfill pressure (assumed 565 psia) are summed to obtain the total fuel rod pressure.

The cask pressure is obtained using Dalton's Law of Partial Pressures:

$$P = P_A + P_B$$

where:

P = total pressure

P_A = partial pressure of gas A (cask cavity helium gas backfill)

P_B = partial pressure of gas B (fuel rod backfill and fission gas)

The reported cask and fuel rod backfill pressures are at standard temperature (72°F) and must be corrected to the normal transport temperature. Given that the internal volumes of the NAC-LWT Cask and the fuel rods remain constant, the resultant pressure is proportional to the temperature change according to the ideal gas law:

$$P_2 = P_1\left(\frac{T_2}{T_1}\right)$$

where:

P_1 = 14.7 psia (cask backfill pressure)

T_1 = 72°F (532°R) (cask backfill temperature)

T_2 = 472°F (932°R) (maximum normal operating condition cavity gas temperature)

Thus, the cask backfill pressure at normal operating temperature equals:

$$P_2 = 14.7\text{ psia}\left(\frac{932°R}{532°R}\right)$$

$$P_2 = 25.8\text{ psia}$$

For the fuel rod backfill pressure at normal operating temperature:

P_1 = 565 psia (fuel rod backfill pressure)

T_1 = 72°F (532°R) (fuel rod backfill temperature)

T_2 = 472°F (932°R) (maximum normal operating condition cavity gas temperature)

and:

$$P_2 = 565\text{ psia}\left(\frac{932°R}{532°R}\right)$$

$$P_2 = 989.8\text{ psia}$$

The partial pressure of the cask backfill distributed over the cask free volume (including 3% failed rods) is calculated by:

$$P_{cask\ backfill} = P_{initial}\left(\frac{V_{cask}}{V_{total}}\right)$$

where:

$P_{initial}$ = 25.8 psia (temperature adjusted cask backfill pressure)

V_{cask} = 5.196 ft^3 (147,134 cm^3)

$V_{rod\ void}$ = 244 cm^3 (volume of 3% failed fuel rods)

V_{total} = V_{cask} + $V_{rod\ void}$

V_{total} = 147,378 cm^3

Thus, the cask backfill partial pressure at normal operating temperature, including the volume of failed fuel rods equals:

$$P_{\text{cask backfill}} = 25.8\ \text{psia}\left(\frac{147{,}134\text{cm}^3}{147{,}378\text{cm}^3}\right)$$

$$P_{\text{cask backfill}} = 25.8\ \text{psia}$$

The partial pressure of the failed fuel rod gases in the cask cavity is calculated by:

$$P_{\text{fuelrods}} = P_{\text{initial}}\left(\frac{V_{\text{rodvoid}}}{V_{\text{total}}}\right)$$

where:

P_{initial} = 1,521.3 psia (fission gas pressure (0.30 x 1,771.5 psia) plus rod backfill pressure (989.8 psia))

$V_{\text{rod void}}$ = 244 cm^3

Vtotal = 147,378 cm3

Thus, the failed fuel rod partial pressure at normal operating temperature, including fission gases and the volume of cask cavity void equals:

$$P_{\textit{fuel rods}} = 1{,}521.3\ psia\left(\frac{243.7cm^3}{147{,}378cm^3}\right)$$

$$P_{\text{fuelrods}} = 2.5\ \text{psia}$$

Summing the two partial pressures yields the total cask pressure.

$$P_{\text{Total}} = P_{\text{cask backfill}} + P_{\text{fuelrods}}$$

$$P_{\text{Total}} = 25.8\ \text{psia} + 2.5\ \text{psia}$$

$$P_{\text{Total}} = 28.3\ \text{psia}$$

3.4.4.2 High Burnup Fuel Rod Canister Maximum Normal Conditions Internal Pressure

The high burnup fuel sealed canister is filled to one atmosphere (14.7 psia) upon loading. The canister internal pressure is calculated assuming that the average helium backfill gas temperature is 600°F (1060 R) and that 3 percent of the fuel rods fail in normal conditions of transport. The temperature of the canister gas is selected to conservatively bound the temperatures given in Table 3.4-10, Table 3.4-14 and Table 3.4-15. On failure, the fuel rods are assumed to release

30% of the fission gas and 100% of the rod backfill gas. To bound both the PWR and BWR analysis, the fuel type with the highest fission source, on a per rod basis, and smallest free gas volume inside the sealed canister is selected. This fuel type is the Exxon 7 × 7 BWR fuel. The fission gas inventory for this fuel is shown in Table 3.4-11, which reports the fission gas inventory for the assembly, and on a per rod basis. The design parameters for the Exxon 7 × 7 fuel rod are:

Parameter	Value
Number of Rods	49
Rod Diameter (in)	0.570
Clad Thickness (in)	0.036
Pellet Diameter (in)	0.4900
Active Fuel Length (in)	150
Rod Length (in)	170

From the values shown, the void volume of a single fuel rod is calculated as 4.82 in^3 (78.99 cm^3) by subtracting the volume of the fuel pellets from the volume of an empty fuel rod (the plenum spring volume is disregarded). For the analysis, 3% of 25 rods is taken to fail, which is 0.75 rods. Conservatively, the number of failed rods is defined as one, which is equal to a 4% fuel rod failure. The equivalent void volume is then equal to one rod, or 4.82 in^3. The fission gas inventory, provided in Table 3.4-11, and the maximum normal transport temperature (600°F) are applied to the ideal gas law (PV = nRT) to obtain the pressure in the unruptured fuel rods due to the fission gases. This fission gas pressure, 4,251 psia (Table 3.4-11), is based on 100% availability of fission gases, which is adjusted to account for the 30 percent release of the fission gas. The releasable fission gas and rod backfill pressures are summed to obtain the total fuel rod pressure.

The ideal gas law is used to analyze the effects of pressure, temperature, volume, and gas inside the cask the ideal gas law states:

$$pV = nRT$$

where:

p = pressure (atm)

V = volume (liters)

n = gram-moles of material

R = gas constant (0.0831 atm-liters/K g-mole)

T = temperature (K)

After the rods rupture, the resultant internal cask pressure is impacted by three sources: the 1-atm inert gas backfill of the canister, the fission product gas escaping from the fuel rods, and the fuel rod inert gas backfill escaping from the ruptured fuel rods. To calculate the resultant internal cask pressure after the fuel rods rupture, partial pressures are calculated using Dalton's law:

$$P = P_a + P_b$$

where:

P = total pressure

P_a = partial pressure of gas A

P_b = partial pressure of gas B

The void volume of the fuel rod is simply the volume contained within the cladding less the fuel volume. The rod is modeled as a cylinder with a 0.570-in outside diameter, a 0.036-in. wall thickness, and a 150-in. active fuel length. The volume of the plenum spring is disregarded. The void volume, which includes the plenum volume, is 4.82 in^3 per rod.

The partial pressure of the canister is calculated by:

$$P_{canister} = P_{initial}\left(\frac{V_{canister}}{V_{total}}\right)$$

where:

$$P_{initial} = P_{atm} * \frac{T_{norm}}{T_{stand}} = 14.7 \text{ psia} * \frac{588.7 \text{ K}}{295.35 \text{ K}} = 29.3 \text{ psia}$$

Pinitial = 29.3 psia

The minimum free gas volume is calculated as:

$$V_{canister} = 2{,}800 \text{ in}^3 - \pi * r_{OD}^2 * L_{rod} * 25 \text{ rods} = 2{,}800 \text{ in}^3 - \pi * \left(\frac{0.57 \text{ in}}{2}\right)^2 * 170 \text{ in.} * 25 \text{ rods}$$

$$= 2{,}800 \text{ in}^3 - 1085 \text{ in}^3 = 1715 \text{ in}^3$$

Vcanister = 28.1 liters

$$V_{void} = 4.82 \text{ in}^3 * (2.54 \text{ cm/in})^3 * 0.001 \text{ liters/cc} = 0.079 \text{ liters}$$

$$V_{total} = V_{canister} + 25 * 0.04 * V_{void} = 28.1 \text{ liters} + 0.04 * 25 * 0.079 =\sim 28.2 \text{ liters}$$

$V_{total} = \sim 28.2$ liters

This results in a $P_{canister}$ of 29.3 psia.

Fission product gas inventories were obtained from Table 3.4-11. Using the ideal gas law, the initial pressure of each fission product gas is calculated based upon these inventories, the normal condition temperature (600°F), and the calculated void volume of the fuel (25 rods). The partial pressure of the fuel rod volume is calculated by:

$$P_{\text{fuel rods}} = P_{\text{initial}}\left(\frac{V_{\text{fuel rods}}}{V_{\text{total}}}\right)$$

where:

$$P_{\text{initial}} = 0.3 * P_{\text{fission}} + P_{\text{backfill}}$$

$$P_{\text{fission}} = 4251 \text{ psia}$$

$$P_{\text{backfill}} = P_{\text{initial}}^{\text{backfill}} * \frac{T_{\text{norm}}}{T_{\text{stand}}} = 75 \text{ psia} * \frac{588.7 \text{ K}}{295.35 \text{ K}} = 150 \text{ psia}$$

$P_{initial}$ = ~1,425 psia

$V_{\text{fuel rods}}$ = ~0.079 liters (at 4% of the total fuel rod volume)

V_{total} = 28.2 liters = $V_{canister} + 25*0.04(V_{void})$

$$P_{\text{fuel rods}} = 1425 \text{ psia}\left(\frac{0.079 \text{ liters}}{28.2 \text{ liters}}\right) = \sim 4.00 \text{ psia}$$

then:

$$P_{\text{total}} = P_{\text{canister}} + P_{\text{fuel rods}} = 29.3 \text{ psia} + \sim 4.00 \text{ psia} = \sim 33.3 \text{ psia (2.3 atm)}$$

An additional analysis was performed for BWR high burnup rods (>45 GWd/MTU) with a 56% failure fraction to envelope damaged fuel rod shipments. This evaluation is conservative since damaged rods are likely to have released their gas inventory prior to shipment.

Following the methodology used for calculating the pressure given above and the calculated canister free gas volume of 29.2 liters, the resulting internal canister pressure from a 56% failed fuel fraction is 82.3 psia (~ 5.6 atm). The calculation follows.

$P_{canister}$ = $P_{initial} * V_{canister} / V_{total}$

$P_{initial}$ = 29.3 psia

$V_{canister}$ = 28.1 liters

V_{total} = $V_{canister}$ + 14*V_{void} = (28.1 liters) + 14*(0.079 liters) = 29.2 liters

$P_{canister}$ = (29.3 psia)*(28.1 liters)/(29.2 liters) = 28.2 psia

$P_{fuelrods}$ = $P_{initial}$*$V_{fuelrods}$/V_{total}

$P_{initial}$ = 1,425 psia

$V_{fuelrods}$ = 14*V_{void} = 1.108 liters

$P_{fuelrods}$ = (1,425 psia)*(1.108 liters)/(29.2 liters) = 54.1 psia

P_{total} = $P_{canister}$ + $P_{fuelrods}$ = 28.2 psia + 54.1 psia = 82.3 psia = 5.6 atm

3.4.4.3 25-Rod Maximum Cask Cavity Internal Pressure-Normal Conditions

Following the methodology used for calculating pressure in Section 3.4.4.2, the cask free gas volume is calculated as:

$$V_{cask} = 6{,}534 \text{ in}^3 - \pi * r_{OD}^2 * L_{rod} * 25 \text{ rods} = 6{,}534 \text{ in}^3 - \pi * \left(\frac{0.57 \text{ in}}{2}\right)^2 * 170 \text{ in.} * 25 \text{ rods}$$
$$= 6{,}534 \text{ in}^3 - 1{,}085 \text{ in}^3$$
$$= 5{,}449 \text{ in}^3$$

V_{cask} = 89.32 liters

Using this free gas volume in place of $V_{canister}$ and the temperatures in Section 3.4.4.2, the cask cavity pressure resulting from a 3% fuel rod failure is 31 psia (~2.1 atm). This pressure is based on the assumption that the gases in the canister are released to the cask cavity. There are no design basis events that could result in the release of the gas in the canister to the cask cavity.

An additional analysis was performed for a bounding 25 BWR high burnup rod configuration (>45 GWd/MTU) containing up to 14 damaged rods. The damaged fuel rods are conservatively assumed to release the rod gas inventory during transport.

Following the methodology used for calculating the pressures given above and the cask free gas volume of 90.4 liters, the resulting internal cask pressure from a 56% failed fuel fraction is 46.4 psia (~ 3.2 atm). The calculation is outlined below.

P_{cask} = $P_{initial}$*V_{cask}/V_{total}

$P_{initial}$ = 29.3 psia

V_{cask} = 89.3 liters

V_{total} = V_{cask} + 14*V_{void} = (89.3 liters) + 14*(0.079 liters) = 90.4 liters

P_{cask} = (29.3 psia)*(89.3 liters)/(90.4 liters) = 28.9 psia

$P_{fuelrods}$ = $P_{initial} * V_{fuelrods} / V_{total}$

$P_{initial}$ = 1,425 psia

$V_{fuelrods}$ = $14 * V_{void}$ = 1.108 liters

$P_{fuelrods}$ = (1,425 psia)*(1.108 liters)/(90.4 liters) = 17.5 psia

P_{total} = $P_{cask} + P_{fuelrods}$ = 28.9 psia + 17.5 psia = 46.4 psia = 3.2 atm

PWR rods may contain Integral Fuel Burnable Absorbers (with rods referred to as IFBA rods). These rods may contain integral neutron absorber comprised of gadolinium, erbium, or boron. Only boron has the potential to impact pressure calculations as the ^{10}B isotope alpha decays upon neutron activation, thereby adding gas to the system. Activation of erbium and gadolinium does not produce additional gases. The effect on system pressure of the additional gas was evaluated based on a 2.4 g ^{10}B/in coating level [ORNL/TM-200/321], considering 100% conversion to gas, and conservatively applying the fixed absorber level to the full active fuel length. A normal condition failure of 3% of the rods (1 rod, rounded to 4%) increases the cask pressurizing gas quantity by less than 0.04 mole, which translates to a pressure change of 0.3 psi. When considering a conservative in-cask failure of 14 rods, the potential increase in pressure rises to 4 psi for the IFBA rods. Given the conservative cask free volume and fission gas generation (both based on significantly larger BWR rods), in combination with a conservative (rounded up) system temperature applied in the analysis, there is no significant effect on system pressure with the inclusion of IFBA rods.

3.4.4.4 Maximum Cask Cavity Internal Pressure for the General Atomics IFM

The combined heat load of the two GA IFM FHUs is 13 watts. This heat load is distributed between two separate canisters, which have a length of approximately 40 inches. As a result, the heat generation, which would result in a temperature differential between the cavity and ambient, is insignificant.

The internal pressure in the LWT cask cavity is due to the fission gas release from the TRIGA fuel or HTGR fuel pellets in conjunction with the cavity being heated by solar insolance. No credit is taken for the pressure retention capability of the FHUs. The internal pressure that may result from the 20 TRIGA fuel rods in the GA IFM is significantly enveloped by that of the 120 TRIGA fuel rods, which are authorized for transport in the NAC-LWT cask. Likewise, the fission gas release by the HTGR elements is considered to be bounded by the current design basis PWR fuel. Since the free volume for the GA IFM shipment is significantly larger than for the design basis PWR fuel assembly with the PWR basket, the pressure increase in the cavity gas

due to the GA IFM shipment is considered to be significantly bounded by the design basis condition in the current NAC-LWT cask. Therefore, the current design pressure of 50 psig for the cask cavity envelopes the cask cavity pressure for the GA IFM contents.

3.4.4.5 TPBAR Shipment Cask Cavity Internal Pressure-Normal Conditions

The method employed in the TPBAR (Tritium Producing Burnable Absorber Rod) shipment evaluation is similar to that employed in the fuel rod evaluations where the cask cavity free volume and molar gas quantities are combined with the ideal gas law (PV=nRT) to determine system pressure. The bounding TPBAR content condition consists of up to 300 production TPBARs (of which two can be prefailed) placed in a consolidation canister and loaded into a NAC-LWT with a TPBAR basket installed in the cavity.

A typical TPBAR is composed of a steel clad rod 0.381 inch in diameter, with a maximum length of 154.15 inches, and a minimum internal free volume of 5.727 $inch^3$. The TPBARs are located in a consolidation canister composed of three primary pieces: canister body, top insert, and bail. A spacer, attached to the NAC-LWT cask lid, assures that rods remain within the canister envelope and provides support to both the basket and canister under end-drop conditions. The TPBAR basket is a modified NAC-LWT PWR basket that increases the cavity free volume from that provided by a standard PWR basket design.

For conservatism in determining the cask internal pressure, the 298 TPBARs that are not prefailed at loading are assumed to undergo cladding failure during normal transport conditions. Prefailed rods have cladding damage that allows reactor coolant or spent fuel pool water to enter the rod cavity. Cladding failure during transport results in the release of the rod helium backfill gas, helium gas generated during the tritium production, and a portion of the tritium gas produced. For rods not prefailed, the majority of the tritium is locked in the TPBAR structure and is not released during normal or accident conditions of cask transport. Tritium release from intact or in-cask prefailed rods is limited to 55 Ci/rod (0.0019 mole – See Chapter 1, Appendix 1-B) versus a helium release of 0.42 mole per rod after the 90-day cool-down period. A conservative tritium release of 100 Ci per rod is applied in this calculation. After the 90-day cool-down period, the helium release increases according to the decay of tritium.

$$^{3}\mathrm{He}[\mathrm{moles}](t) = 0.398 \times \left(1 - \exp\left(\frac{-\ln(2)t}{12.33}\right)\right)$$

The remaining two rods in the 300 TPBAR shipment are assumed to be prefailed and waterlogged. These rods contain a maximum of 7.5 moles of H_2O and 0.199 moles of T_2O (1.2 grams H_2), 2% of which dissociate into their constituent gases due to elevated temperatures in the cask cavity (see Chapter 1 appendices). The NAC-LWT is vacuum dried prior to transport, removing water from the cask cavity. This process is expected to remove water from the

prefailed rod. The water content is conservatively assumed to remain in the rods for the pressure calculated. After dissociation, the total gaseous inventory in each prefailed TPBAR is 7.78 moles.

Cask cavity gases after rod failure are therefore comprised of the cask helium backfill (one atmosphere at loading); the combined helium rod backfill, helium generated during the tritium production, helium production from tritium decay, and the tritium release itself (298 rods); and the molar inventory of the two prefailed, waterlogged rods. The total free volume available for the gas is the cask cavity volume plus the internal free volume of the failed rods, minus the canister, spacer, basket, and rod volumes.

Description (Based on 300 Rods Failing)	Volume [cm^3]
Cavity (Empty)	4.10E+05
TPBAR Rod (Based on Exterior Rod Dimension)	-8.64E+04
TPBAR Minimum Free Interior	2.82E+04
TPBAR Consolidation Canister	-1.40E+04
TPBAR Basket	-8.49E+04
Cask Cavity Spacer	-5.55E+03
Cask Free volume (300 Rods Failed)	2.47E+05

The free volume in the cask cavity for intact rods is 2.19×10^5 cm^3. Applying a conservative volume 2.4×10^5 cm^3 to calculate the cask backfill molar quantity yields 9.98 moles of helium. The backfill conditions at sealing used in the calculation are one atmosphere pressure and a temperature of 68°F. The backfill pressure is specified in the operating procedure, while 68°F is conservative for the cask with a heat-generating payload.

Again employing the ideal gas law with a total 152 moles of gas (298 rods releasing 0.42 moles of helium and 0.003 mole of tritium, two waterlogged rods releasing 7.78 moles each, plus the 10.27 moles cask backfill), a conservative free volume of 2.47×10^5 cm^3, and the normal condition average gas temperature of 246°F, yields an operating pressure of 276 psig at the end of a 90-day cooldown. For a period of one year following the 90-day cooldown and considering a fixed gas temperature of 246°F, the pressure increases to 289 psig (MNOP). System pressure at cool times greater than 90 days will be lower due to the decreased heat loads associated with the radioactive decay of the payload.

The TPBAR content condition of up to 25 TPBARs contained in a 5×5 rod holder is bounded by the pressure analysis performed for the fully loaded TPBAR consolidation canister. Approximately 80% of the free volume in the consolidation canister analysis is the space

associated with the volume between radial extent of the consolidation canister and cask shell (i.e., void volume within the radial cross-section of the TPBAR basket). This volume is available regardless of TPBAR basket payload. Combining the 25 TPBAR reduced releasable gases (20% of the 300 TPBAR payload for 23 intact and two prefailed TPBARs) with a similar free volume assures lower pressure in the 25 TPBAR shipment configuration.

The TPBAR content condition of 55 segmented TPBARs contained in a sealed waste container is bounded by the pressure analyses performed for the fully loaded TPBAR consolidation canister. The contents include segments, debris and vented shrouds, all placed in a vented inner storage container within the welded waste container. Due to the condition of the TPBAR segments and the cooling period since irradiation, the heat load of the waste container is 0.127 kW, significantly less than the 0.693 kW analyzed for the production TPBAR content condition of 300 TPBARs in an open consolidation canister.

Each of the 55 TPBARs is assumed to contain a maximum of 1.2 grams of tritium at the time of sealing the waste container, and all backfill gases have been vented. For the purpose of the maximum pressure analysis, all of the contained tritium is assumed to decay to He^3, resulting in a total of 66 grams of He^3. Note that the confinement boundary of the welded waste container is assumed to fail during normal transport conditions. Due to the state of the TPBAR segments and the loading of the materials in dry loading conditions, no water will be present in the waste container. Conservatively assuming that the cask free volume and gas temperature for the transport of the waste container is the same as that for the production TPBAR contents listed previously (2.47×10^5 cm^3), the calculated maximum normal operating pressure (MNOP) for the 55 segmented TPBARS in the waste container is less than 65 psia. Therefore, the MNOP for the 55 segmented TPBAR content condition is conservatively bounded by the MNOP of the 300 production TPBARs in the consolidation canister of 289 psig.

Combustible Hazard Assessment

Hydrogen may be released by prefailed, waterlogged TPBARs (TPBARs damaged during in-core use or in-pool storage) primarily in the form of water, tritiated water, and potentially some tritiated methane. Each prefailed TPBAR has the potential to release the tritium assumed to dissociate from tritiated water (0.004 moles) as well as up to 0.15 moles hydrogen gas dissociated from 7.5 moles of H_2O. Both the hydrogen and the tritium gas, as well as the water and tritiated water, will be removed from the cask during vacuum drying prior to helium backfill.

The flammability/ignitability characteristics of tritium (T_2) in the presence of oxygen are substantially the same as for hydrogen (H_2). Hydrogen in air reaches a lower flammability limit at 4% volume.

Tritium escapes intact TPBARs in the form of molecular tritium gas at a rate of less than 0.12 mCi/hr/TPBAR. For a one-year transport period and a 300 TPBAR payload this rate results in a release of less than 0.01 moles of T_2 gas. Given a helium gas back-fill of approximately 10 moles helium (1 atmosphere) no flammability hazard exists for intact TPBARs.

Tritium may be released by event-failed (in-cask failed) TPBARs in the form of tritiated methane (CH_4) or tritiated water. Event-failed TPBARs may release up to 55 Ci of tritium. This translates to approximately 0.002 mole of tritium that may be released from a TPBAR in conjunction with 0.42 mole of helium.

The reduced available hydrogen content of the 25 TPBAR payload, including two prefailed TPBARs, in combination with a similar free volume to the consolidation canister configuration, and cask helium backfill quantity, produces a lower maximum hydrogen concentration in the up to 25 TPBAR configuration than that of the 300 TPBAR consolidation canister configuration. No flammability hazard exists for the up to 25 TPBAR shipment configuration.

The 55 equivalent TPBARs, in segments and debris, may release up to 100% of the tritium contained in the pellets during transport. The pellets contain approximately 40% of the tritium quantity in the TPBAR. At NAC-LWT normal and accident conditions temperatures, the TPBAR components release tritium primarily as tritiated water with only a small fraction (maximum 2%) as gaseous tritium (see Appendix 1-B of Chapter 1). During a one-year transport, an additional maximum 1% of the tritiated water may undergo radiolysis and dissociate. Conservatively applying a maximum 3% release rate to the 55 equivalent TPBAR total inventory of 66 grams (1.2 grams per rod) yields an inventory of 0.33 moles T_2. Based on an inert gas cask backfill in excess of 10.3 moles, a bounding estimated maximum hydrogen (T_2) volume fraction of 3.1% is calculated. Therefore, no flammability hazard will exist for the 55 segmented TPBAR content condition.

3.4.4.6 Maximum Internal Pressure for PULSTAR Fuel Element Payload

Based on the allowable loading configurations for PULSTAR fuel elements, cask internal pressures are calculated for a payload of 28 intact assemblies and a mixed payload of 14 intact assemblies and the equivalent of 14 canned assemblies. A payload of 28 4×4 intact rod inserts is bounded by either of these evaluated payloads.

The ideal gas law and Dalton's law of partial pressures are used to calculate internal pressures. Cask, can, and element backfill initial conditions are taken as a pressure of 1 atm and a temperature of 68°F.

PULSTAR fuel element and fuel assembly dimensions are summarized in Table 3.4-18. Elements are UO_2 pellets clad with zirconium alloy. A PULSTAR fuel assembly is a 5×5 rectangular array of elements with aluminum upper and lower fittings.

Based on the PULSTAR fuel element, PULSTAR failed fuel can, MTR basket stack, and NAC-LWT cavity dimensions, volumes are calculated and summarized in Table 3.4-19.

The remaining two inputs to the pressure calculation are the temperature and fission gas inventory. For conservatism, the average gas temperature applied is the maximum TRIGA fuel clad temperature of 295°F. The TRIGA temperatures are applicable to the PULSTAR fuel element evaluation as discussed in Section 3.4.1.13. The fission gas inventory is taken from the SAS2H results discussed in Chapter 5 and is shown in Table 3.4-20.

For a payload of 28 intact PULSTAR fuel assemblies, the partial pressures of the cask, element (rod) backfill, and fission gases are summed. The cask free volume is 217 liters and is calculated by subtracting the basket stack volume and the assembly envelope volume (multiplied by 28) from the cavity volume. The partial pressure of the cask, P_{Cask}, is simply the initial backfill pressure multiplied by the temperature ratio:

$$P_{Cask} = 1\,\text{atm}\frac{419.26\,\text{K}}{293.15\,\text{K}} = 1.430\,\text{atm}$$

The cask partial pressure due to a 100% release of element backfill, $P_{RodBackfill}$, is the initial backfill pressure multiplied by the temperature ratio and the backfill-to-cask volume ratio:

$$P_{RodBackfill} = 1\,\text{atm}\frac{419.26\,\text{K}}{293.15\,\text{K}}\frac{2.7\,\text{liters}}{217.0\,\text{liters}} = 0.018\,\text{atm}$$

Only 3% of this pressure contributes to the total pressure under normal conditions.

The cask partial pressure due to a 100% release of the element fission gases, $P_{FissionGas}$, is calculated using the Ideal Gas Law:

$$P_{FissionGas} = \frac{28 \cdot 0.448 \cdot 0.08205 \cdot 419.26}{217} = 1.989\,\text{atm}$$

Only 30% of the fission gases are released, and only 3% of the resultant pressure contributes to the total pressure under normal conditions.

The total cask pressure is the sum of the partial pressures, adjusted by the relevant release fractions:

$$P_{Total} = P_{Cask} + 0.03 \cdot P_{RodBackfill} + 0.03 \cdot 0.30 \cdot P_{FissionGas}$$

$$P_{Total} = 1.430 + 0.03 \cdot 0.018 + 0.03 \cdot 0.30 \cdot 1.989 = 1.449\,\text{atm}$$

Cask internal pressure for a mixed payload is calculated in a similar fashion, with a smaller cask free volume due to the difference in can and assembly envelope volume, and an assumed 100% failure rate of PULSTAR elements in either the screened or sealed can. The calculated maximum cavity pressure is 1.8 atm. Pressure in the sealed can is based on a 100% failure rate, the can cavity volume, and a payload equivalent in volume to 25 intact PULSTAR fuel elements. Normal condition pressure in the sealed can is 4.4 atm.

A summary of the pressure calculations is given in Table 3.4-21.

3.4.4.7 Maximum Internal Pressure for 16 PWR MOX/UO_2 Fuel Rods in a Rod Holder

Based on the allowable loading of up to 16 PWR MOX/UO_2 fuel rods, cask internal pressures are calculated. Bounding cask free volume, gas temperatures, and rod backfill pressure are directly obtained from the BWR high burnup rod evaluations in Section 3.4.4.3.

Variable	Unit	Value
Cask Free Volume (PWR Basket with Insert/Canister/Rod Holder)	in^3	5908
Normal Condition Cask Average Gas Temperature	°F	600
Normal Condition Cask Backfill Partial Pressure (at temperature)	psia	29.3
PWR Fuel Rod Backfill Pressure	psia	565

These values are combined with a conservative 2.9 in^3 fuel rod free volume and SAS2H calculated fission and actinide gas inventories to determine system pressure. The 2.9 in^3 free rod volume applied here is larger than the UO_2 rod volume previously employed (2.5 in^3) to account for additional volume designed into the MOX rods to counter any potential increase in fission gas release from the PuO_2 / UO_2 MOX fuel mixture.

The ideal gas law and Dalton's law of partial pressures are used to calculate internal pressures by combining cask backfill, rod backfill, and fission/actinide gases. Fill temperature applied to the rod gases is 22°C (standard temperature). Maximum fission and actinide gas inventories were obtained from 80 GWd/MTHM fuel rod, 3% enriched ^{235}U or 3 wt % fissile Pu, SAS2H output sets. The fuel rod corresponds to the maximum fissile mass defined in the shielding source term calculations. SAS2H runs produced a total gas inventory of 0.29 moles per rod (99+% fission gas), with bounding values obtained from the UO_2 rods (MOX rods produce approximately 98% of the UO_2 rod fission gas). Gas inventories increase as a function of reduced initial fissile material content. A 3% enrichment and/or 3% fissile Pu content is significantly below levels required to reach an 80 GWd/MTHM burnup level.

The resulting normal condition pressure for a failure fraction of 1/16 (bounds the 3% normal condition PWR rod failure fraction in the Standard Review Plan, NUREG-1617, Supplement 1) and 30% fission gas release is 17.2 psig (31.9 psia, or 2.2 atm).

Parametric studies are performed on the number of rods failing and the release fraction under normal conditions with an applied limit of 50 psig (normal condition structural analysis input value). Normal condition failure of up to 13 rods, at 100% gas release, remains below 50 psig. A similar analysis results in a maximum normal condition pressure of 48.5 psig for a normal condition failure of all 16 rods at a 75% fission gas release fraction (100% of backfill gas is released). Given that each of the rods is individually located within a support tube, no normal condition rod failures are expected during transport.

UO_2 or MOX rods included in the payload may be IFBA rods. As presented in Section 3.4.4.3, IFBA rods are expected to contribute in the range of 0.04 mole per rod to system pressure, assuming the absorber material is boron. As the MOX/UO_2 pressure calculations assumed a conservative 100% fission gas release of 0.29 mole per rod, a rod backfill of 0.075 mole, and a cask backfill of approximately 3.6 moles, the release of IFBA boron-generated gases would not significantly affect system pressure.

3.4.4.8 Maximum Internal Pressure for Aluminum-Based Fuels

This section determines the bounding NAC-LWT transportation system internal pressure for the cask during normal conditions for aluminum-based research reactor fuel payloads (i.e., ANSTO, DIDO and MTR fuels).

This analysis uses a combination of thermodynamic principles and dimensional analysis to calculate internal pressure. The basic functions employed are the Ideal Gas Law ($Pv = NRT$) and Dalton's Law of Partial Pressure. For a given cask free gas volume, internal pressure is a function of fission gas and cask backfill gas. The aluminum-based plate element does not contain any backfill gases or free volume within the clad.

Volume, temperature and backfill inputs required for the system pressure evaluations are summarized in Table 3.4-23.

Standard temperature (22.2°C) is used for the cask backfill initial temperature. This is a reasonable assumption, as backfill gas will rapidly increase in temperature during cask fill operations. As the payload generates decay heat, the average temperature at sealing is expected to be significantly higher than the standard temperature. Minor changes in temperature, translated to absolute temperature for pressure calculations, would not affect the results of the calculation significantly.

3.4.4.8.1 Fuel Fission Gas Content

SAS2H source term calculations documented in Chapter 5 were used to generate fuel gamma and neutron sources. Included in this determination are gram quantities of light elements, fission products and actinides. Fission gas inventories are extracted from the ANSTO spiral fuel, DIDO LEU, MEU and HEU, and maximum fuel mass MTR LEU, MEU and HEU cases. Only the ANSTO spiral fuel is required as ANSTO DIDO fuel is bounded by the standard DIDO fuel definition, and ANSTO MOATA fuel is bounded by the generic MTR fuel definition. Minimum transport cool times are chosen for the analysis. None of the payloads generate significant quantities of actinide alpha decay gases, as plutonium generation is limited in the fuel elements modeled at 19% or greater ^{235}U enrichments. The negligible buildup of alpha decay gases makes the choice of cool time insignificant to the analysis results.

Fission product and actinide gas inventories in grams extracted from the SAS2H outputs are listed in Table 3.4-24. Fission gas inventories in grams are converted to inventories in moles using Avogadro's number and the atomic mass of each isotope. As illustrated in Table 3.4-25, the total molar quantity of fission gas does not vary significantly between various enrichment levels for a given fuel type. The MTR elements produce the bounding fission gas content. The majority of fission gases, ~85%, is comprised of Xenon isotopes. There is no significant quantity of helium or tritium.

3.4.4.8.2 Normal Condition Pressures

Using Dalton's Law of partial pressures, the NAC-LWT cask cavity pressure may be calculated by first determining the partial pressure of the released fission gases and adding it to the cask backfill gas. Gas available for release from the fuel elements depends on the fueled surface area exposed by clad-through damage.

Cask Backfill Gas

Based on the ideal gas law, the pressure of the cask backfill gas is simply the ratio of the backfill temperature at testing (assumed at standard temperature) to the operating condition temperature.

$$P_{\text{Cask Backfill}} = 14.7 \text{ psi} \times \frac{T_{\text{Operating Temperature}}}{T_{\text{Standard Conditions}}}$$

Partial pressures of the cask backfill at normal and accident conditions are 23.8 psi and 25.9 psi for ANSTO/DIDO and MTR payloads, respectively.

Fission Gas

The pressure of rod fission gas is calculated using release fraction (or surface area fraction assuming 100% release from the unclad fuel meat), the quantity of fission gas in the element, the cask cavity backfill temperature and the cask cavity gas temperature.

$$n = \text{Fission Gas Moles (Cask)} \times \text{Release Fraction}$$

$$P = \frac{\text{nRT}}{\text{V}}$$

For the MTR LEU fuel, a sample calculation based on a 50% surface area exposed with a 100% gas release from the exposed surface area is:

$$P_{\text{Fission Gas}} == \frac{0.455\dfrac{\text{moles}}{\text{element}} \times 42\dfrac{\text{elements}}{\text{cask}} \times 50\% \times 0.08206\dfrac{\text{liters} - \text{atm}}{\text{k} - \text{mole}} \times 470.2\text{K}}{229.3\,\text{liters}}$$

$$P_{\text{Fission Gas}} = 1.61\,\text{atm} = 23.7\,\text{psi}$$

Normal Pressure

Normal and accident pressures can now be generated at the various release/surface area fractions. Only LEU MTR and DIDO elements are summarized as they produce the maximum MTR and DIDO fission gas quantities and, therefore, pressures. Results are summarized in Table 3.4-26 as partial pressure of the fission gas and total system pressure in psia and psig. To meet a 50 psig system structural analysis limit, a maximum 80% of the MTR and 100% of the DIDO/ANSTO gases can be assumed to escape from the plates. As MTR plates with significant through-clad damage will have released a portion of their gas inventory prior to transport (i.e., during in-core use, storage and cask vacuum drying), system pressure is expected to remain below 50 psig when considering all fission gas released from the MTR plates.

Note that experimental data summarized in WSRC-TR-98-00317, October 1998, "Bases for Containment Analysis for Transportation of Aluminum–Based Spent Nuclear Fuel," Section 5.3.1, indicates no significant release of gases from exposed fuel material occurs at the temperature (200°C -300°C) of the NAC-LWT cask cavity and contents with aluminum-based fuel payload.

3.4.5 Maximum Thermal Stresses

The conditions within the range of normal transport conditions and fabrication that result in the limiting combination of thermal gradient and isothermal stresses have been evaluated. The analyses are performed in Sections 2.5 through 2.7. The resulting isothermal temperature plots are presented in Section 2.10.3.

3.4.6 Evaluation of Package Performance for Normal Conditions of Transport

Section 3.4 provides analyses of the NAC-LWT cask thermal performance for normal transport conditions. The analyses demonstrate that the NAC-LWT cask thermal performance meets the criteria of 10 CFR 71 for normal transport conditions.

The maximum fuel rod cladding temperature under normal transport conditions is 472°F. This is well below the temperatures that can cause fuel rod cladding deterioration. Components important to safety remain within their safe operating ranges (Section 3.3) during normal transport conditions. Thermally induced stresses (in combination with pressure and mechanical load stresses) are less than allowable stresses as shown in Section 2.6. Thus, the analyses of Section 3.4 demonstrate that the NAC-LWT cask fulfills the heat rejection criteria established in Section 3.1 for normal transport conditions.

Figure 3.4-1 HEATING5 Normal Transport Conditions Thermal Model

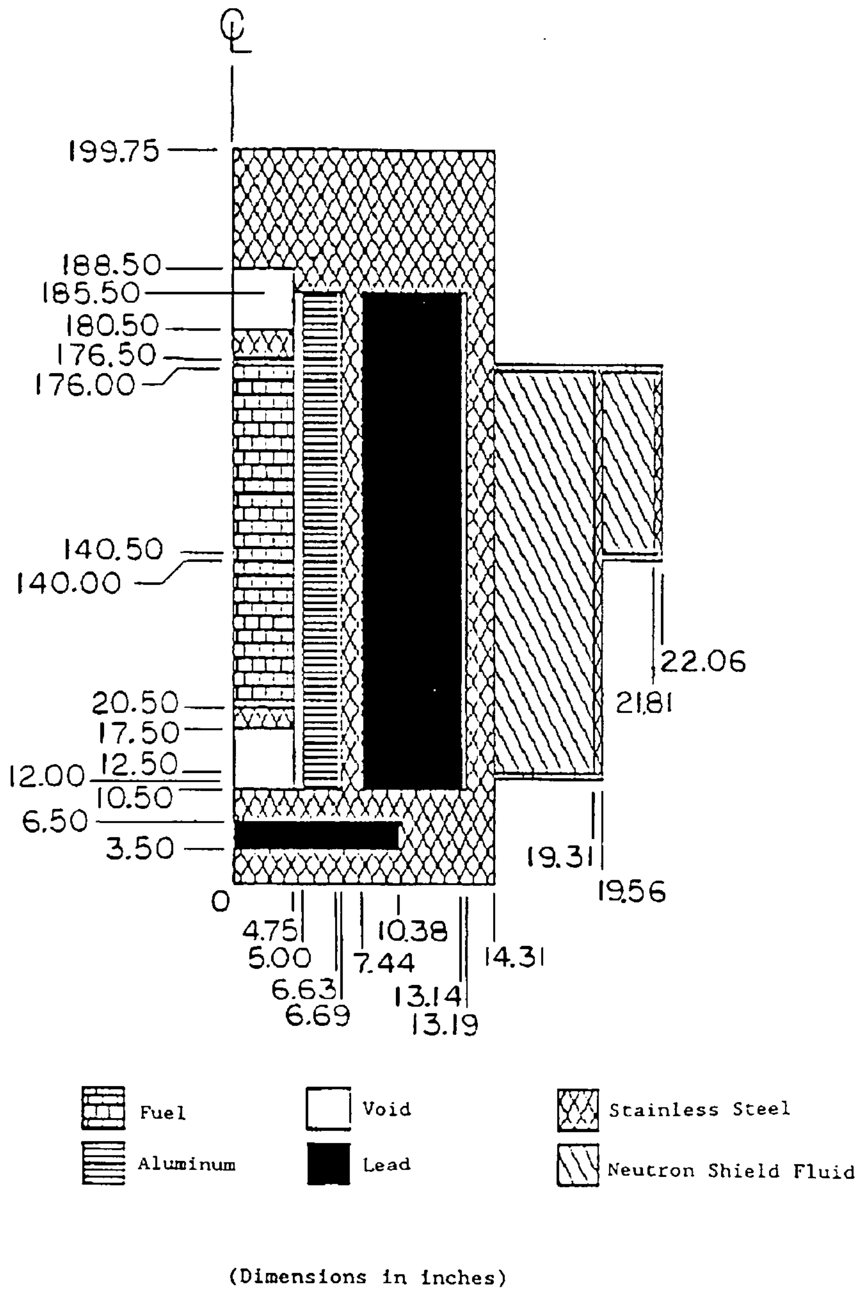


Figure 3.4-2 Design Basis PWR Fuel Assembly Axial Flux Distribution

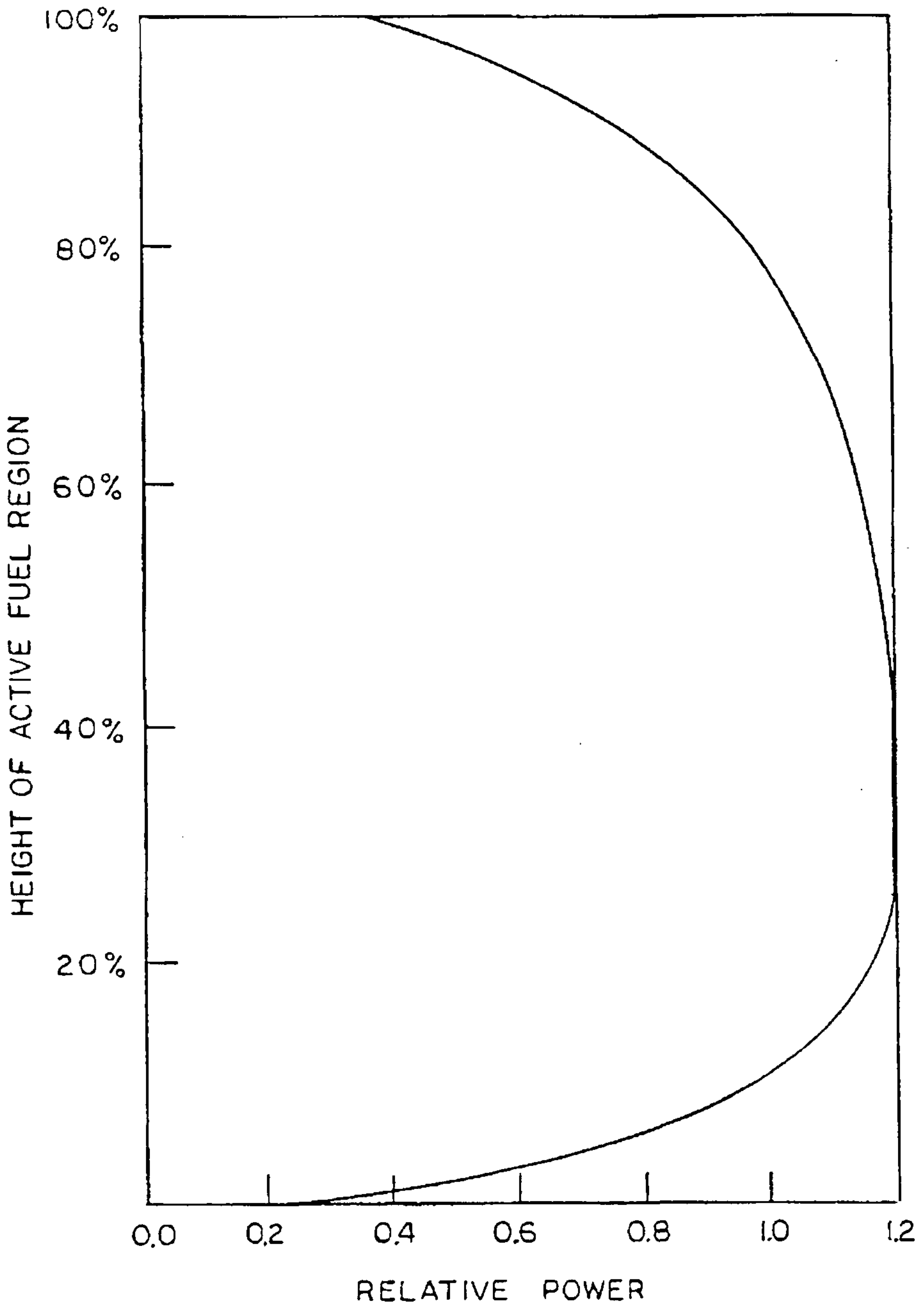


Figure 3.4-3 ANSYS MTR Fuel Design Basis Heat Load Thermal Model (Uniform 30-Watt/Element Configuration Heat Load)

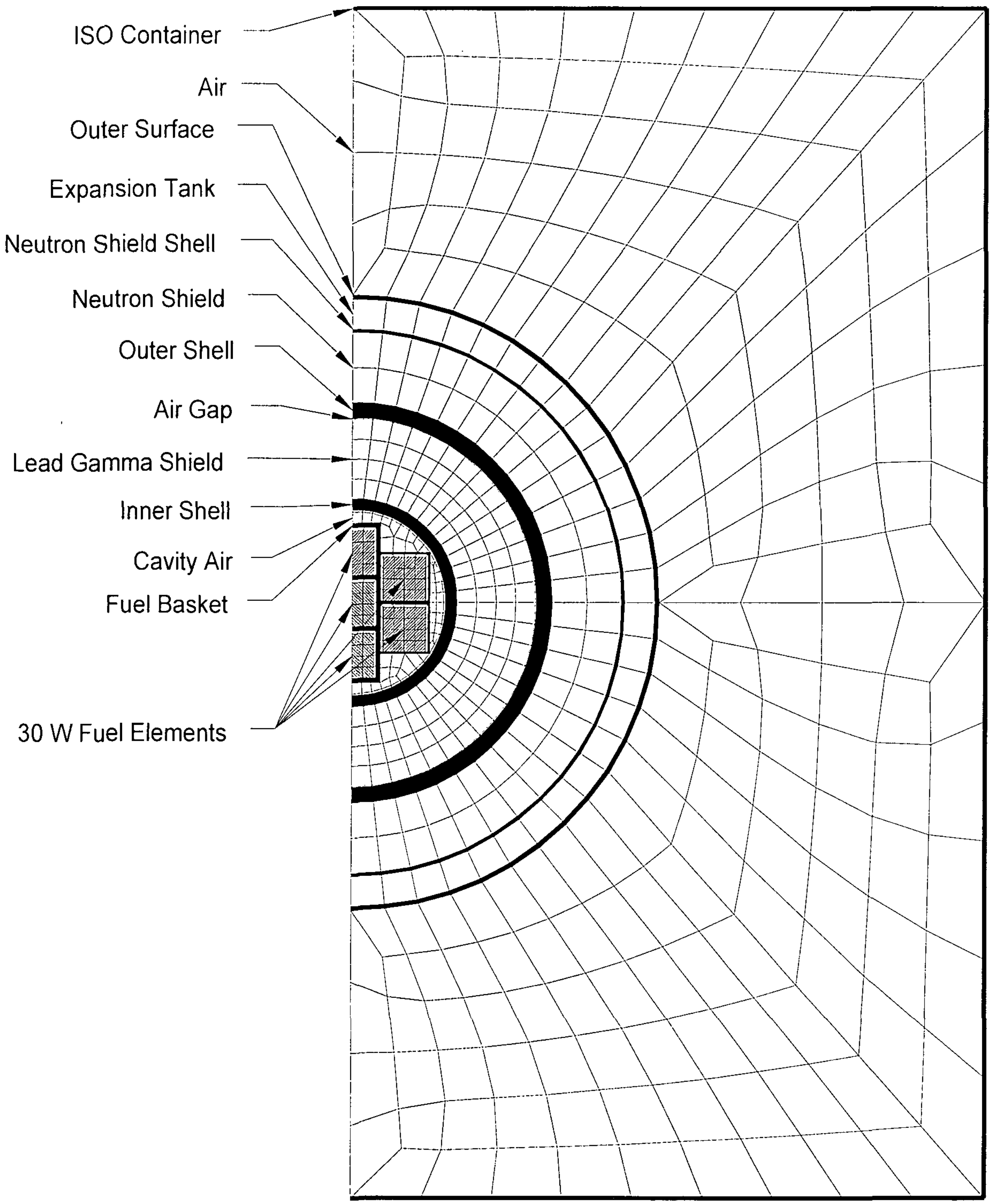


Figure 3.4-4 MTR Fuel Variable Decay Heat ANSYS Thermal Model
(120-Watt / 70-Watt / 20-Watt Configuration Heat Load)

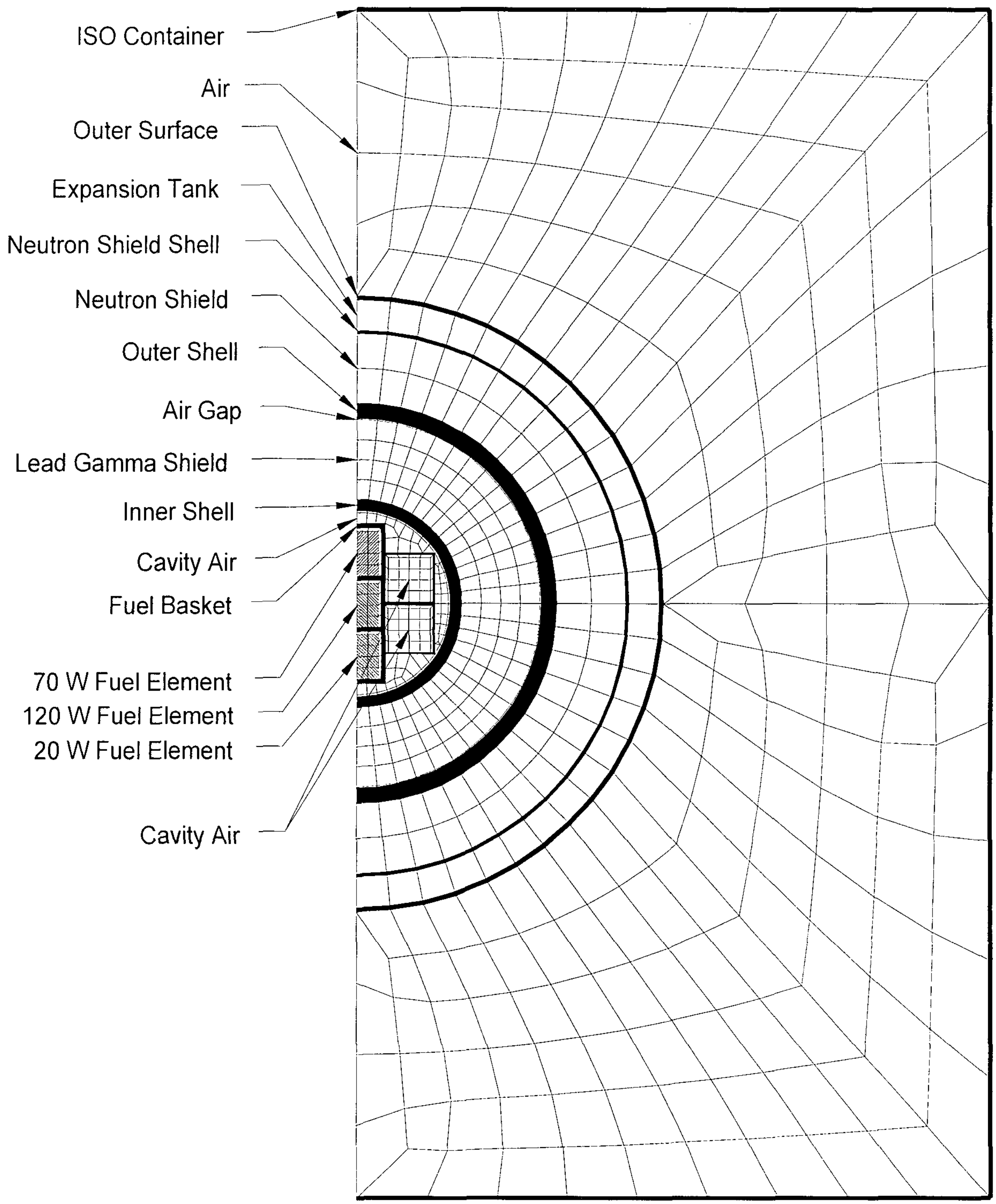
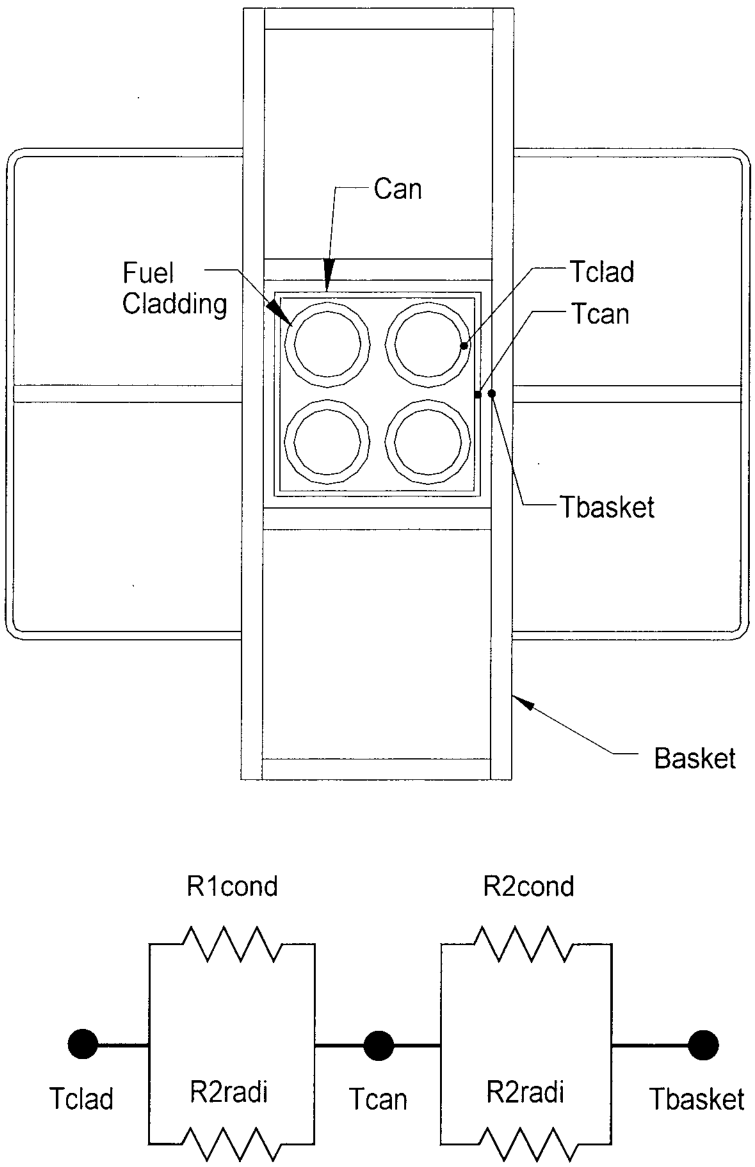


Figure 3.4-5 Thermal Resistance Model for TRIGA Fuel Elements



Where:

Rcond = Thermal resistance for conduction

Rradi = Thermal resistance for radiation

Tbasket = Maximum basket temperature

Tcan = Maximum can temperature

Tclad = Maximum fuel cladding temperature

Figure 3.4-6 Modeling Details for the MTR Fuel Assembly Resting on the Surface of the NAC-LWT MTR Basket

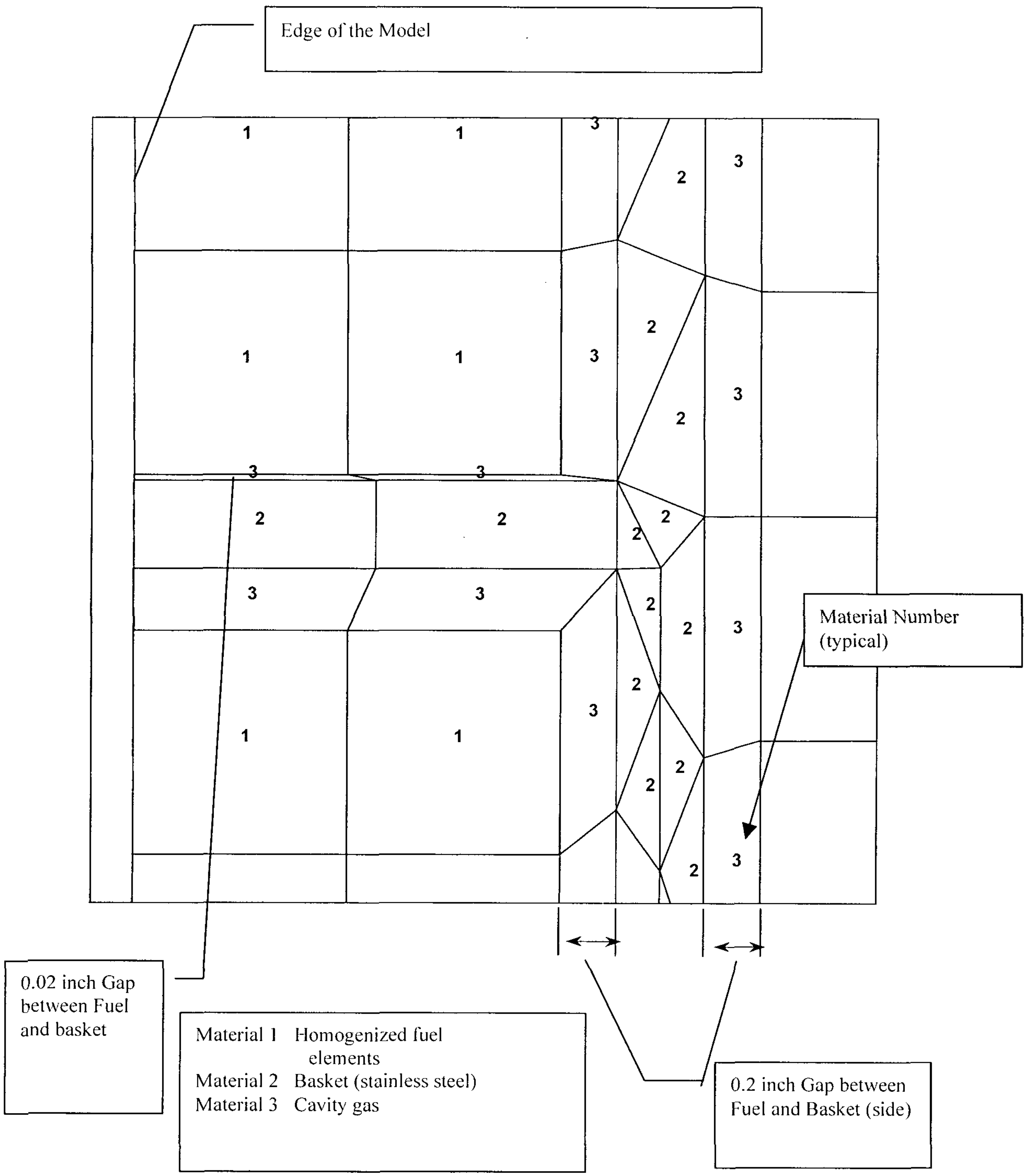


Figure 3.4-7 Finite Element Thermal Model for TRIGA Fuel Cluster Rods

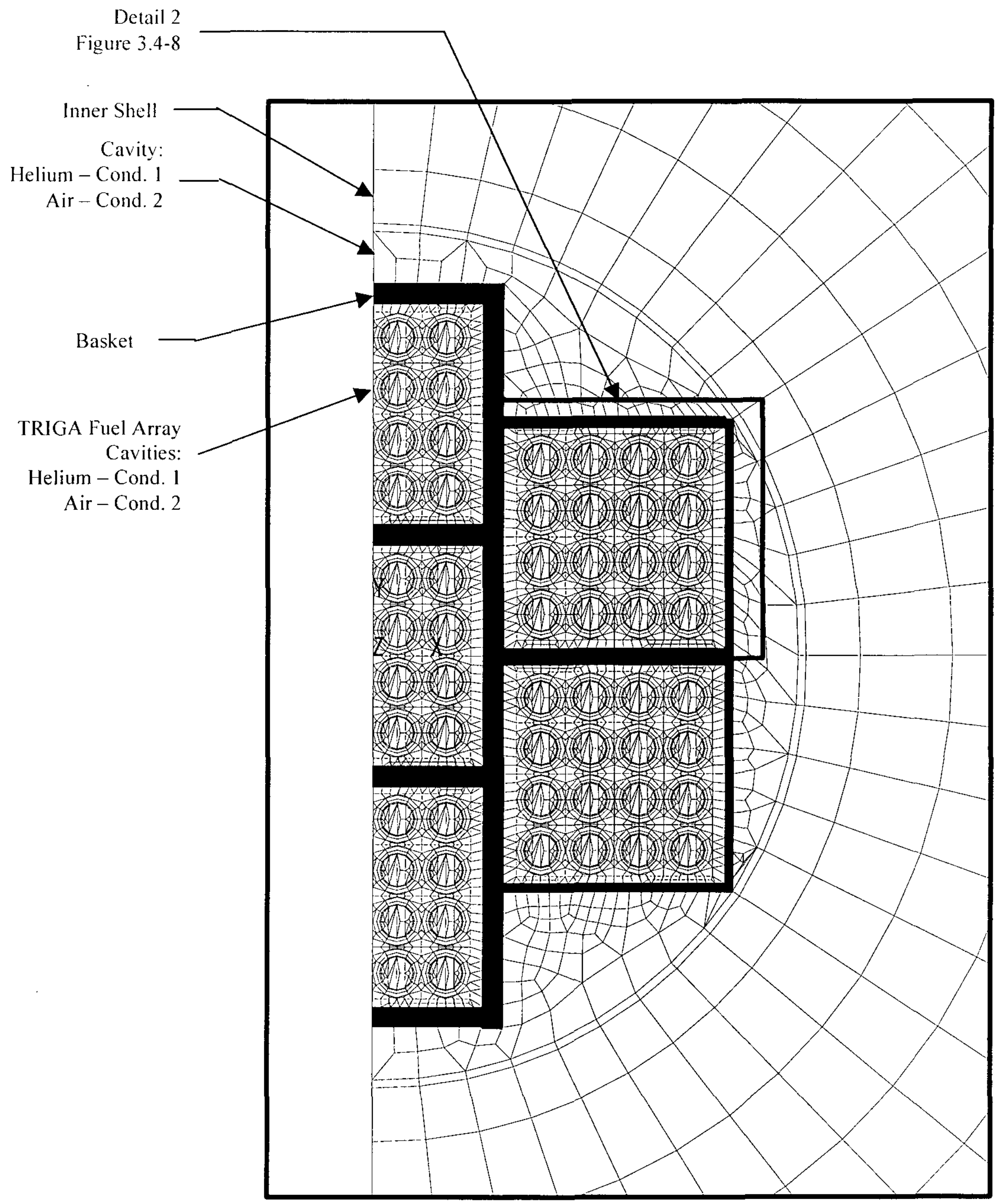


Figure 3.4-8 Details of the TRIGA Fuel Cluster Rods in the Finite Element Model

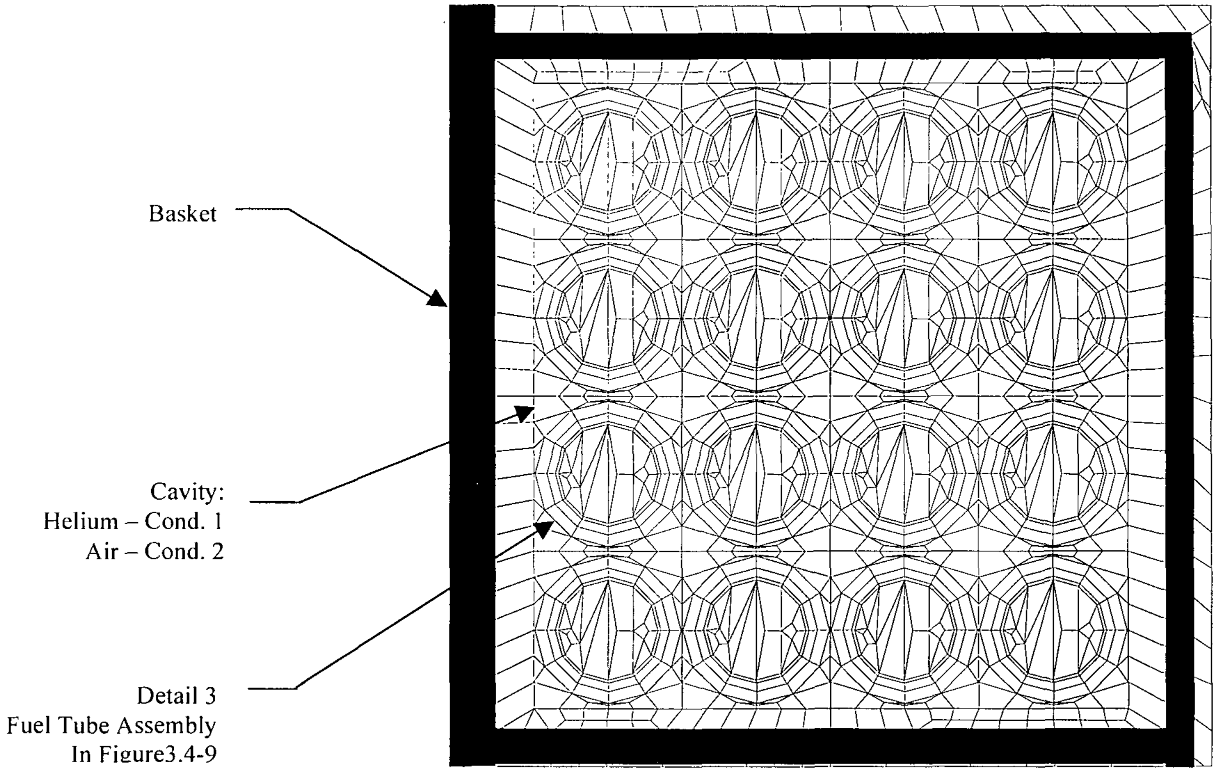


Figure 3.4-9 Individual TRIGA Fuel Cluster Rod Finite Element Model Details

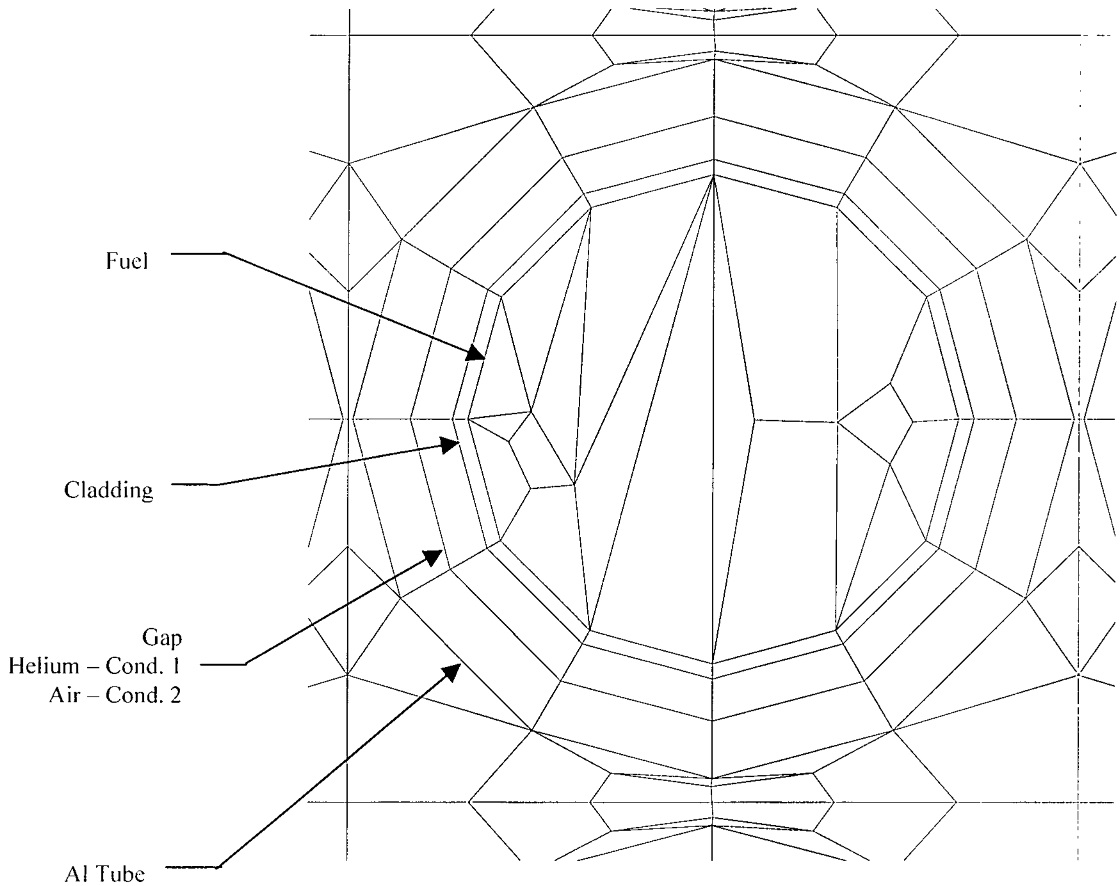


Figure 3.4-10 PWR and BWR High Burnup Fuel Rods Normal Condition ANSYS Thermal Model (Condition 1)

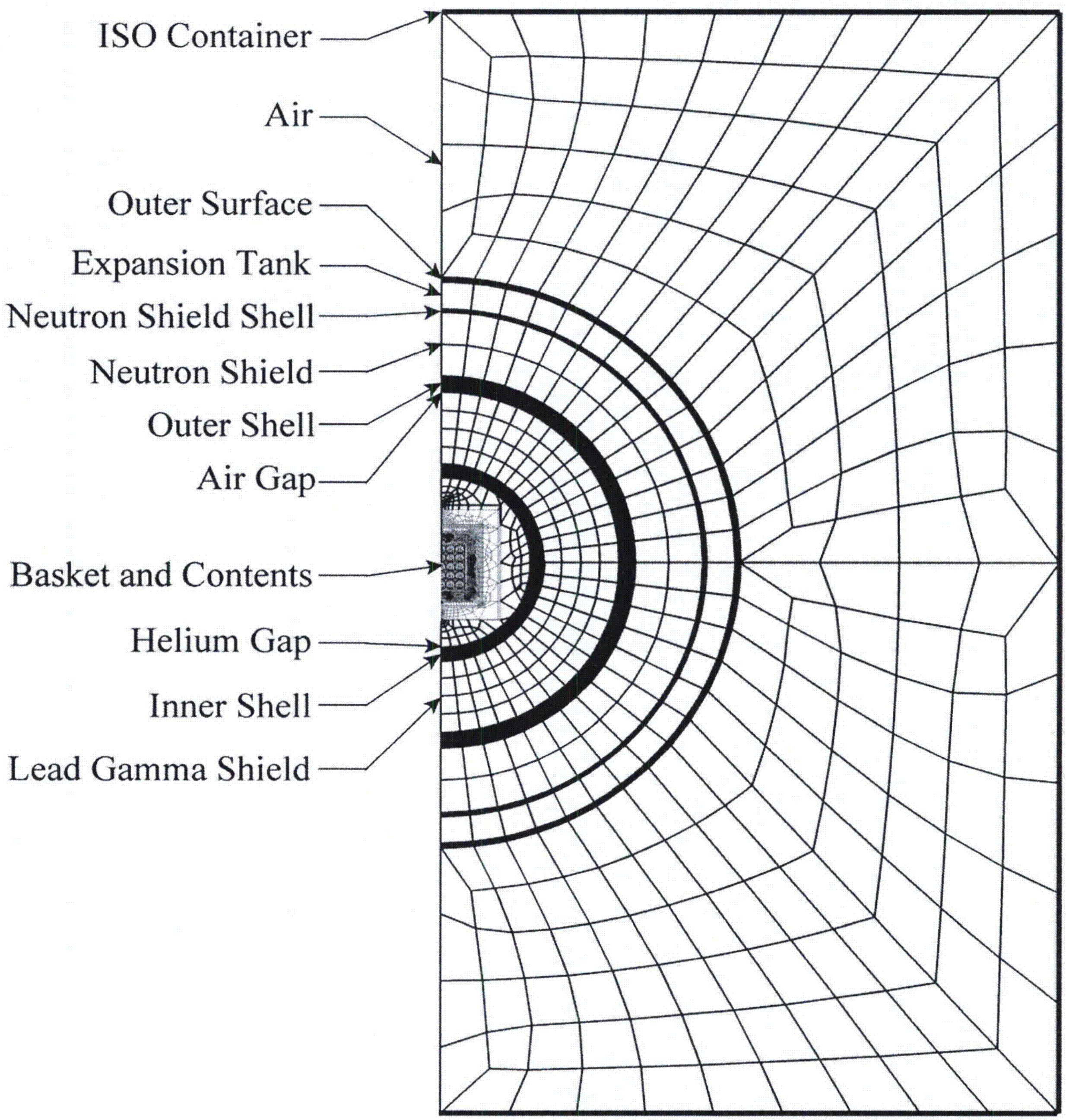
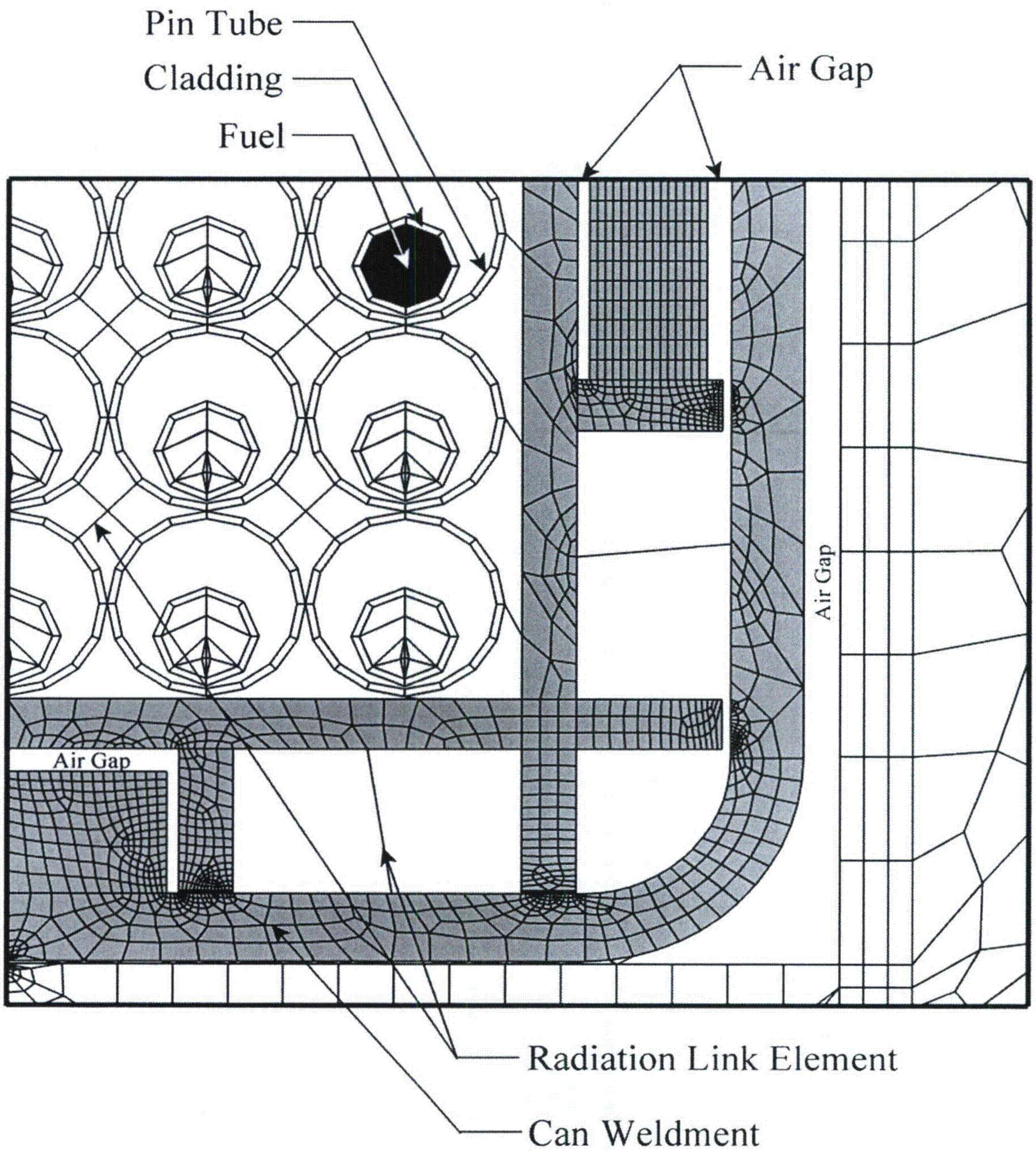


Figure 3.4-11 Close-up of PWR and BWR High Burnup Fuel Rods Normal Condition ANSYS Thermal Model



Note: air elements are not shown for clarity.

Figure 3.4-12 PWR and BWR High Burnup Fuel Rods Normal Condition ANSYS Thermal Model (Condition 2)

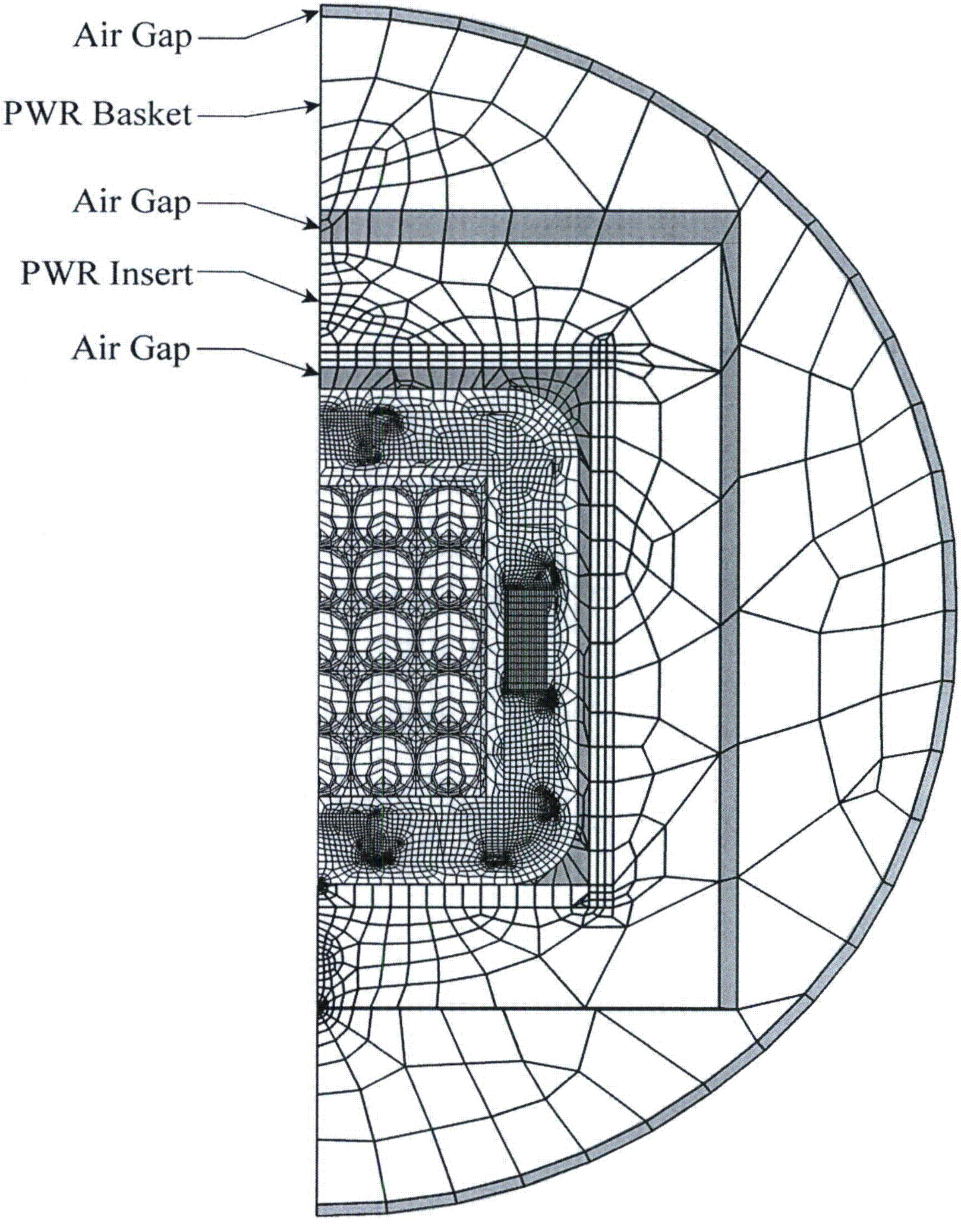
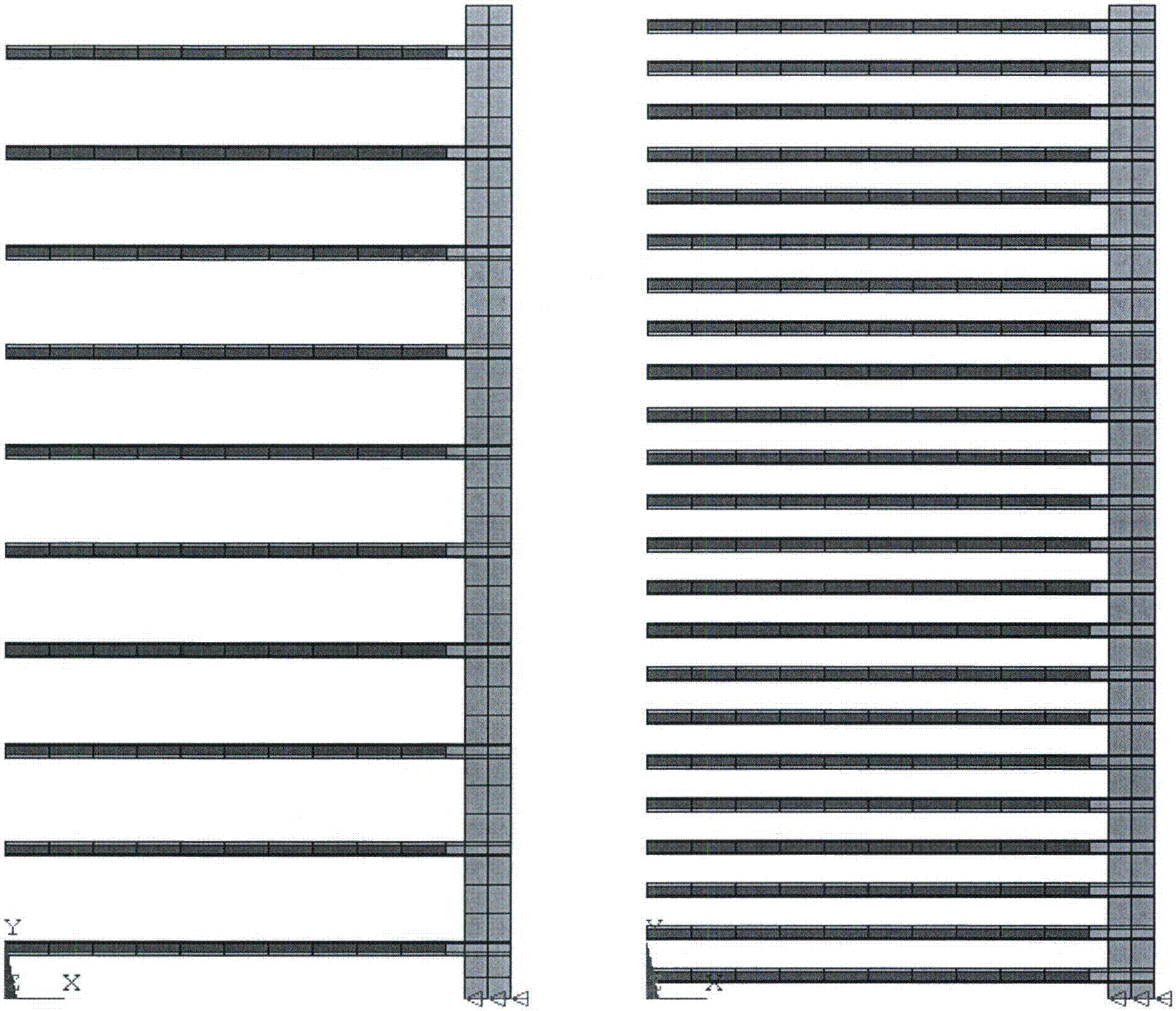


Figure 3.4-13 Finite Element Thermal Model for MTR Fuel Element



10 Fuel Plates

23 Fuel Plates

Application of temperature boundary condition

(Air elements omitted for clarity)

Figure 3.4-14 Detailed DIDO Basket Module Finite Element Model

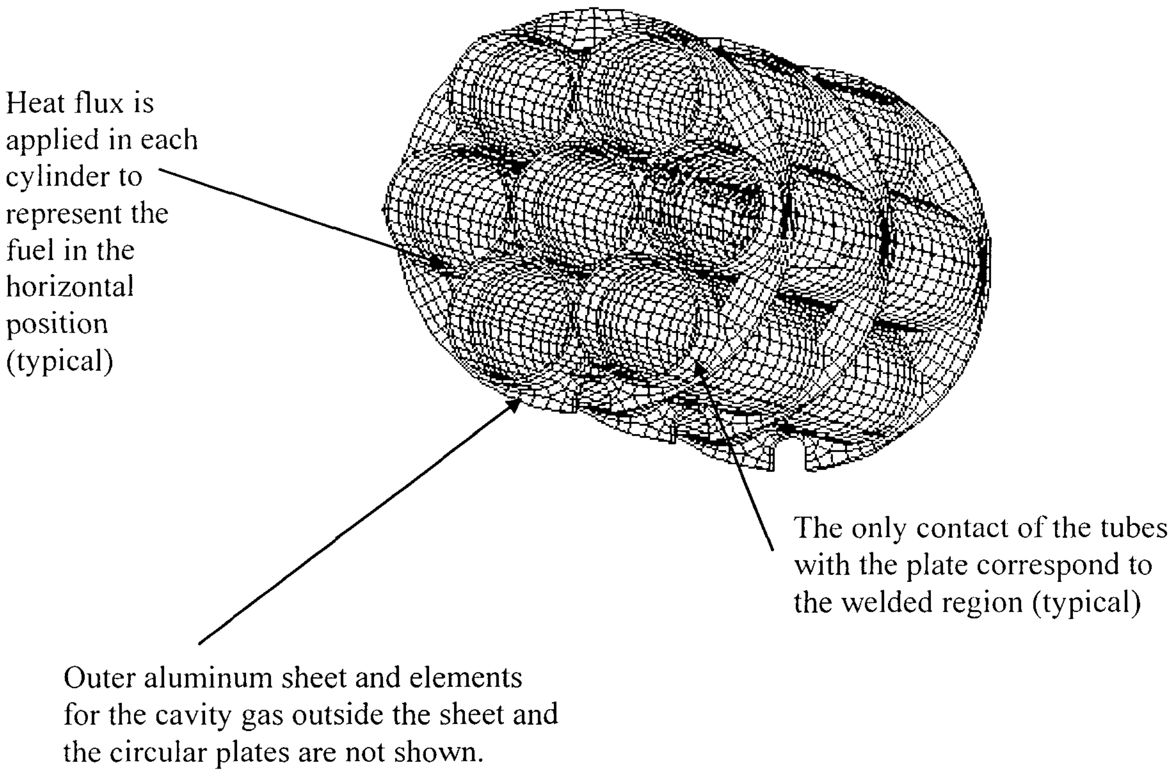


Figure 3.4-15 Detailed DIDO Fuel Assembly Model

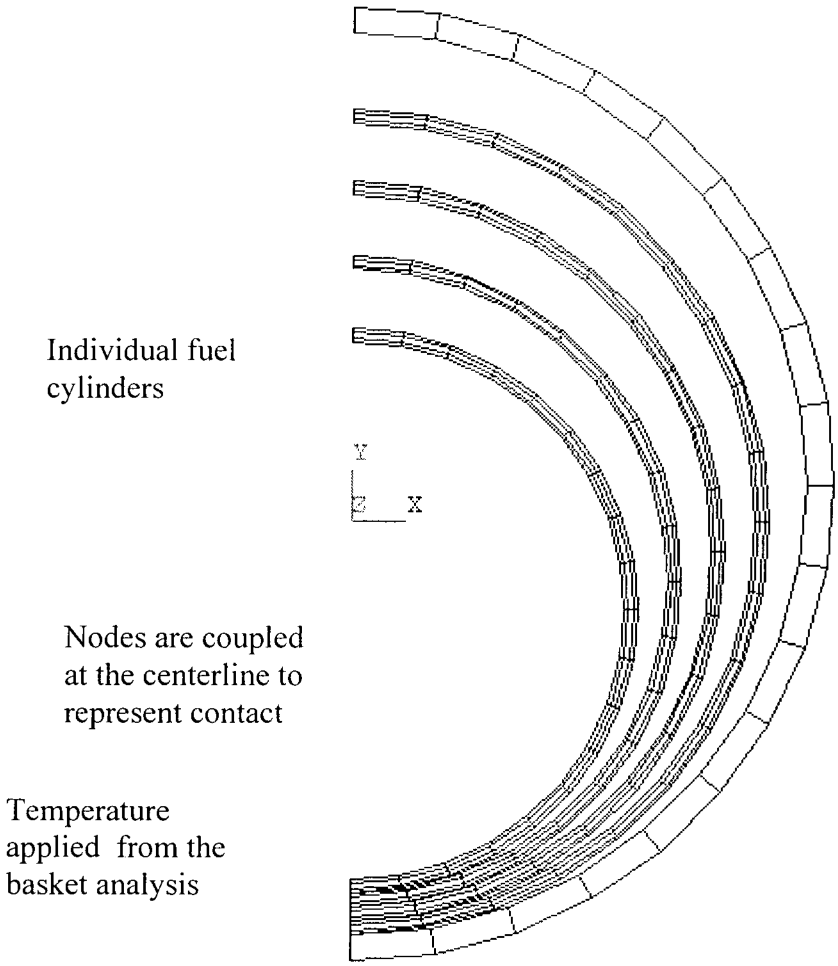


Figure 3.4-16 ANSYS Model for BWR 7 × 7 Fuel Lattice with 25 High Burnup Fuel Rods

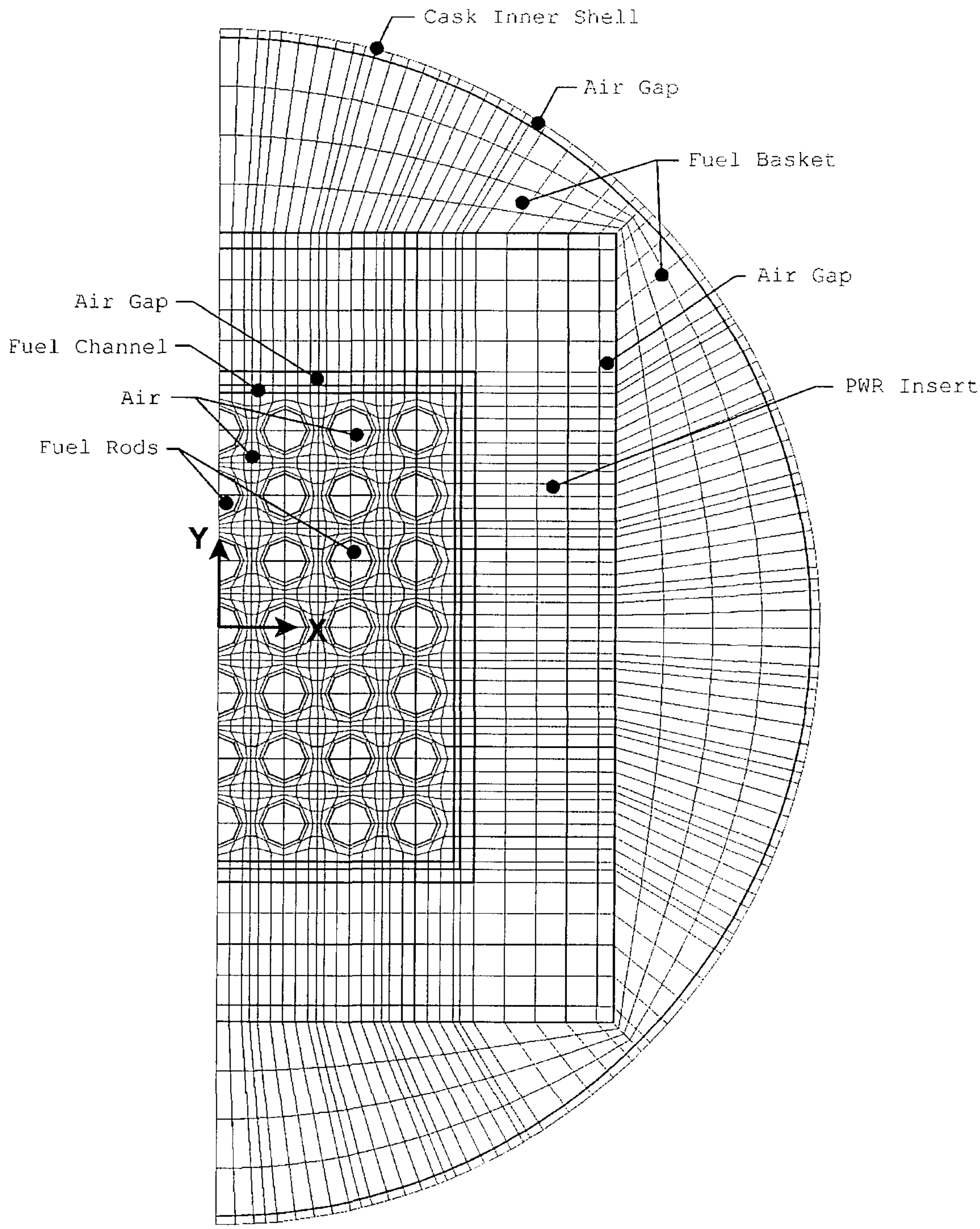


Figure 3.4-17 Fuel Rod Locations in the Thermal Model for Damaged Fuel

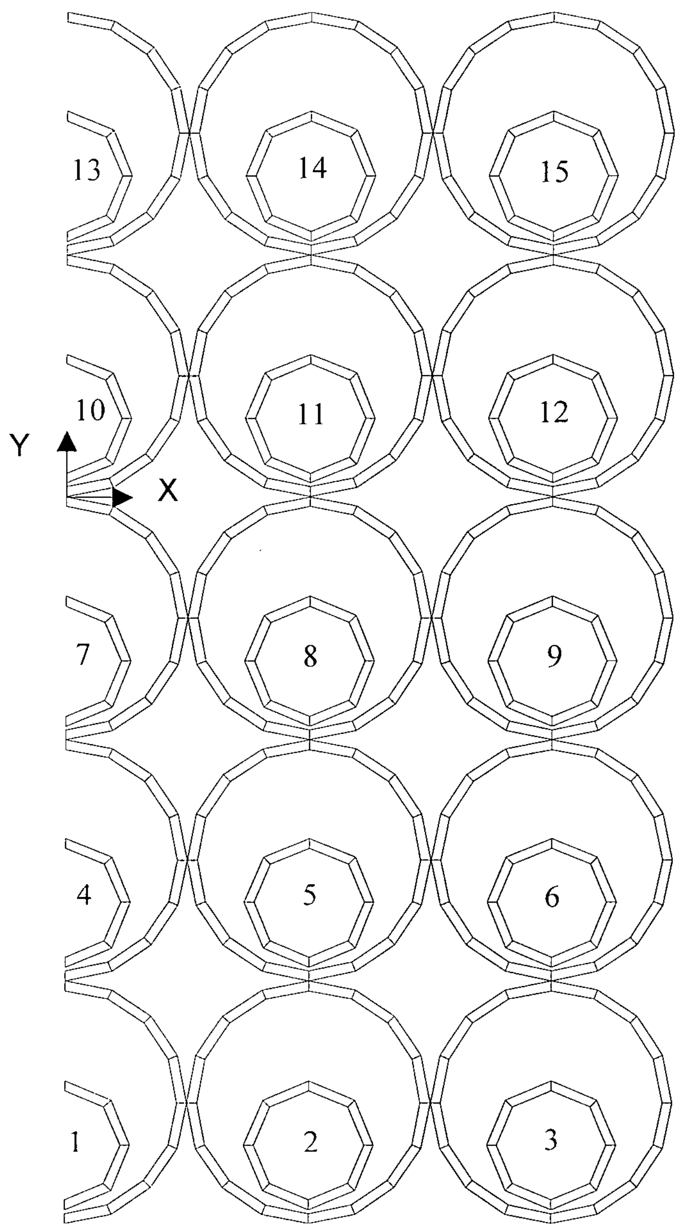
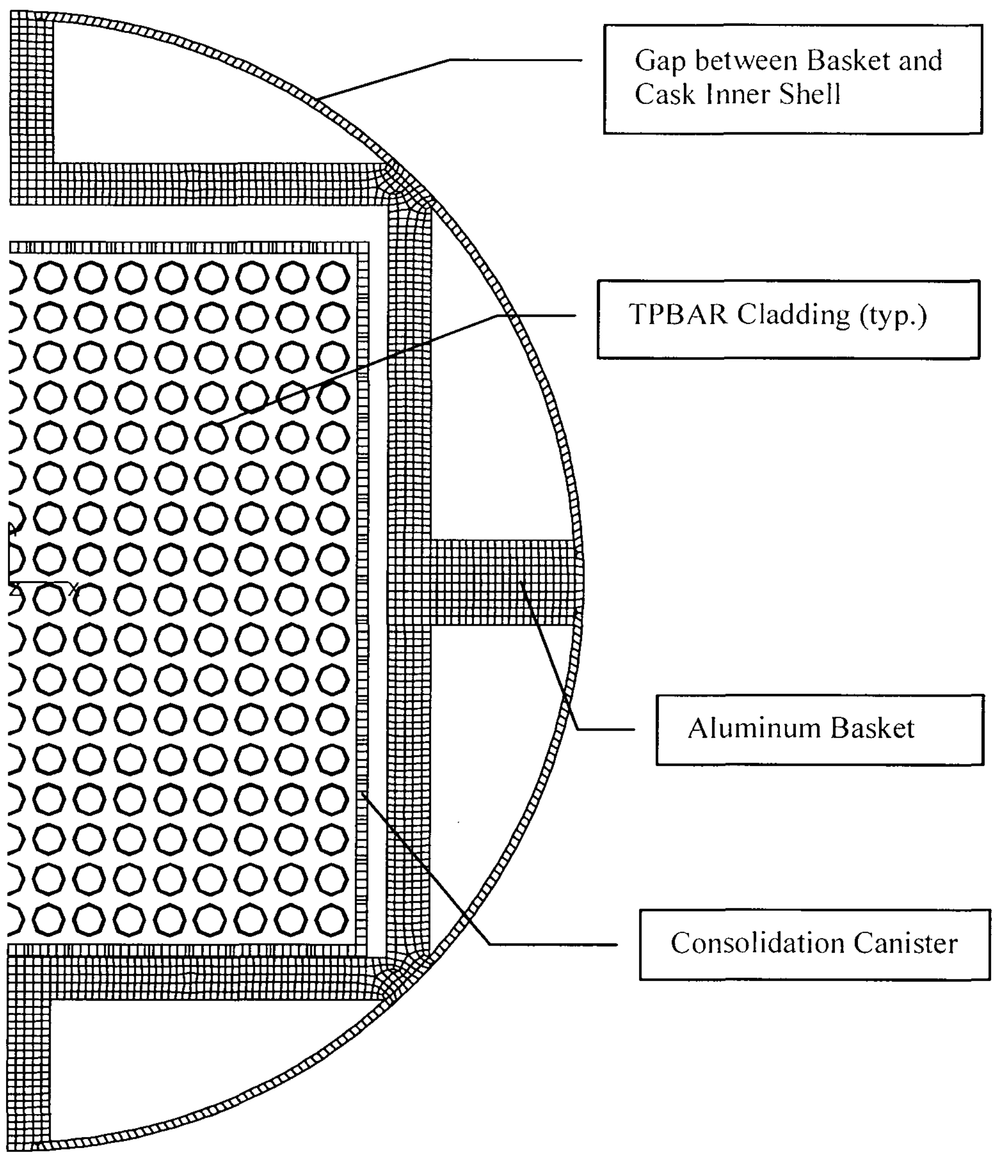


Figure 3.4-18 Finite Element Model for TPBARs



Note: Helium elements, except the gap between the basket and cask inner shell, are not shown for clarity.

Figure 3.4-19 Finite Element Model for MOATA Plate Fuel – ANSTO

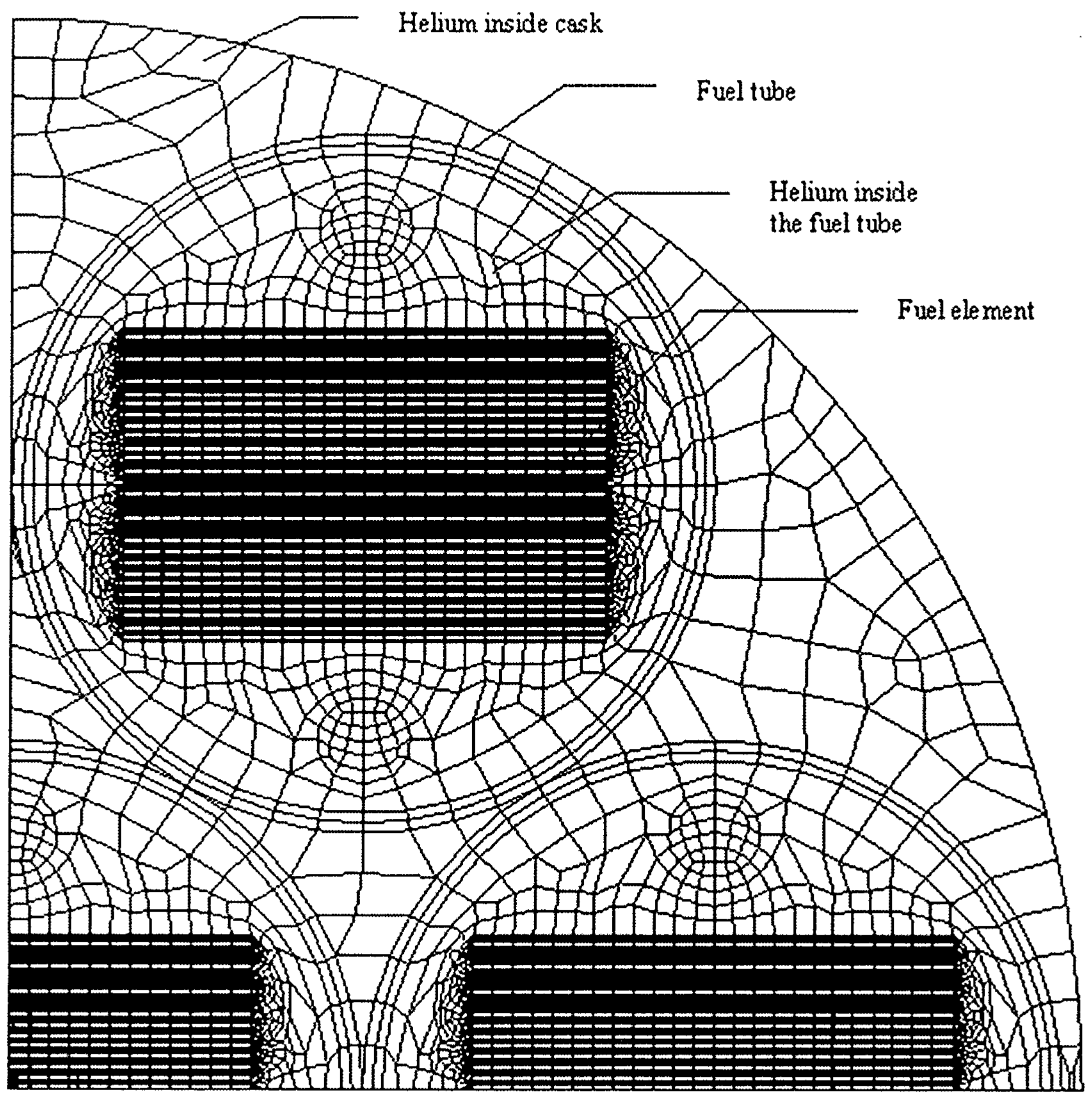


Figure 3.4-20 Finite Element Model for Mark III Spiral Fuel – ANSTO

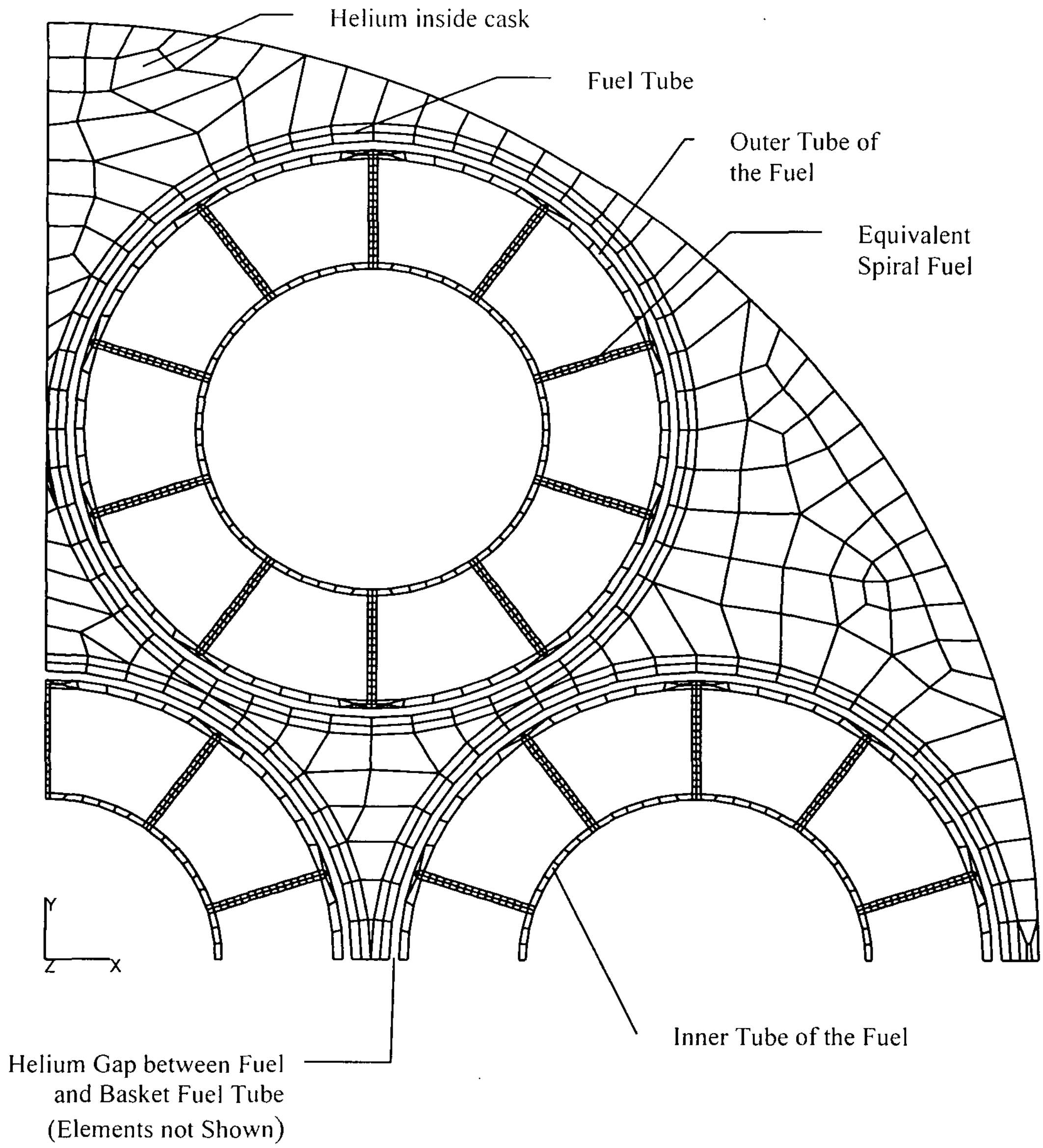


Table 3.4-1 Temperatures for Metallic Fuel Transport

Normal Transport Conditions

Component	Temperature (°F)
O-rings	200
Valves	201
Cask Radial Outer Surface	173
Neutron Shield	252
Radial Lead Gamma Shield	254
Bottom Lead Gamma Shield	210
Inner Stainless Steel Shell	255
Fuel Basket Outer Wall	255
Maximum Fuel Rod Cladding	270

Table 3.4-2 Maximum Component Temperatures – Design Basis PWR Fuel

Normal Transport Conditions

Component	Temperature (°F)
O-rings	227
Valves	231
Cask Radial Outer Surface	229
Neutron Shield	238
Radial Lead Gamma Shield	273
Bottom Lead Gamma Shield	239
Inner Stainless Steel Shell	274
Fuel Basket Outer Wall	276
Maximum Fuel Rod Cladding	472

Table 3.4-3 **Limiting Cold Case Component Temperatures – Design Basis PWR Fuel**

Normal Transport Conditions

Maximum Decay Heat Load, Minimum Ambient

Component	Temperature (°F)
O-rings	124
Valves	129
Cask Radial Outer Surface	128
Neutron Shield	110
Radial Lead Gamma Shield	167
Bottom Lead Gamma Shield	150
Inner Stainless Steel Shell	167
Fuel Basket Outer Wall	170
Maximum Fuel Rod Cladding	336

Table 3.4-4 Fission Product Gas Inventories and Pressures for Design Basis PWR Fuel Assembly

Fission Product	Inventory per Fuel Assembly (moles)	Initial Pressure (psia)
H-3	0.008	0.615
Kr-80	0.000	0.000
Kr-81	0.000	0.000
Kr-82	0.004	0.308
Kr-83	0.234	17.989
Kr-84	0.687	52.814
Kr-85	0.129	9.917
Kr-86	1.060	81.489
I-127	0.167	12.838
Xe-128	0.010	0.769
I-129	0.704	54.121
Xe-129	0.000	0.000
Xe-130	0.032	2.460
Xe-131	1.641	126.154
Xe-132	4.159	319.728
Xe-134	5.679	436.580
Xe-136	8.529	655.678
Total	**23.044**	**1771.5**

Table 3.4-5 NAC-LWT Cask Thermal Performance Summary

Normal Transport Conditions

Component	Minimum Temperature °F	Maximum Temperature °F	Safe Operating Range °F
TFE O-rings	-40	227	-40 to +735[1]
Metallic O-rings	-40	227	-40 to +800
Viton® O-rings	-40	227[2]	-40 to +550[3]
Lead gamma shield	-40	273	-40 to +600
Liquid neutron shield	-40	238	-40 to +350

[1] Verified through testing (Certified Test Report D9-3362-1, Applied Technical Services, Inc., Marietta, GA, February 8, 1989)

[2] Normal Transport Condition maximum O-ring temperatures were not calculated. The Viton® O-rings are located in close proximity to the TFE O-rings and there is substantial thermal margin, a new O-ring temperature is not calculated

[3] Verified through testing (Certified Test Report 43939-01, Wyle Laboratories, Inc., Huntsville, AL, February 21, 2000).

Table 3.4-6 MTR Fuel Maximum Component Temperatures – Normal Transport Condition

Conditions: 100°F Ambient Temperature
Solar Insolation
1.26 Kilowatts Decay Heat Load

Condition 1: NAC-LWT (Transported in an ISO Container)
Cavity gas: Helium

Component	Temperature (°F)	
	Design Basis Decay Heat Load [1]	Variable Decay Heat Load [2]
Liquid Neutron Shield	198	198
Outer Shell	199	199
Lead Gamma Shield	212	214
Inner Shell	214	215
Basket (maximum)	256	292
Fuel (maximum)	< 363 [3]	< 363 [3]

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity gas: Air

Component	Temperature (°F)	
	Design Basis Decay Heat Load [1]	Variable Decay Heat Load [2]
Liquid Neutron Shield	160	160
Outer Shell	161	160
Lead Gamma Shield	180	180
Inner Shell	181	180
Basket (maximum)	267	312
Fuel (maximum)	< 363 [3]	363

[1] Uniform 30-Watt/Element Configuration Heat Load.
[2] 120-Watt / 70-Watt / 20-Watt Configuration Heat Load.
[3] Fuel not modeled for this condition. Fuel temperature is bounded by the variable decay heat load in air case.

Table 3.4-7 PWR Rods (25 Total) Maximum Component Temperatures – Normal Transport Condition

Conditions: 100°F Ambient Temperature

Cask Inside ISO Container

Solar Insolation

1.41 Kilowatts Decay Heat Load

Component	Temperature (°F)
O-rings	< 249
Valves	< 249
Cask Radial Outer Surface	185
Lead Gamma Shield	248
Inner Shell	249
Outer Shell	235
Basket	252
Liquid Neutron Shield	235
Maximum Cladding Temperature	358

Table 3.4-8 TRIGA Fuel Element Maximum Component Temperatures – Normal Conditions of Transport

Conditions: 100°F Ambient Temperature
Solar Insolation
1.05 Kilowatts Decay Heat Load

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity Gas: Air

Component	Temperature (°F)
Liquid Neutron Shield	< 160
Outer Shell	< 161
Lead Gamma Shield	< 180
Inner Shell	< 181
Basket (maximum)	267[1]
Cladding (maximum)	326[1]

[1] As shown in Table 3.4-6, the Condition 2 analysis produces higher basket temperatures than Condition 1. Therefore, the Condition 2 analysis for TRIGA fuel bounds transport of the cask in an ISO container.

Table 3.4-9 TRIGA Fuel Cluster Rod Temperatures – Normal Conditions of Transport

Conditions: 100°F Ambient Temperature
Solar Insolation
1.05 Kilowatts Decay Heat Load

Condition 1: NAC-LWT (Transported in an ISO Container)
Cavity gas: Helium

Component	Temperature (°F)
Liquid Neutron Shield	207
Outer shell	207
Lead Gamma shield	221
Inner shell	222
Basket (maximum)	263
Aluminum insert tube	265
Cladding (maximum)	266

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity gas: Air

Component	Temperature (°F)
Liquid Neutron Shield	159
Outer shell	160
Lead Gamma shield	177
Inner shell	178
Basket (maximum)	278
Aluminum insert tube	292
Cladding (maximum)	295

Table 3.4-10 PWR and BWR High Burnup Fuel Rods Maximum Component Temperatures – Normal Transport Condition

Conditions: 100°F Ambient Temperature Solar Insolation
2.1 Kilowatts Decay Heat Load
Condition 1: NAC-LWT (Transported in an ISO Container)
Cavity Gas: Helium

Component	Component Temperature (°F)	Allowable Temperature (°F)
Liquid Neutron Shield	306	N/A
Outer Shell	308	800
Lead Gamma Shield	375	600
Inner Shell	385	800
Lid Metallic Containment Shell	385	800
Port Cover Containment Shell	385	550
Basket (maximum)	387	800(1)
Cladding (maximum)	671	752(2)
Aluminum PWR Insert	394	700(3)
Stainless Steel Can Weldment	500	800(2)
Average Cavity Gas	506	N/A

(1) Allowable temperatures greater than 800°F for stainless steel can e used provided stress limits in ASME III, Subsection NH, are employed in the stress evaluations.

(2) The maximum allowable temperature under NCT for PWR, BWR and PWR MOX fuel rod cladding is 752°F (400°C) per ISG-11, Revision 3.

(3) The aluminum insert is not a structural component. The primary consideration in establishing the safe operating range of the aluminum is maintaining the integrity of the aluminum. According to MIL-HDBK-5F, it can be shown that aluminum at 700°F retains component performance.

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity Gas: Air

Component	Temperature (°F)
Inner Shell	274
Basket (maximum)	280
Aluminum PWR Insert	286
Stainless Steel Can Weldment	538
Cladding (maximum)	896
Average Cavity Gas	541

Table 3.4-11 Fission Product Gas Inventories and Pressures for the Exxon 7 × 7 BWR Fuel Assembly

Fission Product	Inventory per Fuel Assembly (moles)	Initial Partial Pressure per Rod (psia)
H-3	7.670E-03	1.408E+00
Kr-80	0.000E+00	0.000E+00
Kr-81	0.000E+00	0.000E+00
Kr-82	8.110E-03	1.489E+00
Kr-83	1.270E-01	2.331E+01
Kr-84	7.060E-01	1.296E+02
Kr-85	9.590E-02	1.760E+01
Kr-86	9.330E-01	1.713E+02
I-127	1.770E-01	3.249E+01
Xe-128	3.000E-02	5.507E+00
I-129	7.030E-01	1.290E+02
Xe-129	4.260E-04	7.819E-02
Xe-130	9.040E-02	1.659E+01
Xe-131	9.710E-01	1.782E+02
Xe-132	5.030E+00	9.233E+02
Xe-134	5.690E+00	1.044E+03
Xe-136	8.590E+00	1.577E+03
Total	2.32E+01	4.251E+03

Table 3.4-12 DIDO Fuel Maximum Component Temperatures – Normal Transport Condition

Conditions: 100°F Ambient Temperature
Solar Insolation
1.05 Kilowatts Decay Heat Load

Condition 1: NAC-LWT (Transported in an ISO Container)
Cavity gas: Helium

Component	Temperature (°F)
	Design Basis Decay Heat Load
Liquid Neutron Shield	198[1,2]
Outer Shell	199[1,2]
Lead Gamma Shield	212[1,2]
Inner Shell	214[1,2]
Basket (maximum)	299[3]
Fuel (maximum)	306[3]

1 Uniform 30-Watt/Assembly Configuration Heat Load for MTR fuel.
2 Bounding values obtained from Table 3.4-6 for MTR fuel.
3 Uniform 25-Watt/Assembly Configuration Heat Load for DIDO fuel.

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity gas: Air

Component	Temperature (°F)
	Design Basis Decay Heat Load
Liquid Neutron Shield	160[1,2]
Outer Shell	161[1,2]
Lead Gamma Shield	180[1,2]
Inner Shell	181[1,2]
Basket (maximum)	327[3]
Fuel (maximum)	338[3]

1 Uniform 30-Watt/Assembly Configuration Heat Load for MTR fuel.
2 Bounding values obtained from Table 3.4-6 for MTR fuel.
3 Uniform 25-Watt/Assembly Configuration Heat Load for DIDO fuel.

* The temperatures in the above tables bound the maximum temperatures of the fuel and components in the ANSTO-DIDO baskets presented in Section 3.4.1.16.

Table 3.4-13 General Atomics IFM Maximum Component Temperatures – Normal Transport Condition

Conditions: 100°F Ambient Temperature
Solar Insolation
13 W Decay Heat Load

NAC-LWT (Transported in an ISO Container)

Component	Temperature (°F)
	Design Basis Decay Heat Load
Liquid Neutron Shield	198[1]
Outer Shell	199[1]
Lead Gamma Shield	212[1]
Inner Shell	214[1]
Basket (maximum)	250[2]
FHU contents (maximum)	326[3]

[1] Bounding values obtained from Table 3.4-6 for MTR fuel.
[2] 13-Watt Configuration Heat Load for General Atomics fuel.
[3] Bounding value obtained from Table 3.4-8 for the 1.05 kW TRIGA fuel.

Table 3.4-14 PWR and BWR High Burnup Fuel Rods in a Fuel Assembly Lattice Maximum Component Temperatures—Normal Transport Condition

Conditions: 100°F Ambient Temperature
Solar Insolation
2.1 Kilowatts Decay Heat Load (BWR)
2.3 Kilowatts Decay Heat Load (PWR)
Transport Condition 2 (no ISO container) with air in the cavity

Component	Temperature (°F)
Inner Shell	274
Basket (maximum)	276
Aluminum PWR Insert	336
Cladding (maximum)	664
Average Cavity Gas	430

Table 3.4-15 Maximum Component Temperatures for High Burnup Fuel Rods with Damaged Fuel Rods in a Rod Holder

Case[1]	Maximum Temperatures (°F)					
	Basket	Aluminum Insert	Rod Holder Weldment	Fuel Rod Tube[2]	Fuel Cladding[3]	Cavity Gas Average
Damaged Rods at Locations #4, 5, 6, 7, 8, 9, 10, 11, 12	280	285	523	835	809	479
Damaged Rods at Locations #7, 8, 9, 10, 11, 12, 13, 14, 15	280	286	567	866	653	482
Damaged Rods at Locations #1, 2, 3, 4, 5, 6, 7, 8, 9	280	284	474	743	749	465

[1] See Figure 3.4-17 for fuel rod locations. The nine locations in the half-symmetry model correspond to fifteen actual fuel rod locations.
[2] The structural analysis of the fuel tubes in Section 2.6.7.10.2.3 uses a maximum temperature of 925°F.
[3] Maximum temperatures are reported for intact fuel rods only.

Table 3.4-16 Maximum Component Temperatures for TPBAR Shipment – Normal Conditions of Transport

Component	Temperature (°F)
Liquid Neutron Shield	207[1]
Outer Shell	207[1]
Lead Gamma Shield	221[1]
Inner Shell	222[1]
TPBARs	290
Aluminum Basket	228
Consolidation Canister	245
Gas (average)	246

[1] Cask component temperature conservatively obtained from Table 3.4-9, Condition 1 for TRIGA Fuel Cluster Rod.

Table 3.4-17 Maximum Component Temperatures – PULSTAR Fuel in MTR Basket

Conditions: 100°F Ambient Temperature
Solar Insolation
840 watts Decay Heat Load
(30 watts in Each Basket Cell)

Condition 1: NAC-LWT (Transported in an ISO Container)
Cavity gas: Helium

Component	Temperature (°F)
Liquid Neutron Shield	207
Outer shell	207
Lead Gamma shield	221
Inner shell	222
Basket (maximum)	263
Aluminum insert tube	265
Cladding (maximum)	266

Condition 2: NAC-LWT (Transported via Truck Trailer)
Cavity gas: Air

Component	Temperature (°F)
Liquid Neutron Shield	159
Outer shell	160
Lead Gamma shield	177
Inner shell	178
Basket (maximum)	278
Aluminum insert tube	292
Cladding (maximum)	295

Notes:
1. The temperatures in this table correspond to the temperatures in Table 3.4-9.
2. PULSTAR fuel can (if used) = 295°F (assume same as fuel cladding temperature).

Table 3.4-18 PULSTAR Fuel Dimensions

Description	Value
Fuel Assembly Height (inch)	38
Fuel Assembly Width (inch)	3.15 × 2.74
Active Fuel Region Height (inch)	24.1
Fuel Rod Diameter (inch)	0.47
Fuel Clad Thickness (inch)	0.0185
Fuel Pellet Diameter (inch)	0.423
Rod Length (inch)	26.2
Plenum Length (inch)	0.5
Number of Fuel Rods	25

Table 3.4-19 PULSTAR Payload Volume Summary

Description	Dimension[cm^3]
Fuel Volume (25 Elements)	1,860
Pellet to Clad Volume (25 Elements)	97
PULSTAR Can Free Volume	1,440
PULSTAR Can Total Volume	4,230
Assembly Envelope Volume	5,370
LWT Cavity Volume	409,300
MTR Basket Stack Volume	41,900

Table 3.4-20 PULSTAR Fuel Assembly Fission Product Gas Inventory

Isotope	Moles
^{4}He	2.28E-03
^{3}H	1.30E-04
^{82}Kr	6.32E-05
^{83}Kr	7.03E-03
^{84}Kr	1.66E-02
^{85}Kr	2.41E-03
^{86}Kr	2.81E-02
^{127}I	2.87E-03
^{128}Xe	1.27E-04
^{129}I	1.42E-02
^{130}Xe	3.11E-04
^{131}Xe	3.91E-02
^{132}Xe	8.49E-02
^{134}Xe	1.27E-01
^{136}Xe	1.23E-01
Total	4.48E-01

Table 3.4-21 PULSTAR Fuel Element Normal Condition Internal Pressure Summary

Description	Free Volume (liters)	Pressure (atm)	Pressure (psia)	Pressure (psig)
Cask Pressure -28 Intact Assemblies	217.0	1.4	21.3	6.6
Cask Pressure -14 Intact Assemblies and 14 Cans	233.0	1.8	27.2	12.5
Can Pressure - PULSTAR Failed Fuel Can	1.53	4.4	65.4	50.7

Table 3.4-22 Maximum Component Temperatures – MOATA Plate Fuel and Mark III Spiral Fuel in ANSTO Basket

Conditions: 100°F Ambient Temperature

Solar Insolation

Heat Load: 126 Watts – MOATA Plate Fuel; 756 Watts – Mark III Spiral Fuel

Component	Temperature (°F)
Liquid Neutron Shield[1]	207
Outer Shell[1]	207
Lead Gamma Shield[1]	221
Inner Shell[1]	222
Basket - MOATA Plate Fuel	230
Fuel Cladding – MOATA Plate Fuel	233
Basket – Mark III Spiral Fuel	248*
Fuel Cladding – Mark III Spiral Fuel	250*

[1] The cask component temperatures are conservatively obtain from Table 3.4-9 for the TRIGA Fuel Cluster Rod.

* These maximum temperatures are for both standard loading (all seven tubes are loaded with Mark III fuel) and the combination of the Mark III fuel or MOATA fuel.

Table 3.4-23 Cask and Rod Condition

Variable	Unit	DIDO/ANSTO	MTR
Cask Free Volume	liter	368.1	229.3
Normal Condition Average Gas Temperature	K	433.2	470.2
Accident Condition Fuel Temperature[1]	K	528.2	518.2
Cask Backfill Pressure (assumed at STD temp)	psia	14.7	
Normal Condition Cask Pressure Permitted	psig	50	

[1] Conservatively applied to the cask cavity gas.

Table 3.4-24 Gas Isotopics (Gram)

		ANSTO	DIDO			MTR		
Isotope		Spiral	LEU	MEU	HEU	LEU[1]	MEU	HEU[2]
	g/mole	gram						
h 3	3.016	1.30E-04	1.60E-04	1.59E-04	1.58E-04	3.68E-04	3.72E-04	2.93E-04
i127	126.904	7.59E-02	9.91E-02	9.35E-02	8.85E-02	3.05E-01	2.54E-01	2.02E-01
i129	129.000	4.37E-01	5.39E-01	5.27E-01	5.16E-01	1.58E+00	1.33E+00	1.24E+00
kr 82	81.913	5.59E-04	6.03E-04	5.94E-04	5.89E-04	4.28E-03	4.80E-03	1.07E-02
kr 83	82.914	1.55E-01	1.86E-01	1.85E-01	1.84E-01	4.76E-01	3.23E-01	3.95E-01
kr 84	83.912	4.69E-01	5.37E-01	5.49E-01	5.59E-01	1.48E+00	1.49E+00	1.33E+00
kr 85	84.913	8.78E-02	1.05E-01	1.06E-01	1.08E-01	2.16E-01	2.31E-01	1.85E-01
kr 86	85.911	7.85E-01	9.08E-01	9.23E-01	9.36E-01	2.44E+00	2.26E+00	2.17E+00
xe128	127.904	1.16E-03	1.21E-03	1.21E-03	1.21E-03	1.20E-02	1.26E-02	2.96E-02
xe130	129.904	1.79E-02	1.91E-02	2.02E-02	2.14E-02	7.54E-02	1.21E-01	8.45E-02
xe131	130.905	1.51E+00	1.85E+00	1.82E+00	1.80E+00	4.57E+00	3.55E+00	2.46E+00
xe132	131.904	2.93E+00	3.46E+00	3.46E+00	3.46E+00	1.05E+01	9.99E+00	1.02E+01
xe134	133.905	4.93E+00	5.82E+00	5.84E+00	5.85E+00	1.62E+01	1.48E+01	1.39E+01
xe136	135.907	7.80E+00	9.19E+00	9.19E+00	9.20E+00	2.04E+01	2.30E+01	1.95E+01

[1] 470 gram ^{235}U per element
[2] 460 gram ^{235}U per element

Table 3.4-25 Molar Gas Quantity

Isotope	ANSTO Spiral [mole]	DIDO LEU [mole]	DIDO MEU [mole]	DIDO HEU [mole]	MTR LEU [mole]	MTR MEU [mole]	MTR HEU [mole]
h 3	4.31E-05	5.30E-05	5.27E-05	5.24E-05	1.22E-04	1.23E-04	9.71E-05
i127	5.98E-04	7.81E-04	7.37E-04	6.97E-04	2.40E-03	2.00E-03	1.59E-03
i129	3.39E-03	4.18E-03	4.09E-03	4.00E-03	1.22E-02	1.03E-02	9.61E-03
kr 82	6.82E-06	7.36E-06	7.25E-06	7.19E-06	5.23E-05	5.86E-05	1.31E-04
kr 83	1.87E-03	2.24E-03	2.23E-03	2.22E-03	5.74E-03	3.90E-03	4.76E-03
kr 84	5.59E-03	6.40E-03	6.54E-03	6.66E-03	1.76E-02	1.78E-02	1.59E-02
kr 85	1.03E-03	1.24E-03	1.25E-03	1.27E-03	2.54E-03	2.72E-03	2.18E-03
kr 86	9.14E-03	1.06E-02	1.07E-02	1.09E-02	2.84E-02	2.63E-02	2.53E-02
xe128	9.07E-06	9.46E-06	9.46E-06	9.46E-06	9.38E-05	9.85E-05	2.31E-04
xe130	1.38E-04	1.47E-04	1.56E-04	1.65E-04	5.80E-04	9.31E-04	6.50E-04
xe131	1.15E-02	1.41E-02	1.39E-02	1.38E-02	3.49E-02	2.71E-02	1.88E-02
xe132	2.22E-02	2.62E-02	2.62E-02	2.62E-02	7.96E-02	7.57E-02	7.73E-02
xe134	3.68E-02	4.35E-02	4.36E-02	4.37E-02	1.21E-01	1.11E-01	1.04E-01
xe136	5.74E-02	6.76E-02	6.76E-02	6.77E-02	1.50E-01	1.69E-01	1.43E-01
Total	1.50E-01	1.77E-01	1.77E-01	1.77E-01	4.55E-01	4.47E-01	4.04E-01

Table 3.4-26 Maximum Cask Cavity Pressure

Release Fraction	ANSTO Spiral Fission Gas	ANSTO Spiral Total		DIDO Fission Gas	DIDO Total		MTR Fission Gas	MTR Total	
	psi	psia	psig	psi	psi	psig	psi	psia	psig
1%	0.1	23.9	9.2	0.1	23.9	9.2	0.5	26.3	11.6
5%	0.4	24.2	9.5	0.5	24.3	9.6	2.4	28.2	13.5
10.0%	0.9	24.7	10.0	1.1	24.8	10.1	4.7	30.6	15.9
20.0%	1.8	25.6	10.9	2.1	25.9	11.2	9.5	35.3	20.6
30.0%	2.7	26.5	11.8	3.2	27.0	12.3	14.2	40.0	25.3
40.0%	3.6	27.4	12.7	4.2	28.0	13.3	18.9	44.7	30.0
50.0%	4.5	28.3	13.6	5.3	29.1	14.4	23.7	49.5	34.8
60.0%	5.4	29.1	14.4	6.3	30.1	15.4	28.4	54.2	39.5
70.0%	6.3	30.0	15.3	7.4	31.2	16.5	33.1	58.9	44.2
80.0%	7.1	30.9	16.2	8.4	32.2	17.5	37.9	63.7	49.0
90.0%	8.0	31.8	17.1	9.5	33.3	18.6	42.6	68.4	53.7
100.0%	8.9	32.7	18.0	10.6	34.3	19.6	47.3	73.1	58.4

Table 3.4-27 Maximum Component Temperatures – SLOWPOKE Fuel in MTR Basket

Conditions: 100°F Ambient Temperature
Solar Insolance
5.0 Watts Decay Heat Load (2.5 Watts per basket module)
(Transported in an ISO Container)
Cavity gas: Helium

Component	Max. Temperature (°F)
Liquid Neutron Shield	137
Outer shell	136
Lead Gamma shield	134
Inner shell	134
Basket	135
ISO Container	166
Fuel	136

configurations for the ANSTO-DIDO basket assembly. Therefore, for the fire accident condition, the same basis would confirm that the fuel and component temperatures for the fire accident conditions evaluated in Section 3.5.3.6 for DIDO fuel and Section 3.5.3.12 for ANSTO fuel types are bounding for the maximum fuel temperature in the conditions of the ANSTO-DIDO basket assembly.

3.5.3.15 Evaluation for TPBARs in the PWR/BWR Rod Transport Canister

Similar to Section 3.5.3.10, the maximum heat load of TPBARs in the Rod Transport Canister is bounded by the maximum heat load of the MTR fuel. Therefore, in the accident condition, the maximum temperatures of the cask components for the MTR contents will bound the maximum temperatures for the cask components for the TPBAR contents. It is conservative to use the results of the fire transient evaluated in Section 3.5.3.2 for the cask inner shell temperature. The maximum content temperature (T_{max}) for the TPBARs in a Rod Transport Canister for the accident conditions are determined by adding the temperature difference ($\Delta T_{component}$) between the cask inner shell and the maximum component temperature for normal conditions to the maximum accident cask inner shell temperature ($T_{inner\ shell}$) obtained from the MTR evaluation. The maximum component temperature is computed as follows.

Component	$\Delta T_{component}$[1] (°F)	$T_{inner\ shell}$[2] (°F)	T_{max} (°F)
TPBAR	14	334	348

[1] See Section 3.4.1.17
[2] See Table 3.5-2

The computed maximum contents temperature is less than for the TPBAR analysis of Section 3.5.3.10. Therefore, the component temperatures, namely, the aluminum basket and TPBARs, for the fire accident condition are bounded by the results of Section 3.5.3.10.

3.5.3.16 Evaluation of SLOWPOKE Fuel Contents

The SLOWPOKE fuel maximum heat load is bounded by the maximum heat load of the MTR fuel. Therefore, in the accident condition, the maximum component temperatures for the MTR contents will bound the maximum component temperatures for the components for the SLOWPOKE contents. The maximum component temperatures for the fire accident for the MTR contents are listed in Table 3.5-2 for the design heat load.

3.5.4 Maximum Internal Pressure

3.5.4.1 Maximum Internal Pressure for Design Basis Fuel in Accident Conditions

The accident internal pressure is calculated assuming an accident with 100 percent fuel rod failure combined with the design basis fire described in 10 CFR 71. The fuel rod failure assumes 30 percent of the fission gas and 100 percent of the backfill gas escapes the ruptured fuel rods.

The internal pressure due to the 100 percent fuel rod rupture is calculated using the method described in Section 3.4.4. The total cask pressure of the cask backfill and failed fuel rods is calculated by a two step procedure. First, the pressures documented under normal conditions in Section 3.4.4 are adjusted to include the increased total free volume associated with 100% fuel rod failure. Then, the revised cask pressure at normal operating temperature is adjusted to accident condition temperatures.

Adjusting the partial pressure of the cask backfill:

$$P_{cask} = P_{initial}\left(\frac{V_{cask}}{V_{total}}\right)$$

where:

$P_{initial}$ = 25.8 psia (normal condition temperature adjusted cask backfill pressure)

V_{cask} = 5.196 ft^3 (147,134 cm^3) [Section 3.4.4]

$V_{rod\ void}$ = 8,123 cm^3 [Section 3.4.4]

V_{total} = 155,257 cm^3 (V_{cask} + $V_{rod\ void}$)

$$P_{cask} = 25.8\text{ psia}\left(\frac{147{,}134\text{cm}^3}{155{,}257\text{cm}^3}\right)$$

$$P_{cask} = 24.4\text{ psia}$$

Adjusting the partial pressure of the fuel rod backfill and fission gases:

$$P_{fuel\,rods} = P_{initial}\left(\frac{V_{rod\,void}}{V_{total}}\right)$$

where:

$P_{initial}$ = 1,521.3 psia (fuel rod backfill pressure of 989.8 psia plus fission gas pressure of 0.30 x 1771.5 psia)

$V_{rod\ void}$ = 8,123 cm^3

V_{total} = 155,257 cm^3

$$P_{\text{fuel rods}} = 1{,}521.3\text{ psia}\left(\frac{8{,}123.28\text{cm}^3}{155{,}257\text{cm}^3}\right)$$

$$P_{\text{fuel rods}} = 79.6\text{ psia}$$

Summing the two partial pressures yields the total cask pressure at normal operating condition temperature:

$$P_{\text{Total}} = P_{\text{cask}} + P_{\text{fuel rods}}$$

$$P_{\text{Total}} = 24.4\text{ psia} + 79.6\text{ psia}$$

$$P_{\text{Total}} = 104.0\text{ psia}$$

The fuel cladding has the highest temperature of any barrier with which the gas comes in contact during a design basis fire. As shown in Section 3.5.3.1, the maximum average cavity gas temperature is 605°F during the fire accident condition. For conservatism, a temperature of 667°F is used in the calculation of the maximum accident condition internal pressure. Given that the internal volume of the NAC-LWT Cask remains constant during the fire, the resultant pressure is proportional to the temperature change according to the ideal gas law:

$$P_2 = P_1\left(\frac{T_2}{T_1}\right)$$

Thus, for the design basis fire:

$$P_{\text{fire}} = 104.0\text{ psia}\left(\frac{1127^\circ\text{R}}{932^\circ\text{R}}\right)$$

$$P_{\text{fire}} = 125.8\text{ psia}$$

3.5.4.2 High Burnup Fuel Rod Canister Maximum Internal Pressure

The high burnup fuel rod canister maximum internal pressure in the accident conditions is calculated assuming 100 percent fuel rod failure combined with the design basis fire maximum temperature. The fuel rod failure assumes release of 30 percent of the fission gas and 100 percent of the backfill gas.

The canister internal pressure is calculated using the method described in Section 3.4.4.2, with the BWR used as the bounding fuel type for the analysis. The total canister pressure is calculated in two steps. First, the pressures documented under normal conditions in Section 3.4.4.2 are adjusted to include the increased total free volume associated with 100 percent fuel

rod failure. Then, the canister pressure is adjusted to account for the accident condition temperature.

The partial pressure of the canister volume is calculated by:

$$P_{canister} = P_{initial}\left(\frac{V_{canister}}{V_{total}}\right)$$

where:

P_{inital} = 29.3 psia (from earlier)

$V_{canister}$ = 28.2 liters (from earlier)

V_{void} = 0.079 liters (from earlier)

V_{void} = $25^{*}V_{void} + V_{canister} = 30.2$ liters

V_{total} = 30.2 liters

Therefore, $P_{canister}$ is equal to 27.4 psia. The partial pressure of the fuel rods is calculated by:

$$P_{fuel\ rods} = P_{initial}\left(\frac{V_{fuel\ rods}}{V_{total}}\right)$$

where:

$P_{initial}$ = 1,425 psia (earlier from Section 3.4.4.2)

$V_{fuel\ rods}$ = $25^{*}V_{void}$

$V_{fuel\ rods}$ = ~1.97 liters (at 100% of the total fuel rod volume)

V_{total} = 30.2 liters ($V_{canister} + (25^{*}V_{void})$)

then:

$$P_{fuel\ rods} = 1{,}425\ \text{psia}\left(\frac{1.97\ \text{liters}}{30.2\ \text{liters}}\right) = \sim 94\ \text{psia}$$

and

$$P_{total} = P_{canister} + P_{fuel\ rods} = 27.4\ \text{psia} + \sim 94\ \text{psia} = \sim 121\ \text{psia}\ (\sim 8.2\ \text{atm})$$

For the 100% fuel rod failure and the design basis fire accident temperature of 725°F, the pressure is calculated by multiplying the 100% rod failure pressure by the inverse ratio of the normal condition temperature (588.7 K) to the accident temperature (658.15 K). The pressure thus calculated is 135 psia (~9.2 atm)

3.5.4.3 25-Rod Maximum Internal Pressure-Cask Cavity

Using the same methodology used to calculate the cavity pressure in Section 3.5.4.2, the pressure from the 100% fuel rod failure and the design basis fire accident temperature of 725°F is calculated using the cask cavity free gas volume (89.32 liters from earlier). The resulting pressure in the cask cavity, assuming that the gases within the canister are released to the cask cavity, is 67 psia (~4.5 atm).

3.5.4.4 TPBAR Shipment Cask Cavity Internal Pressure-Accident Conditions

Employing the normal condition TPBAR result in Section 3.4.4.5 of 276 psig for the 300 production TPBAR content condition and adjusting system pressure to the average accident gas temperature of 358°F yields a maximum accident condition pressure of 322 psig. For a period of one year following the 90-day cooldown, the pressure for this content condition increases to 337 psig. As discussed in Section 3.4.4.5, these values bound those of the up to 25 TPBAR transport configuration within the 5×5 rod holder. The rod holder combination contains significantly lower releasable gas quantities at similar free volume.

Utilizing the same assumptions as presented in Section 3.4.4.5 and the post-accident thermal conditions discussed above, the pressure for the 55 segmented TPBARs in the waste container will be less than 75 psia and, therefore, bounded by the 300 TPBAR content condition.

3.5.4.5 Maximum Internal Pressure for PULSTAR Fuel Payload

Maximum internal pressures under accident conditions are calculated using the same methodology as that employed in Section 3.4.4.6. The accident condition temperature is set to 394°F, and 100 percent of the fuel rods are assumed to fail. The resulting calculated pressures are summarized as follows.

Description	Free Volume	Pressure		
	(liters)	(atm)	(psia)	(psig)
Cask Pressure -28 Intact Assemblies	217.0	2.3	34.0	19.3
Cask Pressure -14 Intact Assemblies and 14 Cans	233.0	2.4	35.4	20.7
Can Pressure - PULSTAR Failed Fuel Can	1.53	5.0	73.9	59.2

3.5.4.6 Maximum Internal Pressure for 16 PWR MOX/UO_2 Fuel Rods in a Rod Holder

Using the same methodology used to calculate the cavity pressure in Section 3.4.4.4, the pressure from the 100% fuel rod failure with 100% gas release and the design basis fire accident temperature of 725°F is calculated. The resulting pressure in the cask cavity is 65.3 psig (80.0 psia, 5.4 atm).

3.5.4.7 Maximum Internal Pressure for Aluminum-Based Fuels

The maximum normal condition 100% fission gas release MTR payload is evaluated for accident pressure. This payload bounds the remaining aluminum based fuel payloads. The 100% release normal condition pressure in Section 3.4.4.9 is simply increased by the ratio of accident (528K) to normal (470K) temperature.

$$P_{Accident} = P_{Normal} \times \frac{T_{Accident}}{T_{Normals}} = 73.1\text{psia} \times \frac{528\text{K}}{470\text{K}} = 82.1\text{psia} = 67.4\text{psig}$$

As the DIDO accident temperature is higher than the MTR temperature, the DIDO value is used as the ratio basis. Maximum accident pressure for aluminum-based fuel is conservatively calculated to be 68 psig.

3.5.5 Maximum Thermal Stresses

The most severe thermal stress conditions that occur during the fire test and subsequent cooldown have been evaluated. For conservatism, an internal pressure of 168 psig is used, in the analysis that is performed in Section 2.7.3. The temperatures corresponding to the maximum thermal stresses are reported in Table 3.5-1.

3.5.6 Evaluation of Package Performance for Hypothetical Accident Thermal Conditions

The NAC-LWT cask thermal performance has been assessed for the hypothetical accident, as specified in 10 CFR 71. The O-rings and the lead gamma shields remain within their safe operating ranges. The cask does not suffer any adverse structural consequences as a result of the thermal considerations of the hypothetical accident. The NAC-LWT cask maintains containment and does not exceed the dose rate limits of 49 CFR 173 as a result of the hypothetical accident.

3.5.7 Assessment of the Effects of the Fission Gas Release in the Fire Accident Condition

During the fire, the release of the fission gas is expected to reduce the effective thermal conductivity of the gas in the cavity or inside the sealed canisters. To assess the reduction of the thermal conductivity, the helium conductivity is factored by the ratio of the conservative initial fill pressure of 565 psia (Section 3.4.4) for the PWR fuel assemblies and the end of life pressure (which contains the fill gas plus the fission gas release) of 1,521 psia (Section 3.5.4). This ratio is computed to be 0.37. A conservative ratio of 0.24 is applied to the conductivity of helium, assuming that all fission product gases have a conductivity of zero.

For the temperatures shown, which envelope the maximum temperatures of the cavity gas in the accident condition, the reduced helium properties are larger than the thermal conductivity of air.

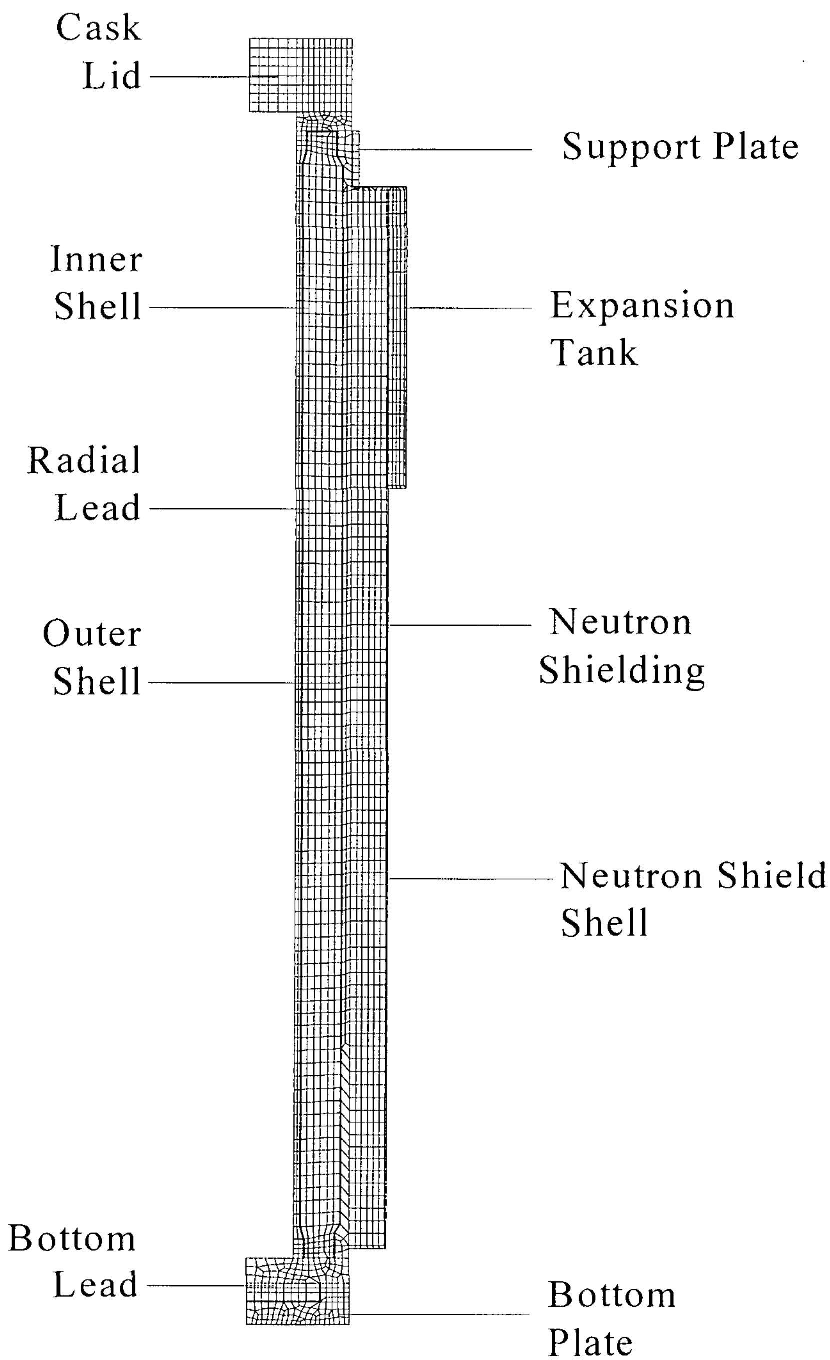
This is bounding because, as shown in Table 4.2-2, the volume of fission product gas produced by the design basis PWR assembly is higher than that for any other fuel loading.

The data below (Krieth) reflects the comparison of the air conductivity and the factored helium conductivity.

Temperature (°F)	Air Conductivity (K_{air}) (Btu/hr-in-F)	Helium Conductivity (Btu/hr-in-F)	Factored Helium Conductivity (K_{He}) (Btu/hr-in-F)	Ratio K_{He}/K_{air}
300	0.00161	0.00883	0.00212	1.32
400	0.00177	0.00958	0.00230	1.30
500	0.00193	0.01017	0.00244	1.26
600	0.00208	0.01075	0.00258	1.24
700	0.00223	0.01113	0.00267	1.20
800	0.00238	0.01150	0.00276	1.16

The analyses performed for the contents employed air as the gas in the cavity and containers for the accident condition. This demonstrates that the evaluation of the accident condition using air bounds the "reduced helium properties" case.

Figure 3.5-1 Transient Thermal Analysis Finite Element Model of the NAC-LWT



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5.3.20 SLOWPOKE Fuel Configuration

Results of a shielding analysis for up to 800 fuel rods in the LWT cask are presented in this section. Maximum dose rates are calculated to demonstrate that dose rate limits of 10 CFR 71.47 are not exceeded.

Dose rates are calculated using the MCNP three-dimensional transport code. Source terms are calculated using the TRITON/NEWT module of the SCALE package.

5.3.20.1 SLOWPOKE Fuel Source Term

Source terms are calculated to bound the irradiation history of the SLOWPOKE fuel rods. Fuel rod characteristics are summarized in Table 5.3.20-1. Inputs for irradiation and material parameters required by TRITON are given in Table 5.3.20-2. Key parameters differing between the input and analysis are reduced enrichment, increased fuel mass, and increased irradiation time. All parameters revised to produce bounding source terms.

TRITON input is shown in Figure 5.3.20-3, with the resulting TRITON material model shown in Figure 5.3.20-2. Neutron and gamma source terms for a cool time of 14 years from discharge are presented in Table 5.3.20-3 and Table 5.3.20-4, respectively. The calculated heat load at this cool time is 0.0027 W/rod or 2.17 W/cask (800 rods).

The effect of subcritical neutron multiplication is directly computed in the MCNP analysis.

5.3.20.2 SLOWPOKE Fuel Shielding Model

MCNP three-dimensional shielding analysis allows detailed modeling of the fuel, basket, and cask shield configurations. Some fuel rod detail is homogenized in the model to simplify model input and improve computational efficiency. The basket and cask body details are explicitly modeled, including the axial extents described by the License Drawings.

The geometric description of a MCNP model is based on the combinatorial geometry system embedded in the code. In this system, bodies such as cylinders and rectangular parallelepipeds, and their logical intersections and unions, are used to describe the extent of material zones.

Source and Cansiter Models

Options for loading include fuel rod arrays of 4x4 and 5x5 rods. Only the 5x5 array is modeled as it contains maximum fuel/source inventory. These arrays are stacked four high within a canister that also contains a handle. The canister is made of aluminum. Dimensions for the tube array and canister are shown in Table 5.3.20-7. The source region is modeled as a smear within the canister tubes. The fuel homogenization, shown in Table 5.3.20-5, is based on an area bounded by the aluminum tube. The source height is the active fuel height, 22 cm.

While the fuel rods may be damaged, the results of this model will bound both normal and accident conditions. Aluminum metallic fuel, even when damaged, will not disperse through the canister. The material is also placed into individual tubes which will retain larger fuel sections. Shifts in the material within the canister will also be well bounded by having shifted the canister and payload.

Cross section of the VISED model of the source region are shown in Figure 5.3.20-4 and Figure 5.3.20-5. As shown, the model is moved to its maximum axial elevation which brings it closest to the reduced shielding area of the NAC-LWT. The lowest shielding region is the tapered area of the lead gamma neutron shield, the area below the cask cavity top with no lead shielding.

Basket Model

For a given fuel type, the MCNP description of the basket stack forms a common sub-model employed in the analysis. The key features of the model are the detailed representation of the basket structural members, base plates, and support plates. Basket models are identical to those described in Section 5.3.14. Only four of the basket openings are loaded with SLOWPOKE fuel and only the top two baskets are loaded. The lower two baskets are modeled as void, conservatively removing shielding material and increasing dose rates. Maximum of eight canisters per cask.

MCNP NAC-LWT Model

The three-dimensional model of the NAC-LWT cask is based on the following features.

Normal conditions:

- Radial neutron shield and shield shell
- Aluminum impact limiters with 0.5 g/cm^3 density (calculated based on the impact limiter weight and dimensions) and a diameter equal to the neutron shield shell diameter

Accident conditions:

- Removal of radial neutron shield and shield shell
- Loss of upper and lower impact limiters

Common to both the normal and accident conditions models is a 0.1374 cm gap between the lead outer diameter and the cask outer shell. As stated previously, the elevation of the source regions is set at its maximum axial extent. Detailed model parameters used in creating the three-dimensional model are taken directly from the License Drawings. Elevations associated with the three-dimensional features are established with respect to the center bottom of the NAC-LWT cask cavity for the MCNP combinatorial model. The cask model is identical to that described in Section 5.3.14. A sample input file is provided in Figure 5.3.14-5.

Shield Regional Densities

Based on the homogenization described for the source, the resulting fuel regional densities are shown in Table 5.3.20-5. Material compositions for structural and shield materials are shown in Table 5.3.20-6.

5.3.20.3 SLOWPOKE Fuel Shielding Evaluation

Calculational Methods

The shielding evaluation is performed using MCNP.

The MCNP shielding model described in Section 5.3.20.2 is utilized with the source terms described in Section 5.3.20.1 to estimate the dose rate profiles at various distances from the side, top and bottom of the cask for both normal and accident conditions. The method of solution is continuous energy Monte Carlo with a Monte Carlo based weight window generator to accelerate code convergence. Weight window and problem convergence is verified by the 10 statistical checks performed by MCNP. Radial or axial biasing is performed depending on the desired dose location.

Significant validation literature is available for MCNP as it is an industry standard tool for spent fuel cask evaluations. Available literature covers a range of shielding penetration problems ranging from slab geometry to spent fuel cask geometries. Confirmatory calculations against other validated shielding codes (SCALE and MCBEND) on NAC casks have further validated the use of MCNP for shielding evaluations.

MCNP Flux-to-Dose Conversion Factors

The ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors are employed in the MCNP analysis.

Three-Dimensional Dose Rates for SLOWPOKE Fuel

Table 5.3.20-8 provides maximum and average dose rates for the tabulated distances and transport conditions (normal and accident). Table 5.3.20-9 contains key results.

Calculated normal condition radial surface dose rates are below 200 mrem/hr, therefore do not require an exclusive use designation for the NAC-LWT. The maximum dose rate is dominated by the gamma component. The radial surface dose rate profile is shown in Figure 5.3.20-7. The normal condition maximum radial 2-meter dose rate is 0.002 mrem/hr. The dose rate profile is skewed towards the top of the cask, as shown Figure 5.3.20-8.

Accident condition radial 1-meter dose rates are well below the 1,000 mrem/hr limit. The dose rate profile is shown in Figure 5.3.20-9.

As shown in the dose summary table (Table 5.3.20-9), axial surface dose rates are well below limits for all three source models. Significant margin is present for the normal condition 2-meter and accident condition 1-meter dose rate limits.

Figure 5.3.20-1 SLOWPOKE Fuel Element

Figure Withheld Under 10 CFR 2.390

Figure 5.3.20-2 SLOWPOKE Core Model

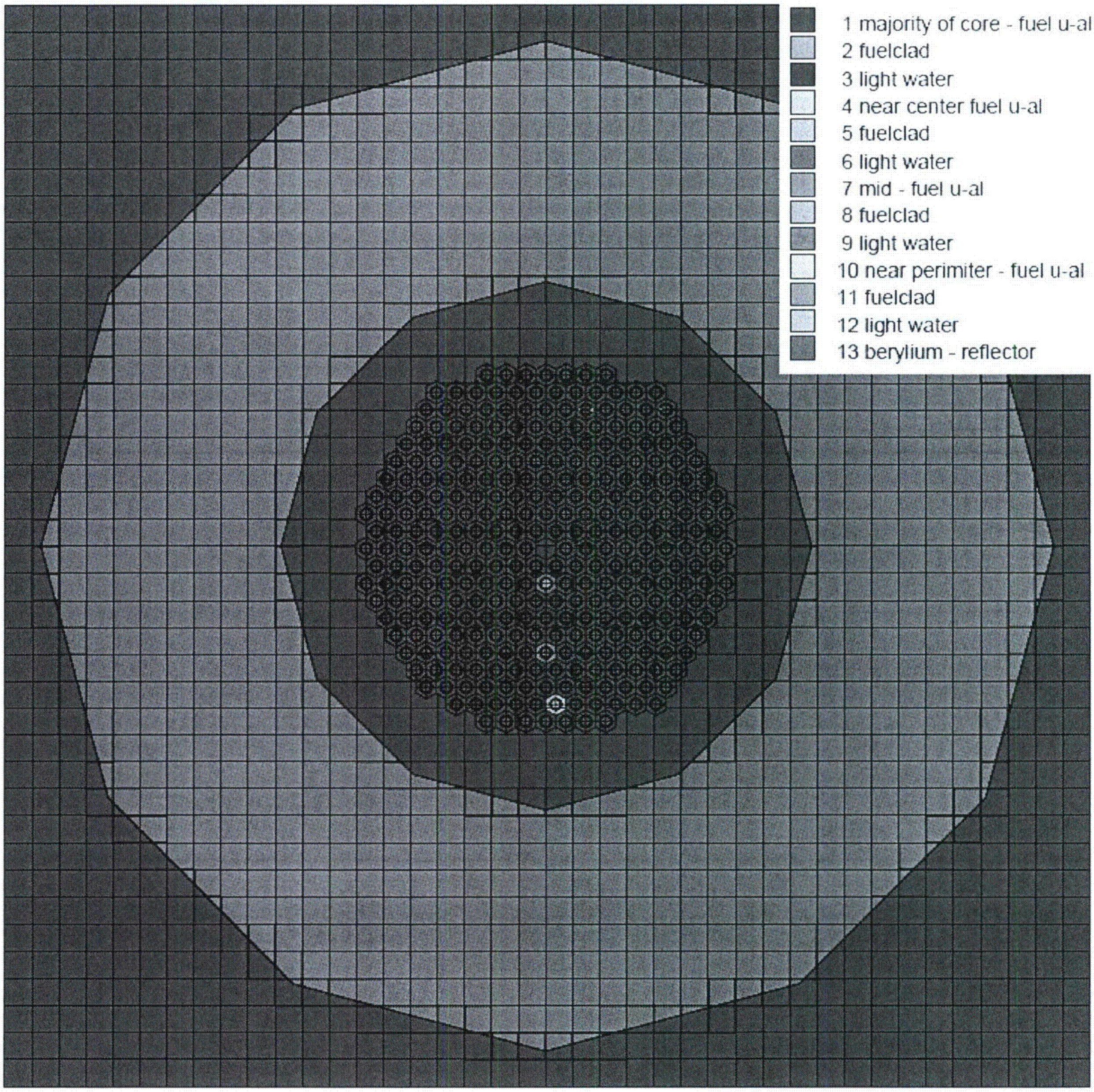


Figure 5.3.20-3 TRITON Input for SLOWPOKE Fuel

```
=t-depl
SLOWPOKE CORE NEWT / CENTRM Depletion - 0.85 cm Rod Pitch - 30 GWD/MTU
V7-238
read comp
U    1 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL   1 DEN=3.51 0.712 373.0  END
AL   2  1.0  363.0  END
H2O  3  1.0  313.0  END
U    4 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL   4 DEN=3.51 0.712 373.0  END
AL   5  1.0  363.0  END
H2O  6  1.0  313.0  END
U    7 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL   7 DEN=3.51 0.712 373.0  END
AL   8  1.0  363.0  END
H2O  9  1.0  313.0  END
U   10 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL  10 DEN=3.51 0.712 373.0  END
AL  11  1.0  363.0  END
H2O 12  1.0  313.0  END
BE   13  1.0  313.0  END
end comp
read celldata
latticecell triangpitch pitch=0.85 3 fueld=0.422 1 cladd=0.524 2 end
latticecell triangpitch pitch=0.85 6 fueld=0.422 4 cladd=0.524 5 end
latticecell triangpitch pitch=0.85 9 fueld=0.422 7 cladd=0.524 8 end
latticecell triangpitch pitch=0.85 12 fueld=0.422 10 cladd=0.524 11 end
end celldata
read depletion 1 4 7 10 end depletion
read opus
matl= 1 4 7 10 0 end units=grams
new case
units=watts
new case
typarams=gspectrum
units=part
new case
typarams=nspectrum
units=parts
end opus
read burndata
' 980 gram fuel - 20kW/Core  - Core Diameter 22 cm  - 7.8 l Water in Core
power=20 burn=1470 down=5100 end
end burndata
read model
SLOWPOKE 315 Rod Assembly - Berylium Reflector - Collapse 44-group
read parm
  prtflux=no  drawit=yes   collapse=yes
  xnlib=4 run=yes prtmxsec=no prtbroad=no
  prtmxtab=yes cmfd=no echo=yes
end parm
read materials
  1 1 !majority of core - fuel u-al! end
  2 1 !fuelclad! end
  3 2 !light water! end
  4 1 !near center fuel u-al! end
  5 1 !fuelclad! end
  6 2 !light water! end
  7 1 !mid - fuel u-al! end
  8 1 !fuelclad! end
  9 2 !light water! end
  10 1 !near perimiter - fuel u-al! end
  11 1 !fuelclad! end
  12 2 !light water! end
  13 2 !berylium - reflector! end
end materials
read collapse
```

```
7r1 2 3 2r4 5 6 7 8 8 8r9 14r10 6r11 10r12 13 7r14 11r15 12r16 30r17 16r18 2r19
6r20 3r21 6r22 14r23 3r24 5r25 4r26 5r27 5r28 5r29 10r30 5r31 32 33 34 2r35
36 37 38 2r39 2r40 3r41 2r42 43 44 45 46 47 3r48 9r49 end collapse
read geom
' Balance Core
unit 1
cylinder 10 0.2011
cylinder 20 0.262
hexprism 30 0.425
media 1 1 10
media 2 1 20 -10
media 3 1 30 -20
boundary 30 2 2
' Near Center
unit 2
cylinder 10 0.2011
cylinder 20 0.262
hexprism 30 0.425
media 4 1 10
media 5 1 20 -10
media 6 1 30 -20
boundary 30 2 2
' Mid Range
unit 3
cylinder 10 0.2011
cylinder 20 0.262
hexprism 30 0.425
media 7 1 10
media 8 1 20 -10
media 9 1 30 -20
boundary 30 2 2
' Near Perimeter
unit 4
cylinder 10 0.2011
cylinder 20 0.262
hexprism 30 0.425
media 10 1 10
media 11 1 20 -10
media 12 1 30 -20
boundary 30 2 2
global unit 10
cylinder 110 11.0
cylinder 120 21.0
cuboid 130 23.0 -23.0 23.0 -23.0
array 1 110 place 10 10 -0.425 -0.850
media 3 1 110
media 13 1 120 -110
media 3 1 130 -120
boundary 130 40 40
'
end geom
read array
ara=1 typ=shexagonal nux=21 nuy=21
fill
0 0 0 0 0 0 0 1 1 1 1 1 1 1 0 0 0 0 0 0 0
0 0 0 0 0 1 1 1 1 1 4 1 1 1 1 1 0 0 0 0 0
0 0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0 0
0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0 0
0 0 0 1 1 1 1 1 1 1 3 1 1 1 1 1 1 1 0 0 0
0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0
0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 0
0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 1 1 1 1 1 1 1 1 1 0 1 1 1 1 1 1 1 1 1 0
0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0
0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0
0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0
0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0
```

```
0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0 0
0 0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0 0
0 0 0 0 1 1 1 1 1 1 1 1 1 1 1 1 1 0 0 0 0 0
0 0 0 0 0 0 0 1 1 1 1 1 1 1 1 0 0 0 0 0 0 0
end fill
end array
read bounds all=vacuum end bounds
end model
end
```

Figure 5.3.20-4 VISED X-Y Slice – SLOWPOKE – Normal Conditions

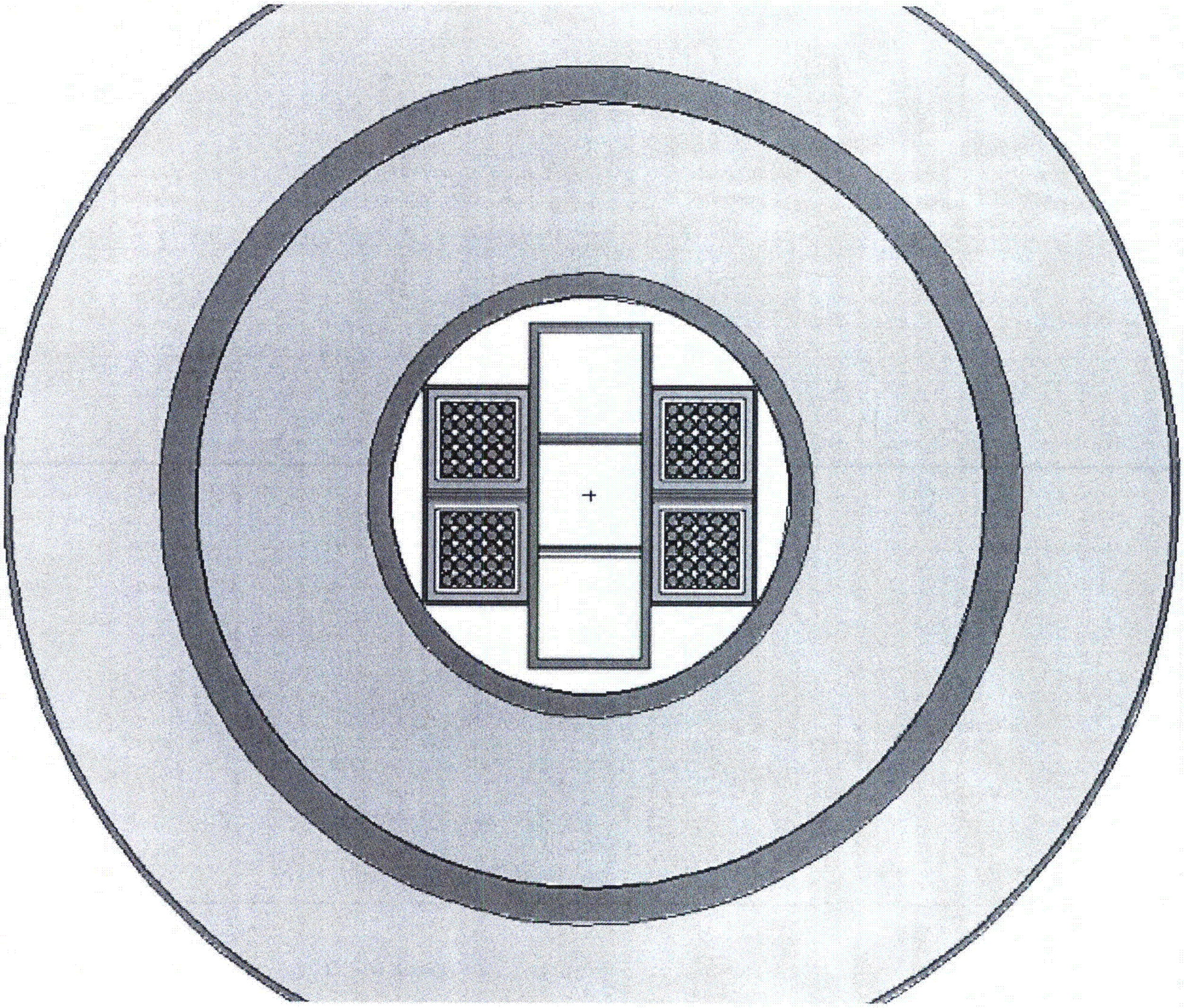
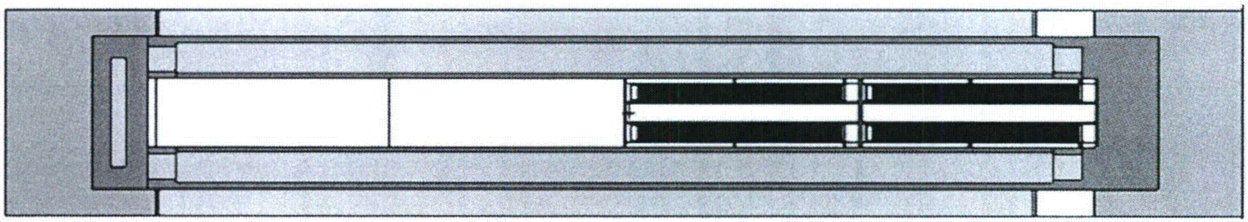


Figure 5.3.20-5 VISED Y-Z Slice – SLOWPOKE – Normal Conditions



Note: Conservatively moved material to cask cavity top.

Figure 5.3.20-6 Sample MCNP Input File – Normal Conditions

```
NAC-LWT Cask - Assy_30b90e14y - Normal Transport Conditions
C Radial Biasing - Fuel Gamma Source
C Fuel Assembly Cells
1 1 -0.8543 -1 +3    u=7 $ Homogenized Fuel Meat + Clad
2 0      -1 -3     u=7 $ Void
3 4 -2.7000 -2 +1    u=7 $ Tube OD
4 0      +2      u=7 $ Outside Tube
5 4 -2.7000 -5      u=6 $ Tube Base Plate
6 0      -4 fill=7 trcl = ( -2.5400 2.5400 0.6351 )   u=6 $ Tube 1
7 like 6 but fill=7 trcl = ( -1.2700 2.5400 0.6351 )   u=6 $ Tube 2
8 like 6 but fill=7 trcl = ( 0.0000 2.5400 0.6351 )   u=6 $ Tube 3
9 like 6 but fill=7 trcl = ( 1.2700 2.5400 0.6351 )   u=6 $ Tube 4
10 like 6 but fill=7 trcl = ( 2.5400 2.5400 0.6351 )   u=6 $ Tube 5
11 like 6 but fill=7 trcl = ( -2.5400 1.2700 0.6351 )   u=6 $ Tube 6
12 like 6 but fill=7 trcl = ( -1.2700 1.2700 0.6351 )   u=6 $ Tube 7
13 like 6 but fill=7 trcl = ( 0.0000 1.2700 0.6351 )   u=6 $ Tube 8
14 like 6 but fill=7 trcl = ( 1.2700 1.2700 0.6351 )   u=6 $ Tube 9
15 like 6 but fill=7 trcl = ( 2.5400 1.2700 0.6351 )   u=6 $ Tube 10
16 like 6 but fill=7 trcl = ( -2.5400 0.0000 0.6351 )   u=6 $ Tube 11
17 like 6 but fill=7 trcl = ( -1.2700 0.0000 0.6351 )   u=6 $ Tube 12
18 like 6 but fill=7 trcl = ( 0.0000 0.0000 0.6351 )   u=6 $ Tube 13
19 like 6 but fill=7 trcl = ( 1.2700 0.0000 0.6351 )   u=6 $ Tube 14
20 like 6 but fill=7 trcl = ( 2.5400 0.0000 0.6351 )   u=6 $ Tube 15
21 like 6 but fill=7 trcl = ( -2.5400 -1.2700 0.6351 )   u=6 $ Tube 16
22 like 6 but fill=7 trcl = ( -1.2700 -1.2700 0.6351 )   u=6 $ Tube 17
23 like 6 but fill=7 trcl = ( 0.0000 -1.2700 0.6351 )   u=6 $ Tube 18
24 like 6 but fill=7 trcl = ( 1.2700 -1.2700 0.6351 )   u=6 $ Tube 19
25 like 6 but fill=7 trcl = ( 2.5400 -1.2700 0.6351 )   u=6 $ Tube 20
26 like 6 but fill=7 trcl = ( -2.5400 -2.5400 0.6351 )   u=6 $ Tube 21
27 like 6 but fill=7 trcl = ( -1.2700 -2.5400 0.6351 )   u=6 $ Tube 22
28 like 6 but fill=7 trcl = ( 0.0000 -2.5400 0.6351 )   u=6 $ Tube 23
29 like 6 but fill=7 trcl = ( 1.2700 -2.5400 0.6351 )   u=6 $ Tube 24
30 like 6 but fill=7 trcl = ( 2.5400 -2.5400 0.6351 )   u=6 $ Tube 25
31 0      #5 #6 #7 #8 #9 #10 #11 #12 #13 #14 #15 #16 #17 #18
        #19 #20 #21 #22 #23 #24 #25 #26 #27 #28 #29 #30
            u=6 $ Void
32 4 -2.7000 -7      u=5 $ Can Base Plate
33 4 -2.7000 -9 +8 +7    u=5 $ Can
34 0      -6 fill=6 trcl = ( 0.0000 0.0000 3.0924 )   u=5 $ Tube Assy 1
35 like 34 but fill=6 trcl = ( 0.0000 0.0000 27.2225 )   u=5 $ Tube Assy 2
36 like 34 but fill=6 trcl = ( 0.0000 0.0000 51.3526 )   u=5 $ Tube Assy 3
37 like 34 but fill=6 trcl = ( 0.0000 0.0000 75.4827 )   u=5 $ Tube Assy 4
38 4 -2.7000 -10      u=5 $ Can Lid Bottom Plate
39 4 -2.7000 -11      u=5 $ Can Lid Top Plate
40 0      #32 #33 #34 #35 #36 #37 #38 #39      u=5 $ Void
C Cells - MTR 7 Element Basket
41 6 -7.9400 -13 +16 +17 +18 +19 +20 +21 +22    u=4 $ Base plate
42 6 -7.9400 -14 +23 +27    u=4 $ Support plate
43 6 -7.9400 -15 +23 +27    u=4 $ Support plate
44 6 -7.9400 -23 +24 #41 #42 #43    u=4 $ Center column
```

```
45 6 -7.9400 -25 #41 #42 #43      u=4 $ Center divider upper
46 6 -7.9400 -26 #41 #42 #43      u=4 $ Center divider lower
47 6 -7.9400 -27 +28 +23 #41 #42 #43    u=4 $ Small side
48 6 -7.9400 -29 #41 #42 #43      u=4 $ Left divider
49 6 -7.9400 -30 #41 #42 #43      u=4 $ Right divider
50 0      #41 #42 #43 #44 #45 #46 #47 #48 #49      u=4 $ Void
C Cells - Basket Cavity
51 0      -12 fill=5 trcl = ( -9.5250 4.6990 3.1877 )    u=3 $ UL
52 like 51 but fill=5 trcl = ( -9.5250 -4.6990 3.1877 )    u=3 $ LL
53 like 51 but fill=5 trcl = ( 9.5250 4.6990 3.1877 )    u=3 $ UR
54 like 51 but fill=5 trcl = ( 9.5250 -4.6990 3.1877 )    u=3 $ LR
55 0      #51 #52 #53 #54      fill=4 u=3 $ Void
C Cells - LWT Cavity
56 0      -38      u=2
57 0      -39      u=2
58 0      -40 fill=3 ( 0.0000 0.0000 227.3300 )    u=2
59 0      -41 fill=3 ( 0.0000 0.0000 339.0900 )    u=2
60 0      #56 #57 #58 #59      u=2
C Cells - LWT Cask Normal Conditions
61 5 -11.344 -45      u=1 $ BotPb
62 0      -44 fill=2      u=1 $ Cavity
63 6 -7.9400 -42 -43 +45    u=1 $ Bottom
64 6 -7.9400 -42 +43 +47 +50 +44 u=1 $ OuterShell
65 6 -7.9400 -46 +49 +44    u=1 $ InnerShellTaper
66 6 -7.9400 -48 +44      u=1 $ InnerShell
67 5 -11.344 -49 +48      u=1 $ Lead
68 5 -11.344 -47 +46 +49    u=1 $ LeadTaper
69 0      -50 +49      u=1 $ LeadGap
70 3 -0.9669 -52 +42      u=1 $ NeutronShield
71 6 -7.9400 -51 +42 +52    u=1 $ NSShell
72 7 -0.4997 -53 +42      u=1 $ UpperLimiter
73 7 -0.4997 -54 +42      u=1 $ LowerLimiter
74 0      -55 +42 +51 +53 +54 u=1 $ Container
75 0      +55      u=1 $ Outside
C Detector Cells - Radial Biasing
100 0 -100 fill=1 $ Surface
200 0 -200 +100 $ 1ft
300 0 -300 +100 +200 $ 1m
400 0 -400 +100 +200 +300 $ 2m
500 0 -500 +100 +200 +300 +400 $ 2m+Convey
600 0 +100 +200 +300 +400 +500 $ Exterior

C Fuel Assembly Surfaces
1 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 23.4950 0.5080 $ Tube ID
2 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 23.4950 0.6349 $ Tube OD
3 PZ 1.4950      $ Fuel Cut Plain
4 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 23.4951 0.6350 $ Tube
5 RPP -3.1750 3.1750 -3.1750 3.1750 0.0000 0.6350  $ Tube Base Plate
6 RPP -3.1751 3.1751 -3.1751 3.1751 0.0000 24.1301  $ Tube Container
7 RPP -4.1910 4.1910 -4.1910 4.1910 0.0000 0.4699  $ Can Base Plate
8 RPP -3.5560 3.5560 -3.5560 3.5560 0.0000 100.6475  $ Can ID
9 RPP -4.1910 4.1910 -4.1910 4.1910 0.0000 100.6475  $ Can OD
10 RPP -3.4925 3.4925 -3.4925 3.4925 99.6823 100.6475  $ Can Lid Lower Plate
```

```
11 RPP -4.1910 4.1910 -4.1910 4.1910 100.6475 102.2223 $ Can Lid Upper Plate
12 RPP -4.1911 4.1911 -4.1911 4.1911 0.0000 102.2223 $ Container
C Surfaces - MTR 7 Element Basket
13 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 1.2700 16.8466 $ Base plate
14 RCC 0.0000 0.0000 52.0700 0.0000 0.0000 1.2700 16.8466 $ Support plate
15 RCC 0.0000 0.0000 104.1400 0.0000 0.0000 1.2700 16.8466 $ Support plate
16 CZ 1.2700 $ Hole CC
17 C/Z 0.0000 9.5250 1.2700 $ Hole UC
18 C/Z 0.0000 -9.5250 1.2700 $ Hole LC
19 C/Z -9.5250 4.6990 1.2700 $ Hole UL
20 C/Z -9.5250 -4.6990 1.2700 $ Hole LL
21 C/Z 9.5250 4.6990 1.2700 $ Hole UR
22 C/Z 9.5250 -4.6990 1.2700 $ Hole LR
23 RPP -5.1604 5.1604 -14.6939 14.6939 1.2700 111.7600 $ Center column outer
24 RPP -4.3667 4.3667 -13.9002 13.9002 1.2700 111.7600 $ Center column inner
25 RPP -4.3667 4.3667 4.3688 5.1626 1.2700 111.7600 $ Center divider upper
26 RPP -4.3667 4.3667 -5.1626 -4.3688 1.2700 111.7600 $ Center divider lower
27 RPP -14.1986 14.1986 -9.3599 9.3599 1.2700 111.7600 $ Small side outer
28 RPP -13.8938 13.8938 -9.0551 9.0551 1.2700 111.7600 $ Small side inner
29 RPP -13.8938 -5.1604 -0.3175 0.3175 1.2700 111.7600 $ Left divider
30 RPP 5.1604 13.8938 -0.3175 0.3175 1.2700 111.7600 $ Right divider
C Surfaces - Basket Cavity
31 RPP -4.3667 4.3667 -4.3688 4.3688 1.2700 111.7600 $ CC
32 RPP -4.3667 4.3667 5.1626 13.9002 1.2700 111.7600 $ UC
33 RPP -4.3667 4.3667 -13.9002 -5.1626 1.2700 111.7600 $ LC
34 RPP -13.8938 -5.1604 0.6350 9.3726 1.2700 111.7600 $ UL
35 RPP -13.8938 -5.1604 -9.3726 -0.6350 1.2700 111.7600 $ LL
36 RPP 5.1604 13.8938 0.6350 9.3726 1.2700 111.7600 $ UR
37 RPP 5.1604 13.8938 -9.3726 -0.6350 1.2700 111.7600 $ LR
C Surfaces - LWT Cavity
38 RCC 0.0000 0.0000 3.8100 0.0000 0.0000 111.7600 16.8467 $ Basket
39 RCC 0.0000 0.0000 115.5700 0.0000 0.0000 111.7600 16.8467 $ Basket
40 RCC 0.0000 0.0000 227.3300 0.0000 0.0000 111.7600 16.8467 $ Basket
41 RCC 0.0000 0.0000 339.0900 0.0000 0.0000 111.7600 16.8467 $ Basket
C Surfaces - LWT Cask Normal Conditions
42 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 507.3650 36.5189 $ Lwt
43 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 26.6700 36.5189 $ Bottom
44 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 452.1200 16.9863 $ Cavity
45 RCC 0.0000 0.0000 -17.7800 0.0000 0.0000 7.6200 26.3525 $ Bottom gamma shield
46 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 20.1740 $ Lead id - taper
47 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 31.5976 $ Lead od - taper
48 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 18.9103 $ Lead id
49 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 33.3271 $ Lead od
50 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 33.4645 $ Lead gap
51 RCC 0.0000 0.0000 3.8100 0.0000 0.0000 419.1000 49.8183 $ Neutron shield shell
52 RCC 0.0000 0.0000 5.0800 0.0000 0.0000 416.5600 49.2189 $ Neutron shield
53 RCC 0.0000 0.0000 450.2150 0.0000 0.0000 70.5612 49.8183 $ Upper limiter
54 RCC 0.0000 0.0000 -68.0212 0.0000 0.0000 71.8312 49.8183 $ Lower limiter
55 RCC 0.0000 0.0000 -68.0212 0.0000 0.0000 588.7974 49.8183 $ Container
C Radial Detector DRA (Surface)
100 RCC 0.0000 0.0000 -68.1212 0.0000 0.0000 588.9974 49.9184
101 PZ -38.6713
102 PZ -9.2215
```

```
103 PZ 20.2284
104 PZ 49.6783
105 PZ 79.1282
106 PZ 108.5780
107 PZ 138.0279
108 PZ 167.4778
109 PZ 196.9276
110 PZ 226.3775
111 PZ 255.8274
112 PZ 285.2772
113 PZ 314.7271
114 PZ 344.1770
115 PZ 373.6269
116 PZ 403.0767
117 PZ 432.5266
118 PZ 461.9765
119 PZ 491.4263
C Radial Detector DRB (1ft)
200 RCC 0.0000 0.0000 -98.6012 0.0000 0.0000 649.9574 80.2984
201 PZ -66.1033
202 PZ -33.6055
203 PZ -1.1076
204 PZ 31.3903
205 PZ 63.8882
206 PZ 96.3860
207 PZ 128.8839
208 PZ 161.3818
209 PZ 193.8796
210 PZ 226.3775
211 PZ 258.8754
212 PZ 291.3732
213 PZ 323.8711
214 PZ 356.3690
215 PZ 388.8669
216 PZ 421.3647
217 PZ 453.8626
218 PZ 486.3605
219 PZ 518.8583
C Radial Detector DRC (1m)
300 RCC 0.0000 0.0000 -168.1212 0.0000 0.0000 788.9974 149.8184
301 PZ -135.2463
302 PZ -102.3714
303 PZ -69.4965
304 PZ -36.6216
305 PZ -3.7467
306 PZ 29.1282
307 PZ 62.0030
308 PZ 94.8779
309 PZ 127.7528
310 PZ 160.6277
311 PZ 193.5026
312 PZ 226.3775
313 PZ 259.2524
```

```
314 PZ 292.1273
315 PZ 325.0022
316 PZ 357.8771
317 PZ 390.7520
318 PZ 423.6269
319 PZ 456.5017
320 PZ 489.3766
321 PZ 522.2515
322 PZ 555.1264
323 PZ 588.0013
C Radial Detector DRD (2m)
400 RCC 0.0000 0.0000 -268.1212 0.0000 0.0000 988.9974 249.8184
401 PZ -226.9130
402 PZ -185.7048
403 PZ -144.4965
404 PZ -103.2883
405 PZ -62.0801
406 PZ -20.8719
407 PZ 20.3364
408 PZ 61.5446
409 PZ 102.7528
410 PZ 143.9611
411 PZ 185.1693
412 PZ 226.3775
413 PZ 267.5857
414 PZ 308.7940
415 PZ 350.0022
416 PZ 391.2104
417 PZ 432.4186
418 PZ 473.6269
419 PZ 514.8351
420 PZ 556.0433
421 PZ 597.2515
422 PZ 638.4598
423 PZ 679.6680
C Radial Detector DRE (2m+Convey)
500 RCC 0.0000 0.0000 -269.1212 0.0000 0.0000 990.9974 321.9200
501 PZ -227.8296
502 PZ -186.5381
503 PZ -145.2465
504 PZ -103.9550
505 PZ -62.6634
506 PZ -21.3719
507 PZ 19.9197
508 PZ 61.2113
509 PZ 102.5028
510 PZ 143.7944
511 PZ 185.0859
512 PZ 226.3775
513 PZ 267.6691
514 PZ 308.9606
515 PZ 350.2522
516 PZ 391.5437
```

```
517 PZ 432.8353
518 PZ 474.1269
519 PZ 515.4184
520 PZ 556.7100
521 PZ 598.0015
522 PZ 639.2931
523 PZ 680.5846

C
C Materials List
C
C Homogenized U-Al Fuel
m1   92235 -1.7661E-01 92238 -1.9624E-02 13027 -8.0376E-01
C Water
m2   1001 6.6667E-01 8016 3.3333E-01
mt2  lwtr.01
C Water/Glycol
m3   1001 -1.03651E-01 8016 -6.75619E-01 6000 -2.20730E-01
C Aluminum
m4   13027 -1.0
C Lead
m5   82000 -1.0
C Stainless Steel 304
m6   26000 -0.695 24000 -0.190 28000 -0.095
     25055 -0.020
C Aluminum Honeycomb Impact Limiter
m7   13027 -1.0
C
C Cell Importances
imp:p 1 79r 0
C
C Source Definition - Fuel Gamma
C 30% burnup, wt % U-235, 14-year cool time, 2.786 g U-235 per rod, 0.003 W/rod
sdef RAD=d1 EXT=d2 ERG=d3 cell=100:62:d4:d5:d6:d7:1
    POS= 0.0000 0.0000 1.4950
    AXS= 0.0000 0.0000 1.0000
si1 0 0.5080
sp1 -21 1
si2 0 22.0000
sp2 -21 1
si3 1.000E-02 4.500E-02 1.000E-01 2.000E-01 3.000E-01 4.000E-01
   6.000E-01 8.000E-01 1.000E+00 1.330E+00 1.660E+00 2.000E+00
   2.500E+00 3.000E+00 4.000E+00 5.000E+00 6.500E+00 8.000E+00
   1.000E+01
sp3 0.0000E+00 4.6619E+09 1.6170E+09 9.2992E+08 2.9371E+08 2.1001E+08
   1.5357E+08 6.8443E+09 3.0713E+07 1.9365E+07 3.4947E+06 5.3655E+05
   3.1683E+04 1.8502E+02 1.2818E+01 1.1290E-03 4.3894E-04 8.3643E-05
   1.7944E-05
si4 l 58 59
sp4 1.0 1.0
si5 l 51 52 53 54
sp5 1.0 1.0 1.0 1.0
si6 l 34 35 36 37
```

```
sp6 1.0 1.0 1.0 1.0
si7 l 6 7 8 9 10 11
    12 13 14 15 16 17
    18 19 20 21 22 23
    24 25 26 27 28 29
    30
sp7 1.0 1.0 1.0 1.0 1.0 1.0
   1.0 1.0 1.0 1.0 1.0 1.0
   1.0 1.0 1.0 1.0 1.0 1.0
   1.0 1.0 1.0 1.0 1.0 1.0
   1.0
mode p
nps 160000000
C
C ANSI/ANS-6.1.1-1977 - Gamma Flux-to-Dose Conversion Factors
C (mrem/hr)/(photons/cm2-sec)
de0 0.01 0.03 0.05 0.07 0.1 0.15 0.2
   0.25 0.3 0.35 0.4 0.45 0.5 0.55
   0.6 0.65 0.7 0.8 1 1.4 1.8
   2.2 2.6 2.8 3.25 3.75 4.25 4.75
   5 5.25 5.75 6.25 6.75 7.5 9
   11 13 15
df0 3.96E-03 5.82E-04 2.90E-04 2.58E-04 2.83E-04 3.79E-04 5.01E-04
   6.31E-04 7.59E-04 8.78E-04 9.85E-04 1.08E-03 1.17E-03 1.27E-03
   1.36E-03 1.44E-03 1.52E-03 1.68E-03 1.98E-03 2.51E-03 2.99E-03
   3.42E-03 3.82E-03 4.01E-03 4.41E-03 4.83E-03 5.23E-03 5.60E-03
   5.80E-03 6.01E-03 6.37E-03 6.74E-03 7.11E-03 7.66E-03 8.77E-03
   1.03E-02 1.18E-02 1.33E-02
C
C Weight Window Generation - Radial
wwg 2 0 0 0 0
wwp:p 5 3 5 0 -1 0
mesh geom=cyl ref=0 13 243 origin=0.1 0.1 -568
    imesh 16.8 17.0 18.9 33.3 36.5 49.2 49.8 549.8
    iints 1 1 1 5 1 1 1 1
    jmesh 500 541 550 558 568 577 1019 1020 1049 1089 1589
    jints 1 1 1 1 1 1 1 1 1 1 1
    kmesh 1
    kints 1
wwge:p 1e-3 1 20
fc2 Radial Surface Tally
f2:p +100.1
fm2 1.18116E+13
fs2 -101 -102 -103 -104 -105 -106
   -107 -108 -109 -110 -111 -112
   -113 -114 -115 -116 -117 -118
   -119 T
tf2
fc12 Radial 1ft Tally
f12:p +200.1
fm12 1.18116E+13
fs12 -201 -202 -203 -204 -205 -206
   -207 -208 -209 -210 -211 -212
```

```
   -213 -214 -215 -216 -217 -218
   -219 T
tf12
fc22 Radial 1m Tally
f22:p +300.1
fm22 1.18116E+13
fs22 -301 -302 -303 -304 -305 -306
   -307 -308 -309 -310 -311 -312
   -313 -314 -315 -316 -317 -318
   -319 -320 -321 -322 -323 T
tf22
fc32 Radial 2m Tally
f32:p +400.1
fm32 1.18116E+13
fs32 -401 -402 -403 -404 -405 -406
   -407 -408 -409 -410 -411 -412
   -413 -414 -415 -416 -417 -418
   -419 -420 -421 -422 -423 T
tf32
fc42 Radial 2m+Convey Tally
f42:p +500.1
fm42 1.18116E+13
fs42 -501 -502 -503 -504 -505 -506
   -507 -508 -509 -510 -511 -512
   -513 -514 -515 -516 -517 -518
   -519 -520 -521 -522 -523 T
tf42
C
C Print Control
prdmp -30 -60 1 2
print
C Random Number Generator
rand gen=2 seed=15617098509349 stride=152917 hist=1
```

Figure 5.3.20-7 Normal Condition Radial Surface Dose Rate Profile by Source Type – SLOWPOKE Fuel

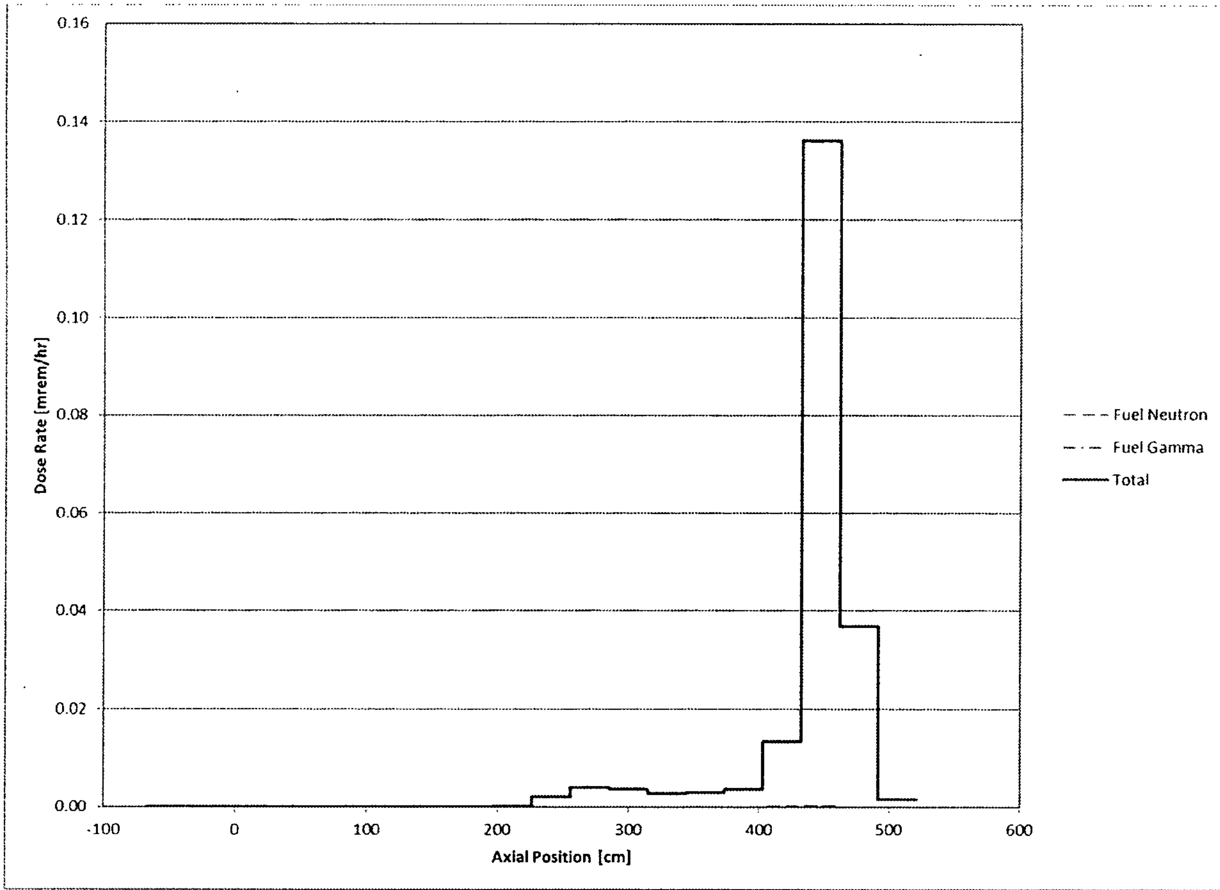


Figure 5.3.20-8 Normal Condition 2-m Radial Surface Dose Rate Profile by Source Type – SLOWPOKE Fuel

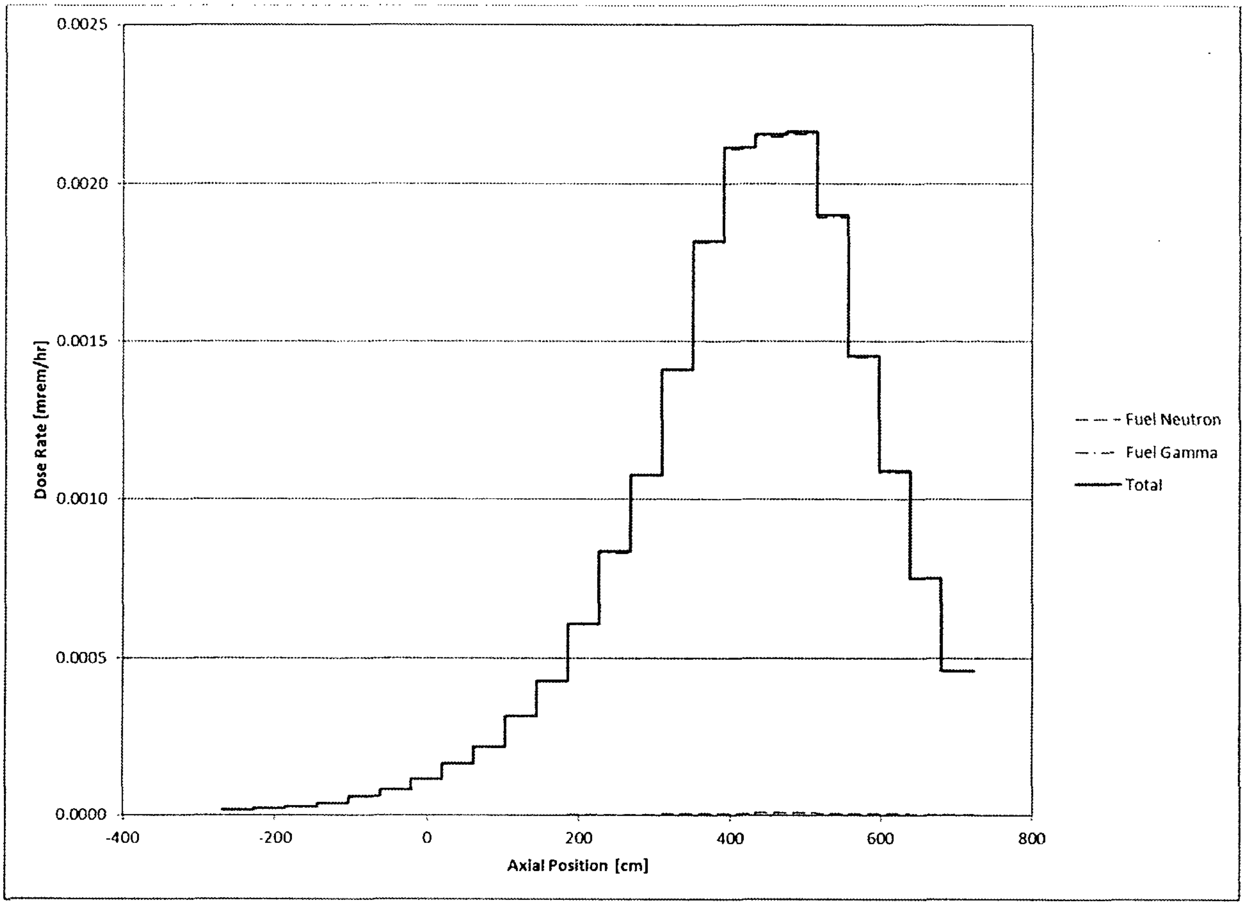


Figure 5.3.20-9 Accident Condition Radial 1m Dose Rate Profile by Source Type – SLOWPOKE

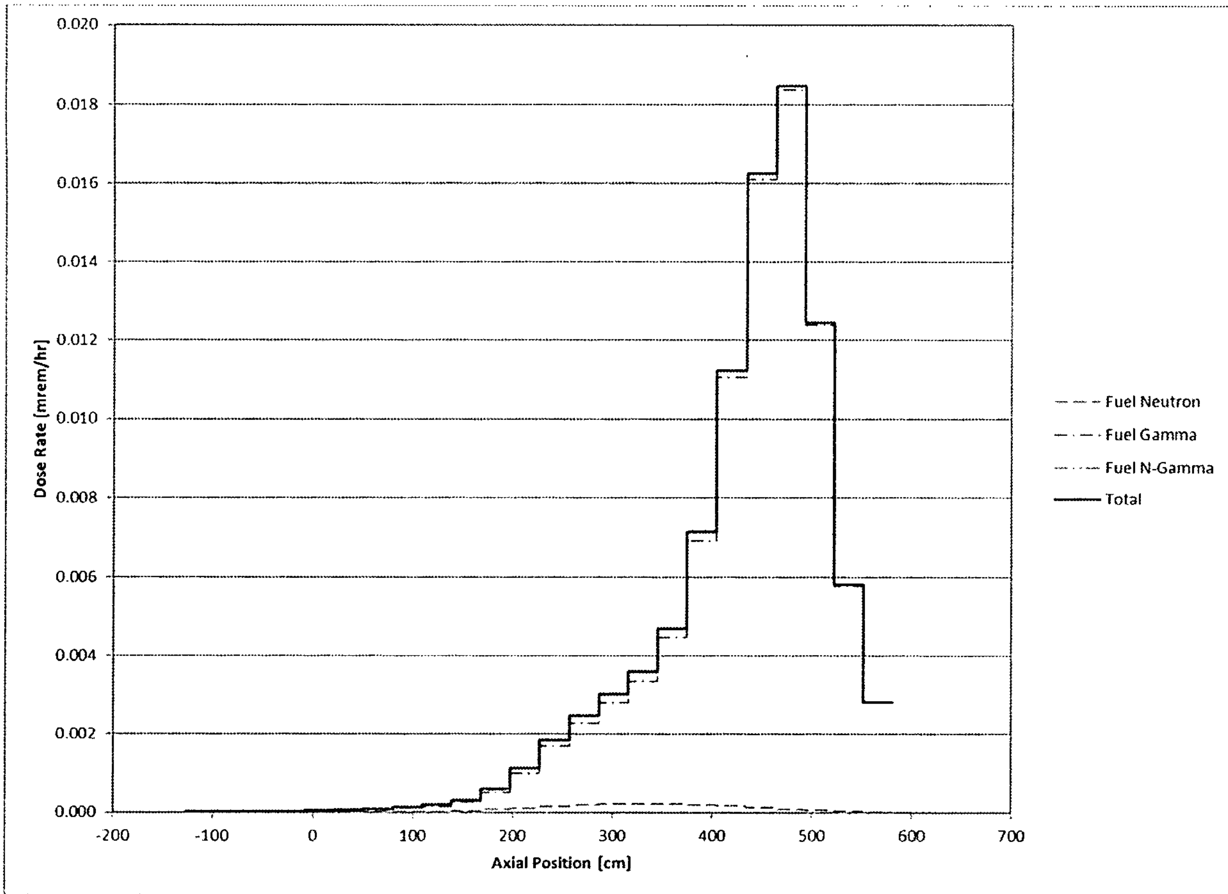


Table 5.3.20-1 SLOWPOKE Fuel Geometry and Materials

Fuel Element Type		Rod
Chemical Form		U-Al Alloy
Active Fuel Length	cm	22
Active Fuel Diameter	cm	0.422
Weight of U-235	g	2.786
Weight of total U	g	2.990
Alloy or compount material weight	g	7.688
Total weight of fuel meat	g	10.678
Clad Thickness	cm	0.051
Clad Weight (including caps)	g	4.981
Clad Material		Aluminum
Element Length	cm	22.83
Diameter (endcaps)	cm	0.61
Diameter (clad)	cm	0.53
Total weight of fuel element	g	15.659
Enrichment %	%	93
Burn Time	hrs	32000
Core Maximum Power	kW	20
Maximum Burnup (^{235}U depletion)	%	2

Table 5.3.20-2 Source Term Generation Parameters for SLOWPOKE Fuel

Parameter	Value
U Mass Per Rod (grams)	3.1
Core Power (kW)	20
Number of Hours Burned	35280
Number of Years Cooled	14
Number of Rods / Core	315
Initial Enrichment (wt % ^{235}U)	90
Burnup (% ^{235}U)	4.5
Burnup (GWd/MTU)	30
Moderator/Box Temperature (C)	40
Clad Temperature (C)	90
Fuel Temperature (C)	100

Table 5.3.20-3 SLOWPOKE Neutron Source Term (per MTU)

Group	E Lower [MeV]	E Upper [MeV]	Source [neutrons/sec]
1	6.380E+00	2.000E+01	8.392E+01
2	3.010E+00	6.380E+00	1.239E+03
3	1.830E+00	3.010E+00	2.221E+04
4	1.420E+00	1.830E+00	5.368E+04
5	9.070E-01	1.420E+00	9.340E+04
6	4.080E-01	9.070E-01	8.055E+04
7	1.110E-01	4.080E-01	3.950E+04
8	1.500E-02	1.110E-01	4.675E+03
9	3.040E-03	1.500E-02	1.404E+02
10	5.830E-04	3.040E-03	1.121E+01
11	1.010E-04	5.830E-04	1.063E+00
12	2.900E-05	1.010E-04	2.284E-02
13	1.070E-05	2.900E-05	3.852E-03
14	3.060E-06	1.070E-05	1.625E-04
15	1.860E-06	3.060E-06	1.011E-05
16	1.300E-06	1.860E-06	3.875E-06
17	1.130E-06	1.300E-06	1.030E-06
18	1.000E-06	1.130E-06	6.480E-07
19	8.000E-07	1.000E-06	1.132E-06
20	4.140E-07	8.000E-07	1.586E-06
21	3.250E-07	4.140E-07	2.069E-07
22	2.250E-07	3.250E-07	3.119E-07
23	1.000E-07	2.250E-07	2.064E-07
24	5.000E-08	1.000E-07	6.687E-08
25	3.000E-08	5.000E-08	3.531E-08
26	1.000E-08	3.000E-08	9.255E-11
27	1.000E-11	1.000E-08	5.251E-11
Total			2.955E+05

Table 5.3.20-4 SLOWPOKE Fuel Gamma Source Term (per MTU)

Group	E Lower [MeV]	E Upper [MeV]	Source [photons/sec]
1	8.00E+00	1.00E+01	5.7677E+00
2	6.50E+00	8.00E+00	2.6885E+01
3	5.00E+00	6.50E+00	1.4109E+02
4	4.00E+00	5.00E+00	3.6288E+02
5	3.00E+00	4.00E+00	4.1202E+06
6	2.50E+00	3.00E+00	5.9472E+07
7	2.00E+00	2.50E+00	1.0184E+10
8	1.66E+00	2.00E+00	1.7246E+11
9	1.33E+00	1.66E+00	1.1233E+12
10	1.00E+00	1.33E+00	6.2244E+12
11	8.00E-01	1.00E+00	9.8721E+12
12	6.00E-01	8.00E-01	2.2000E+15
13	4.00E-01	6.00E-01	4.9361E+13
14	3.00E-01	4.00E-01	6.7505E+13
15	2.00E-01	3.00E-01	9.4406E+13
16	1.00E-01	2.00E-01	2.9890E+14
17	4.50E-02	1.00E-01	5.1975E+14
18	1.00E-02	4.50E-02	1.4985E+15
Total			4.7457E+15

Table 5.3.20-5 Fuel Homogenization for SLOWPOKE Fuel

Component	Area [cm^2]	Area Fraction
Fuel	1.3987E-01	1.7252E-01
Gap	4.0055E-03	4.9406E-03
Clad	7.6746E-02	9.4663E-02
Void	5.9011E-01	7.2788E-01
Total	8.1073E-01	1.0000E+00

Note: Homogenization limited to smear of fuel rod within aluminum canister tube.

Table 5.3.20-6 Canister/Basket/Cask Material Descriptions for SLOWPOKE Fuel

Material	Element	Density [g/cm^3]	Number Density [atom/b-cm]
Aluminum	Al	2.67	7.278E-02
Stainless Steel 304	Fe	7.94	5.9505E-02
	Cr		1.7472E-02
	Ni		7.7392E-03
	Mn		1.7407E-03
Lead	Pb	11.34	3.2967E-02
Neutron Shield	H	0.97	5.9884E-02
	O		2.4595E-02
	C		1.0701E-02
Impact Limiter	Al	0.50	1.1153E-02

Table 5.3.20-7 Canister Dimensions SLOWPOKE Fuel

Description	Dimension [in]
CANISTER:	
Bottom Plate Thickness	0.375
Lid Thickness	1.00
OD	3.30
ID	2.80
Side Wall Height	39.44
Bottom Plate Inset 1	0.130
Bottom Plate Inset 2	0.060
Lid Lower Bottom Thickness	0.38
Lid Top Width	3.30
Lid Bottom Width	2.75
Lid Handle Height	2.500
CANISTER INSERT:	
Tube Length	9.25
Tube OD	0.50
Tube ID	0.40
Base Plate Thickness	0.25
Base Plate Width	2.50

Table 5.3.20-8 Maximum and Average Dose Rates for SLOWPOKE Fuel

Transport Condition	Dose Rate Location	Maximum [mrem/hr]	Maximum FSD	Average [mrem/hr]	Average FSD
Normal	Side Surface of Cask	0.14	7.6%	0.01	23.0%
	Top Surface of Cask	0.004	8.4%	0.002	14.9%
	Bottom Surface of Cask	< 0.00001	--	< 0.00001	--
	Side 1m (Transport Index)	0.01	7.1%	0.002	12.6%
	2m from Truck - Radial	0.002	7.6%	0.001	9.8%
	2m from Top	0.0004	58.3%	0.0004	47.7%
	2m from Bottom	< 0.00001	--	< 0.00001	--
	Edge of Truck - Top	0.0001	30.1%	0.00004	36.2%
	Edge of Truck - Bottom	< 0.00001	--	< 0.00001	--
	Dose at Cab of Truck	0.00005	37.2%	0.00003	46.0%
Accident	Side Surface of Cask	0.29	8.8%	0.02	24.5%
	Top Surface of Cask	0.03	8.6%	0.01	12.6%
	Bottom Surface of Cask	< 0.00001	--	< 0.00001	--
	Side 1m	0.02	7.9%	0.004	12.5%
	Top 1m	0.002	7.4%	0.001	9.7%
	Bottom 1m	< 0.00001	--	< 0.00001	--

Table 5.3.20-9 Summarized Maximum Dose Rates for SLOWPOKE Fuel

Transport Condition	Dose Rate Location	Maximum [mrem/hr]	Limit [mrem/hr]
Normal	Side Surface of Cask	0.14	200
	Side 1m (Transport Index)	0.01	10
	2m from Truck - Radial	0.002	N/A
Accident	Side 1m	0.02	1000

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Note: See separate Section 6.6 for Appendices to this chapter.

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Note: See separate Section 6.6 for Appendices to this chapter, along with the List of Figures for the Appendices.

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6 CRITICALITY EVALUATION

The NAC-LWT cask is designed to transport either 1 pressurized water reactor (PWR) assembly; up to 25 intact PWR or BWR rods in a rod holder or fuel assembly lattice; up to 25 PWR or BWR fuel rods with a maximum of 14 of the rods classified as damaged in a rod holder; up to 16 PWR UO_2 or MOX rods in a rod holder; 2 boiling water reactor (BWR) assemblies; 15 sound metallic fuel rods; 6 failed metallic fuel rods; up to 42 high enriched uranium (HEU), medium enriched uranium (MEU) or low enriched uranium (LEU) Materials Test Reactor (MTR) fuel elements, or DIDO fuel assemblies; up to 140 TRIGA fuel elements; two packages of General Atomics Irradiated Fuel Material (GA IFM); up to 560 TRIGA fuel cluster rods; 1 consolidation canister with up to 300 TPBARs (including up to 2 damaged TPBARs); up to 700 PULSTAR fuel elements; up to 42 spiral fuel assemblies; up to 42 MOATA plate bundles, or up to 800 SLOWPOKE rods. This chapter illustrates that all packages meet the requirements of parts 71.55, 71.59 and 71.71 of 10 CFR 71.

In accordance with the requirements of 10 CFR 71.59 (b), the NAC-LWT cask is assigned a Criticality Safety Index (CSI) for criticality control for the authorized contents as follows:

Approved Contents	CSI
PWR fuel assemblies	100
BWR fuel assemblies	5.0
MTR fuel elements	0.0
Metallic fuel rods	0.0
TRIGA fuel elements (in poisoned TRIGA fuel baskets)	0.0
TRIGA fuel elements (in nonpoisoned TRIGA fuel baskets)	12.5
TRIGA fuel cluster rods	0.0
High burnup PWR (UO_2 or MOX) rods*	0.0
High burnup BWR rods*	0.0
DIDO fuel elements	12.5
General Atomic Irradiated Fuel Material (GA IFM)	0.0
TPBARS and segmented TPBARS	0.0
Intact (uncanned) PULSTAR fuel	0.0
Canned PULSTAR fuel	33.4
ANSTO fuel (spiral and/or MOATA)	0.0
Solid irradiated hardware	0.0
ANSTO-DIDO fuel combination	0.0
SLOWPOKE fuel rods (undamaged or damaged)	0.0

* up to 14 damaged rods

fuel rods (i.e., no gross fuel failure, hairline cracks or pinholes are allowed). All evaluation detail, including input, method, analysis results and critical benchmarks, are included in Section 6.7.1. Included are the fuel rod geometry and material description, the MCNP model used in the rod holder analyses, and the criticality analysis results of the NAC-LWT loaded with up to 16 PWR rods (fueled with either UO_2 or MOX material). The system reactivity of the NAC-LWT with up to 16 undamged PWR rods is evaluated as a function of rod pitch. The fuel is assumed to be fresh, i.e., no burnup credit. An infinite array of casks is analyzed. Variation of moderator density inside and outside the cask is considered. Also included in the analysis are preferential flooding evaluations of the canister that contains the rod array. The results show that the bias adjusted k_{eff} of an infinite array of NAC-LWT casks at optimum fuel rod pitch and at optimum interspersed moderation is significantly below the upper safety limit (USL) for MOX and UO_2 criticality benchmarks.

Analyses are performed on the NAC-LWT with five DIDO baskets containing DIDO elements and an ANSTO top basket module containing DIDO or ANSTO fuel elements. ANSTO basket contents have been evaluated with an aluminum damaged fuel can (DFC). Section 6.3.10 presents the methods (CSAS25) and KENO-VA models used in the analysis. Section 6.4.11 presents the criticality analysis results of the NAC-LWT cask loaded with the combined payload. Criticality of the NAC-LWT cask with the most limiting fuel characteristics and basket configuration is evaluated. The fuel elements are assumed to be unburned. An infinite array of casks in both the radial and axial extent is analyzed. The results of the analysis show that the bias adjusted k_{eff} of an infinite array of NAC-LWT casks with the most-limiting DIDO/ANSTO basket payload under normal and accident conditions at optimum interspersed moderation (void) is below 0.95.

Analyses are performed on the NAC-LWT with up to 800 SLOWPOKE rods. SLOWPOKE fuel rods are permitted with up to 95.0 wt % ^{235}U initial enrichment. The payload consists of undamaged and/or damaged fuel. All evaluation details, including input, method, and analysis results, are included in Section 6.7.2. The criticality benchmark (defined here) analysis for this material is shown in Section 6.5.5. Included in Section 6.7.2 are the fuel rod geometry and material description, the MCNP model used in the canister, and the criticality analysis results of the NAC-LWT loaded with up to 800 SLOWPOKE rods. The fuel is assumed to be fresh, i.e., no burnup credit. An infinite array of casks is analyzed. Variation of moderator density inside and outside the cask is considered. Also included in the analysis are preferential flooding evaluations of the canister that contains the rod array. The results show that the bias adjusted k_{eff} of an infinite array of NAC-LWT casks at optimum fuel rod pitch and at optimum interspersed moderation is significantly below the upper safety limit (USL).

A full set of studies evaluating the reactivity changes associated with varying cavity interior, FHU, and exterior moderator densities is summarized in Table 6.4.8-7 through Table 6.4.8-11. The studies indicate that the most reactive configuration is for flooding of the RERTR and HTGR enclosures (FHUs) with interior and exterior void (loss of neutron shield). This configuration produces the maximum reactivity FHUs, while maximizing interaction between the FHUs within the cask and between casks.

6.4.8.7 Single Cask Evaluation

The 10 CFR 71.55(b)(3) requires an evaluation of the NAC-LWT with the containment system fully reflected by water. The containment for the NAC-LWT is the cask inner shell. While no operating condition results in a removal of the cask outer shell and lead gamma shield, each of the partial flooding cases at four combinations of interior and exterior moderator is reevaluated by removing the lead and outer shells (including neutron shield), and reflecting the system by water at full density on the X and Y faces (the z faces are mirrored to yield an axially infinite model). The results of this analysis are shown in Table 6.4.8-12 and demonstrate that the system reactivity decreases with the removal of the lead, outer shell and neutron shield reflectors.

6.4.8.8 Damaged TRIGA Fuel Evaluation

The combined payload model with a homogenized TRIGA fuel description is used to evaluate the system reactivity in the event that both the intact and sectioned fuel elements become damaged. Models are executed by varying the volume fraction of water in the TRIGA fuel mixture from zero to unity. The maximum volume fraction is 0.6816 based on the FHU cavity volume (7140 cm^3) and the total volume of TRIGA elements (2,273 cm^3), which does not consider the volume of the aluminum tubing within the FHU primary enclosure.

Based on the results summarized in Table 6.4.8-13, homogenized TRIGA elements are more reactive than intact TRIGA elements. Evaluation results documented in Table 6.4.8-13 are based on infinite cask array models. A single cask evaluation of the maximum water volume fraction case yielded a k_{eff} of 0.38885 ± 0.00066.

6.4.8.9 Code Bias and Code Bias Uncertainty Adjustments

As shown in Section 6.4.8.1, the TRIGA elements in the RERTR enclosure are more reactive than the HTGR fuel matrix in its enclosure. Therefore, code bias and code uncertainty adjustments are based on the TRIGA fuel element criticality benchmarks in Section 6.1.1.

A calculation of k_s under normal and accident conditions can now be made based on the previous results and based on the KENO-Va validation statistics presented in Section 6.1.1. The value k_s

is calculated based on the KENO-Va Monte Carlo average plus any biases and uncertainties associated with the methods and the modeling, i.e.:

$$k_s = k_{eff} + \Delta k_{Bias} + \Delta k_{BU} + 2\sigma_{MC} \leq 0.95$$

In the validation presented in Section 6.5.3, a negative bias (allowance for overprediction of k_{eff}) and a 95/95 method uncertainty of ± 0.0168 was determined. The negative bias correction is neglected. Thus, the equation for k_s becomes:

$$k_s = k_{eff} + 0.0168 + 2\sigma_{MC}$$

The k_s values for the relevant analysis are included in Table 6.4.8-1 through Table 6.4.8-13. The maximum k_s, 0.74015, for the GA IFM shipment results from an infinite height model with an infinite number of casks.

For both normal and accident conditions, the calculated k_{eff} values, after correction for uncertainty, are well below the 0.95 limit. The analyses demonstrate that, including all calculational and mechanical uncertainties, an infinite array of NAC-LWT casks with GA IFM remains subcritical under normal and accident conditions.

limitations, are therefore, required to document an acceptable system configuration. KENO models with cans restricted to the top and base modules and intact fuel assemblies in the remaining two modules reduces system reactivity significantly as documented in Table 6.4.9-4. Without the neutron shield (accident conditions) there is substantial neutronic coupling between casks in an array and limiting the number of casks produces a significant additional reactivity reduction. For the mixed loaded three-cask array, the calculated k_{eff} is 0.84910 ± 0.00079. A limit of a three-cask array produces a CSI of 33.4 under the accident conditions modeled. System reactivity for a normal condition infinite array of casks loaded with damaged fuel cans is low, $k_{eff} < 0.2$. The normal condition based CSI is therefore 0.

Damaged (Failed) Elements

Damaged elements are modeled as a homogenized fuel and water mixture within the can cavity. Analysis trends for the homogenized contents are similar to those for the intact elements in a can. Bias and uncertainty adjusted system reactivity of the damaged fuel elements is higher than allowed for a full cask load (28 cans), but restricting the cask array under accident condition to three casks and limiting each cask's contents to 14 damaged fuel cans, seven in each of the top and base basket modules, and intact fuel assemblies or rod holders in the intermediate basket modules, produces acceptable reactivities as shown in Table 6.4.9-5. For the 3-cask array, the maximum calculated k_{eff} is 0.86961 ± 0.00081.

6.4.9.4 Single Cask Evaluation

The 10 CFR 71.55(b)(3) requires an evaluation of the NAC-LWT with the containment system fully reflected by water. The containment for the NAC-LWT is the cask inner shell. While no operating condition results in a removal of the cask outer shell and lead gamma shield, each of the partial flooding cases at four combinations of interior and exterior moderator is reevaluated by removing the lead and outer shells (including neutron shield), and reflecting the system by water at full density on the X, Y, and Z faces. Using the maximum reactivity model from Section 6.4.9.3, the calculated k_{eff} is 0.72674 ± 0.00087.

6.4.9.5 Code Bias and Code Bias Uncertainty Adjustments

PULSTAR fuel elements are similar to LWR fuel rods with a shorter active fuel length and smaller assembly array. While the enrichment for PULSTAR fuel is outside the enrichment range validated for LWR fuel, Figure 6.5.1-2 shows no statistical trend in k_{eff} versus enrichment. Further, any trend that may be postulated from the criticality benchmarks indicates a higher predicted k_{eff} value at higher enrichments. Therefore, code bias and code uncertainty adjustments are based on the LWR fuel assembly criticality benchmarks in Section 6.5.1.

A calculation of k_s under normal and accident conditions can now be made based on the previous results and based on the KENO-Va validation statistics presented in Section 6.5.1. The value k_s is calculated based on the KENO-Va Monte Carlo average plus any biases and uncertainties associated with the methods and the modeling, i.e.:

$$k_s = k_{eff} + \Delta k_{Bias} + \Delta k_{BU} + 2\sigma_{MC} \leq 0.95$$

In the validation presented in Section 6.5.1, a bias of ± 0.0052 and a 95/95 method uncertainty of ± 0.0087 were determined. Thus, the equation for k_s becomes as follows.

$$k_s = k_{eff} + 0.0139 + 2\sigma_{MC}$$

The k_s values for each evaluated payload are summarized in Table 6.4.9-6. The maximum k_s, 0.88513, results from a mixed loading of intact assemblies and canned elements.

For both normal and accident conditions, the calculated k_{eff} values, after correction for uncertainty, are well below the 0.95 limit. The analyses demonstrate that, including all calculational and mechanical uncertainties, an array of NAC-LWT casks with PULSTAR fuel remains subcritical under normal and accident conditions.

6.4.9.6 Allowable Cask Loading

Based on the results of the previous sections, the following cask loadings are permissible. For intact elements, any combination of assemblies and 4×4 rod inserts may be loaded into any module cell. Up to 14 damaged fuel cans each containing up to 25 PULSTAR fuel elements may be loaded in the top and base modules only; module cells loaded without cans may contain any combination of intact assemblies or 4×4 rod inserts. Each can is allowed the equivalent fissile material content of 25 fuel elements in either intact or damaged (failed) form. Damaged fuel may include fuel debris.

6.5 Criticality Benchmarks

The results of the criticality analyses presented in this chapter are corrected for bias and uncertainty resulting from the method using information obtained from the analysis of criticality benchmark experimental data.

6.5.1 CSAS25 Criticality Benchmark for LEU LWR Oxide Fuel

This section provides the validation of the CSAS25 criticality analysis sequence contained in Version 4.3 of the SCALE package. This validation is required by the criticality safety standards ANSI/ANS-8.1. The section describes the method, computer program and cross-section libraries used, experimental data, areas of applicability, and bias and margins of safety.

ANSI/ANS-8.17 prescribes the criterion to establish subcriticality safety margins. This criterion is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from calculation of benchmark criticality experiments using particular calculational method. If calculated k_{eff} values for criticality experiments exhibit trend with parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of system being evaluated.

Δk_s= allowance for:

a. statistical or convergence uncertainties, or both, in computation of k_s,

b. material and fabrication tolerances, and

c. geometric or material representations used in computational method.

Δk_c= margin for uncertainty in k_c which includes allowance for:

a. uncertainties in critical experiments,

b. statistical or convergence uncertainties, or both, in computation of k_c,

c. uncertainties resulting from extrapolation of k_c outside range of experimental data, and

d. uncertainties resulting from limitations in geometrical of material representations used in computational method.

Δk_m= arbitrary margin to ensure subcriticality of k_s.

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined additively.

Equation 1 can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1-k_c) - \Delta k_c \qquad (2)$$

Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$) and the definition of the bias ($\Delta k_{Bias} = 1-k_c$), the Equation 2 can then be written as:

$$k_s \leq 0.95 - \Delta k_s - \Delta k_{Bias} - \Delta k_{BU} \qquad (3)$$

where $\Delta k_{BU} = \Delta k_c$. Thus, the k_s (the maximum allowable value for k_{eff}) must be below 0.95 minus the bias, uncertainties in the bias, and uncertainties in the system being analyzed (i.e., Monte Carlo, mechanical, and modeling). This is an upper safety limit criteria often used in the DOE criticality safety community.

Alternatively, Equation 3 can be rewritten applying the bias and uncertainties to the k_{eff} of the system being analyzed as:

$$k_s \equiv k_{eff} + \Delta k_s + \Delta k_{Bias} + \Delta k_{BU} \leq 0.95 \qquad (4)$$

In Equation 4, k_{eff} replaces k_s, and k_s has been redefined as the effective multiplication factor of the system being analyzed, including the method bias and all uncertainties. This is a maximum calculated k_{eff} criteria often used in LWR spent fuel storage and transport analyses.

For use in criticality evaluations of LWR fuel in storage and transport casks, both k_{Bias} and Δk_{Bias} are evaluated below for KENO-Va with the 27-group ENDF/B-IV library.

6.5.1.1 Benchmark Experiments and Applicability

The criticality safety method is CSAS embedded in SCALE version 4.3 for the PC. CSAS includes the SCALE Material Information Processor, BONAMI-S, NITAWL-S, and KENO-Va. The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross-section processing codes. The BONAMI-S and NITAWL-S codes are used to prepare a resonance-corrected cross-section library in AMPX working format. The KENO-Va code uses Monte Carlo techniques to calculate the model k_{eff}. The 27-group ENDF/B-IV neutron cross-section library is used in this validation.

6.5.2 CSAS25 Criticality Benchmarks for Research Reactor Fuel Elements (MTR and DIDO)

In this section, the CSAS25 SCALE criticality analysis sequence is validated for use with high enrichment uranium (HEU), 80-90 wt % ^{235}U based, research reactor fuel. Not included in this validation section is Zirconium hydride based (TRIGA) fuel. This validation provides an estimate of the method bias and uncertainty to be applied in setting criticality safety limits for the NAC-LWT with MTR and DIDO fuel elements. Spiral type fuel and MOATA plate bundles are composed of MTR type fuel elements (flat or curved metallic plates with a uranium-aluminum alloy fuel meat within an aluminum clad). Bias established for the MTR and DIDO fuels is, therefore, applicable to the spiral fuel assemblies and plate bundles.

A subset of high enrichment critical experiments is selected from (Jordan) and (Johnson). (Johnson) contains detailed critical experiments with SPERT-D MTR plate fuel. The selected experiments are described in more detail below.

Five sets of critical experiments were selected from (Jordan). The first set included five experiments with various uranyl nitrate solutions in an unreflected cylindrical aluminum tank. Criticality was achieved by varying concentration, solution level or tank diameter. The second set included four experiments, which utilized the same solutions and tanks, but surrounded the tank with rectangular plexiglass reflector. The third set included seven experiments with various unreflected and reflected metal shapes. The fourth set included five experiments with various unreflected and reflected arrays of uranium metal cylinders in which criticality was achieved by spacing between arrays. The fifth set included four experiments with rectangular uranyl fluoride solution tanks made of aluminum in which criticality was achieved by adjusting tank spacing and solution height.

Five critical experiments were selected from (Johnson). The experiments selected were 4x4 arrays of MTR fuel plate type elements in which criticality was studied versus spacing between fuel elements. Criticality was achieved by setting up a 4x4 array with the given spacing and loading the outer row of fuel elements with a partial loading of fuel plates. The maximum loading of fuel plates in an element was 22. In order to preserve rectangular geometry, each partial element in a row contained the same number of fuel plates, ± 1 plate.

The results for the thirty cases executed on NAC's version of CSAS25 are shown in Table 6.5.1-1. The average, standard deviation, average Monte Carlo error, method bias and method uncertainty are evaluated as follows:

Average: $\bar{k} = \frac{1}{30}\sum_{1}^{30} k_i = 1.0044$

Raw Standard Deviation: $\sigma = \sqrt{\frac{1}{29}\sum_{1}^{30}(k_i - \bar{k})} = \pm\ 0.0089$

Average Monte Carlo Error: $\sigma_{mc} = \sqrt{\frac{1}{30}\sum_{1}^{30}\sigma_i} = \pm\ 0.0037$

Method Bias: $\Delta k_{bias} = 1 - \bar{k} = -0.0044$

Method Uncertainty: $\Delta k_{BU} = \sqrt{\sigma^2 - \sigma_{mc}^2} = \pm\ 0.0081$

95/95 One Sided Factor for 30 Cases: 2.23 (Owen)
95/95 Method Uncertainty: 2.23 * 0.0081 = 0.0181

These statistical results lead to the following equation for calculating criticality safety limits:

$$k_s = k_{nom} - 0.0044 + 0.0181 + 2\sigma_{mc}$$

where:

k_{nom} is the k_{eff} of the KENO-Va calculation

Δk_{BU} is the uncertainty associated with the benchmark calculations

σ_{mc} is the KENO-Va Monte Carlo Error associated with the calculated k_{eff} value

After conservatively neglecting the negative bias associated with the overprediction in k_{eff} produced by KENO-Va for these cases, the equation becomes:

$$k_s = k_{nom} + 0.0181 + 2\sigma_{mc}$$

No specific benchmarks are available for DIDO fuel assemblies. Since the fuel cylinders for the DIDO assembly are comprised of fuel plates similar in composition to those of the MTR element, the validation for the MTR element is considered to be applicable to the DIDO assembly.

6.5.3 CSAS25 Criticality Benchmarks for TRIGA Fuel Elements

Three core configurations presented in the TRIGA MARK II Benchmark (Mele) were modeled with KENO-Va and the 27 Group ENDF/B-IV neutron cross section library to establish the KENO-Va. bias as part of the SCALE 4.3 package for use on TRIGA fuel elements. The core analyzed is an LEU (low enriched fuel, 20%) fueled core. Due to the relatively high ^{235}U density in the fuel element, the trend of this analysis is expected to be similar for the HEU (high enriched uranium, 70 wt% ^{235}U) elements. The results are summarized below.

Configuration	keff ± σ
Core 132	1.01892 ± 0.00126
Core 133	1.02206 ± 0.00125
Core 134	1.01774 ± 0.00129
Average ± σ	1.0196 ± 0.0022

Thus, the bias for these critical experiments is an approximate 2% over-prediction. This over-prediction will be conservatively ignored in the k_s calculations. The 95/95 Uncertainty Factor from [Owen] for 3 data points is 7.656. Therefore, the uncertainty factory to be applied in the k_s calculation is the standard deviation from the critical experiments multiplied by the 95/95 factor for 3 data points, or 7.656 × 0.0022 = 0.0168.

A review of the calculations performed for the benchmark experiments shows that the average energy causing fission is approximately 0.05 eV. The average energy causing fission for the wet base case is 0.0999 eV, while for the dry base case it is 0.415 eV. Because the benchmark calculation and TRIGA 24 fissions occur at energies below 1 eV, it is reasonable to assume that the benchmarks are applicable to these cases.

In the case of the TRIGA fuel criticality evaluations, the basic form for the application of bias and uncertainty is:

$$k_s = k_{mc} + \Delta k_{\text{Meth Bias}} + \Delta k_{\text{Benchmark Uncertainty}} + \Delta k_{\text{Basket Tolerances}} + 2\sigma_{mc}$$

where:

k_{mc} - CSAS reported reactivity

$\Delta k_{\text{Meth Bias}}$- Method bias for TRIGA Fuel from benchmark calculations

$\Delta k_{\text{Benchmark Uncertainty}}$ - Uncertainty associated with the TRIGA benchmark calculations

$\Delta k_{\text{Basket Tolerances}}$ - Mechanical biases associated with the basket and fuel configuration

σ_{mc} - Uncertainty associated with the CSAS reported reactivity

Based on benchmark information, the k_s equation is written as follows:

$$k_s = k_{mc} + 0.0 + 0.0168 + \Delta k_{\text{Basket Tolerances}} + 2\sigma_{mc}$$

If the worst case fuel element and basket configuration are used, the above equation reduces to:

$$k_s = k_{mc} + 0.0168 + 2\sigma_{mc}$$

6.5.4 MCNP Criticality Benchmarks LEU Oxide and MOX LWR Fuels

The results of the criticality analyses presented in this chapter must be compared to the upper subcritical limit (USL). The USL accounts for bias and uncertainty resulting from the method using information obtained from the analysis of criticality benchmark experimental data.

Criticality code validation is performed for the Monte Carlo evaluation code and neutron cross-section libraries. Criticality validation is required by the criticality safety standard ANSI/ANS-8.1.

6.5.4.1 Benchmark Experiments and Applicability Discussion

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages" (NUREG), provides a guide to LWR criticality benchmark calculations and the determination of bias and subcritical limits in critical safety evaluations. In Section 2 of the NUREG, a series of LWR critical experiments is described in sufficient detail for independent modeling. In Section 3, the critical experiments are modeled, and the results (k_{eff} values) are presented. The method utilized in the NUREG is KENO-Va with the 44-group ENDF/B-V cross-section library embedded in SCALE 4.3. In Section 4, a guide for the determination of bias and subcritical safety limits is provided based on ANSI/ANS-8.1 and statistical analysis of the trending in the bias. Finally, guidelines for experiment selection and applicability are presented in Section 5. The approach outlined in Section 4 of the NUREG is described in detail herein and is implemented for MCNP5 with continuous energy ENDF/B-VI cross-sections.

NUREG/CR-6361 implements ANSI/ANS-8.1 criticality safety criterion as follows.

$$k_s \le k_c - \Delta k_s - \Delta k_c - \Delta k_m \quad \text{(Equation 1)}$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from a calculation of benchmark criticality experiments using a particular calculation method. If the calculated k_{eff} values for the criticality experiments exhibit a trend with an independent parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

Δk_s = allowance for the following:

- statistical or convergence uncertainties, or both, in computation of k_s
- material and fabrication tolerances
- geometric or material representations used in computational method

Δk_c = margin for uncertainty in k_c, which includes allowance for the following:

- uncertainties in critical experiments
- statistical or convergence uncertainties, or both, in computation of k_c
- uncertainties resulting from extrapolation of k_c outside range of experimental data
- uncertainties resulting from limitations in geometrical or material representations used in the computational method

Δk_m = arbitrary administrative margin to ensure subcriticality of k_s

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as shown.

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1 - k_c) - \Delta k_c \quad \text{(Equation 2)}$$

Noting that the definition of the bias is $\beta = 1 - k_c$, Equation 2 can be written as shown.

$$k_s + \Delta k_s \leq 1 - \Delta km - \beta - \Delta\beta \qquad \text{(Equation 3)}$$

where:

$$\Delta\beta = \Delta k_c$$

Thus, the maximum allowable value for k_{eff} plus uncertainties in the system being analyzed must be below 1 minus an administrative margin (typically 0.05), which includes the bias and the uncertainty in the bias. This can also be written as shown.

$$k_s + \Delta k_s \leq \text{Upper Subcritical Limit (USL)} \qquad \text{(Equation 4)}$$

where:

$$USL \equiv 1 - \Delta k_m - \beta - \Delta\beta \quad \text{(Equation 5)}$$

This is the USL criterion as described in Section 4 of NUREG/CR-6361. Two methods are prescribed for the statistical determination of the USL. The "Confidence Band with Administrative Margin (USL-1)" approach is implemented here and is referred to generically as USL. A $\Delta k_m = 0.05$ and a lower confidence band are specified based on a linear regression of k_{eff} as a function of some system parameter. As recommended in NUREG/CR-6361, a simple linear regression is performed on each system parameter, and the line with the greatest correlation is used to functionalize β.

Application specific sections (e.g., low enriched uranium, MOX) contains the the list of criticality benchmarks employed in the validation of MCNP with its continuous energy neutron

cross-section libraries and the processing of the experimental results into the USL. Included in the subsequent sections are linear fits of reactivity (k_{eff}) to each of the system parameters. Experiments were chosen to reflect the fuel geometry and materials as closely as available.

6.5.4.2 LEU (Maximum 5 wt % ^{235}U in UO_2) Results of Benchmark Calculations

The range of parameters included in the low enriched uranium (LEU) benchmarks is shown in Table 6.5.4-1. Experiments are chosen to reflect the fuel evaluated for shipment. This includes the use of arrays of low enriched uranium oxide fuel rods with light water moderation. To cover potential borated water conditions within spent fuel pools or absorbers placed into the basket experiments with criticality control by spacing, borated moderator and/or borated absorber panels and tubes are included in the benchmarks effort. Trending in k_{eff} was evaluated for the following independent variables: wt % ^{235}U, rod pitch, H/U volume ratio, energy of the average neutron lethargy causing fission (EALCF), ^{10}B loading of the absorber sheet, and soluble boron loading. No statistically significant trends were found for any of the system parameters. USLs are, therefore, generated for each of the independent variables. A minimum USL covering the range of applicability of the benchmark set is determined.

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. The linear correlation coefficient, R, is calculated by taking the square root of the R^2 value. In particular, the correlation coefficient, R, is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If R is +1, a perfect linear relationship with a positive slope is indicated. If R is -1, a perfect linear relationship with a negative slope is indicated. When R is 0, no linear relationship is indicated.

Table 6.5.4-2 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter. Linear fits and correlation constants are based on the 183 data-point evaluation sets plotted in Section 6.5.4.3. The cluster gap plot is limited to the 137 data points for experiments containing multiple fuel rod clusters. Single fuel rod cluster experiments documented in LEU-COMP-THERM sets 06, 14, 35 and 50, in addition to LEU-COMP-THERM experiments 01-01, 02-01 to -03, and 08-01 to -15, were, therefore, excluded from the cluster gap study. The 183 data points evaluated for the remaining parameters represent the complete set of experiments listed in Section 6.5.4.3 minus the three high energy lethargy experiments above 0.35 eV (Experiments LEU-COMP-THERM 14-05, -06 and -07). The addition of these points, while not resulting in a significant linear fit, produces a noticeable slope to the USL correlation not representative of the remaining data fits. As this increased slope results in a higher USL, it is acceptable to discard these data points. The three higher energy points are removed from all independent variables for consistency.

As there is no significant correlation to any of the independent variables, the USL for each independent variable is calculated and shown with its range of applicability in Table 6.5.4-2. A sample output for EALCF is shown in Figure 6.5.4-1. Uncertainties included in the USLSTATS evaluation are the Monte Carlo uncertainty associated with the reactivity calculation and experimental uncertainty that was provided in the literature for each of the cases.

Based on all the independent variable correlations, a lower limit constant USL of 0.9376 may be applied. The range of applicability (area of applicability) of this limit may be extended to 5 wt % enriched fuel, as the correlation shows no significant trend with enrichment between 2.35 and 4.74 wt %, and that the limited trending observed increases the USL. Extending the range of applicability for the average neutron lethargy is based on a minimal, but positive, trend of the USL versus EALCF. Studies, including additional data points up to 0.7722 eV, indicate that the trending continues to the higher energy levels.

Figure 6.5.4-1 LEU USLSTATS Output for EALCF

```
*************************************************************************
                    Version 1.4, April 23, 2003
                    Oak Ridge National Laboratory
*************************************************************************
   Input to statistical treatment from file:enrich-183.in
Title: keff vs enrichment
   Proportion of the population = .995
   Confidence of fit            = .950
   Confidence on proportion     = .950
   Number of observations       =  183
   Minimum value of closed band =    0.00
   Maximum value of closed band =    0.00
   Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
2.35000E+00	9.94910E-01	3.42000E-03	2.35000E+00	9.95090E-01	3.46000E-03
2.35000E+00	9.92830E-01	3.38000E-03	2.35000E+00	9.92520E-01	3.47000E-03
2.35000E+00	9.98060E-01	3.38000E-03	2.35000E+00	9.95620E-01	3.50000E-03
2.35000E+00	9.96550E-01	3.42000E-03	2.35000E+00	9.93130E-01	3.55000E-03
2.35000E+00	9.89310E-01	3.44000E-03	2.35000E+00	9.98130E-01	3.58000E-03
2.35000E+00	9.95340E-01	3.41000E-03	2.35000E+00	9.96700E-01	3.56000E-03
2.35000E+00	9.93880E-01	3.44000E-03	2.35000E+00	9.93830E-01	3.55000E-03
2.35000E+00	9.89690E-01	3.36000E-03	2.35000E+00	9.92770E-01	3.47000E-03
4.30600E+00	9.95160E-01	2.79000E-03	2.35000E+00	9.92920E-01	3.50000E-03
4.30600E+00	9.93670E-01	2.54000E-03	2.35000E+00	9.96410E-01	3.46000E-03
4.30600E+00	9.96340E-01	2.76000E-03	2.35000E+00	9.93060E-01	3.49000E-03
4.30600E+00	9.93110E-01	2.64000E-03	2.35000E+00	9.96500E-01	3.45000E-03
4.30600E+00	9.93000E-01	2.49000E-03	2.35000E+00	9.94680E-01	3.50000E-03
2.59600E+00	9.92680E-01	2.10000E-03	2.35000E+00	9.93300E-01	3.47000E-03
2.59600E+00	9.93190E-01	2.14000E-03	2.35000E+00	9.91810E-01	3.46000E-03
2.59600E+00	9.92990E-01	2.13000E-03	2.35000E+00	9.93920E-01	3.47000E-03
2.59600E+00	9.94790E-01	2.13000E-03	2.35000E+00	9.95560E-01	3.55000E-03
2.59600E+00	9.93100E-01	2.12000E-03	2.35000E+00	9.94540E-01	3.51000E-03
2.59600E+00	9.93240E-01	2.12000E-03	2.35000E+00	9.94490E-01	3.47000E-03
2.59600E+00	9.91990E-01	2.12000E-03	2.35000E+00	9.91300E-01	3.52000E-03
2.59600E+00	9.93820E-01	2.12000E-03	2.35000E+00	9.94800E-01	3.47000E-03
2.59600E+00	9.94450E-01	2.12000E-03	2.35000E+00	9.93500E-01	3.60000E-03
2.59600E+00	9.95440E-01	2.13000E-03	2.35000E+00	9.94000E-01	3.45000E-03
2.59600E+00	9.94410E-01	2.12000E-03	2.35000E+00	9.96280E-01	3.53000E-03
2.59600E+00	9.93920E-01	2.15000E-03	2.35000E+00	9.92620E-01	3.45000E-03
2.59600E+00	9.95090E-01	2.14000E-03	2.35000E+00	9.94100E-01	3.53000E-03
2.59600E+00	9.93780E-01	2.12000E-03	2.35000E+00	9.96470E-01	3.52000E-03
2.59600E+00	9.95040E-01	2.14000E-03	2.35000E+00	9.93600E-01	3.47000E-03
2.59600E+00	9.94380E-01	2.11000E-03	2.35000E+00	9.97020E-01	3.49000E-03
2.59600E+00	9.95730E-01	2.12000E-03	2.35000E+00	9.94970E-01	3.50000E-03
2.59600E+00	9.94270E-01	2.14000E-03	2.35000E+00	9.91950E-01	3.55000E-03
2.45900E+00	9.98350E-01	1.34000E-03	2.59600E+00	9.93410E-01	1.93000E-03
2.45900E+00	9.96860E-01	1.36000E-03	2.59600E+00	9.91310E-01	2.05000E-03
2.45900E+00	9.99310E-01	1.24000E-03	4.73800E+00	9.95860E-01	4.36000E-03
2.45900E+00	9.97950E-01	1.36000E-03	4.73800E+00	9.93580E-01	4.53000E-03
2.45900E+00	9.97650E-01	1.38000E-03	4.73800E+00	9.95390E-01	4.58000E-03
2.45900E+00	9.96990E-01	1.35000E-03	4.73800E+00	9.92370E-01	4.54000E-03
2.45900E+00	9.97230E-01	1.37000E-03	4.73800E+00	9.91440E-01	4.62000E-03
2.45900E+00	9.96590E-01	1.40000E-03	4.73800E+00	9.98780E-01	4.82000E-03
2.45900E+00	9.95260E-01	1.40000E-03	4.73800E+00	9.94180E-01	4.94000E-03
2.45900E+00	9.97450E-01	1.36000E-03	4.73800E+00	9.92400E-01	4.90000E-03
2.45900E+00	9.97590E-01	1.38000E-03	4.73800E+00	9.96930E-01	4.98000E-03
2.45900E+00	9.97650E-01	1.36000E-03	4.73800E+00	9.91370E-01	5.05000E-03
2.45900E+00	9.98880E-01	1.39000E-03	2.35000E+00	9.92500E-01	2.34000E-03
2.45900E+00	9.97350E-01	1.37000E-03	2.35000E+00	9.95140E-01	2.43000E-03
2.45900E+00	9.97580E-01	1.39000E-03	2.35000E+00	9.92190E-01	2.33000E-03
2.45900E+00	9.97720E-01	1.39000E-03	2.35000E+00	9.94760E-01	2.40000E-03
2.45900E+00	9.96910E-01	1.35000E-03	2.35000E+00	9.94690E-01	3.67000E-03
4.30600E+00	9.95480E-01	2.84000E-03	2.35000E+00	9.94340E-01	2.49000E-03
4.30600E+00	9.93430E-01	2.78000E-03	2.35000E+00	9.93190E-01	2.39000E-03
4.30600E+00	9.93300E-01	2.81000E-03	4.73800E+00	9.93300E-01	1.28000E-03
4.30600E+00	9.93710E-01	2.85000E-03	4.73800E+00	9.93400E-01	1.23000E-03
4.30600E+00	9.95930E-01	2.73000E-03	4.73800E+00	9.94890E-01	1.25000E-03
4.30600E+00	9.92950E-01	2.85000E-03	4.73800E+00	9.93190E-01	1.25000E-03
4.30600E+00	9.96160E-01	2.89000E-03	4.73800E+00	9.93060E-01	1.28000E-03
4.30600E+00	9.93890E-01	2.73000E-03	2.45900E+00	9.91330E-01	2.03000E-03
4.30600E+00	9.95710E-01	2.96000E-03	2.45900E+00	9.95970E-01	2.43000E-03
4.30600E+00	9.93190E-01	2.60000E-03	2.45900E+00	9.95550E-01	2.42000E-03
4.30600E+00	9.93780E-01	2.75000E-03	2.45900E+00	9.94860E-01	2.42000E-03
4.30600E+00	9.92630E-01	2.84000E-03	2.45900E+00	9.95040E-01	2.42000E-03
4.30600E+00	9.95660E-01	2.75000E-03	2.45900E+00	9.95420E-01	2.42000E-03
4.30600E+00	9.94310E-01	2.82000E-03	2.45900E+00	9.95300E-01	2.42000E-03

Figure 6.5.4-1 LEU USLSTATS Output for EALCF (cont'd)

4.30600E+00	9.96390E-01	2.95000E-03	2.45900E+00	9.95070E-01	2.42000E-03
4.30600E+00	9.96860E-01	2.79000E-03	2.45900E+00	9.93680E-01	1.93000E-03
4.30600E+00	9.97160E-01	2.68000E-03	2.45900E+00	9.92100E-01	1.93000E-03
4.30600E+00	9.92370E-01	2.86000E-03	2.45900E+00	9.94470E-01	1.93000E-03
4.30600E+00	9.97190E-01	2.81000E-03	2.45900E+00	9.90730E-01	1.93000E-03
4.30600E+00	9.94340E-01	2.76000E-03	2.45900E+00	9.86520E-01	2.23000E-03
4.30600E+00	9.96920E-01	2.79000E-03	2.45900E+00	9.86340E-01	1.93000E-03
4.30600E+00	9.96060E-01	2.83000E-03	2.45900E+00	9.90420E-01	2.42000E-03
4.30600E+00	9.97400E-01	2.94000E-03	2.45900E+00	9.89740E-01	2.03000E-03
4.30600E+00	9.92810E-01	2.69000E-03	2.45900E+00	9.91520E-01	2.72000E-03
4.30600E+00	9.92560E-01	2.88000E-03	2.45900E+00	9.90290E-01	2.13000E-03
4.30600E+00	9.93650E-01	2.88000E-03	2.45900E+00	9.89270E-01	1.93000E-03
4.30600E+00	9.94970E-01	2.85000E-03	2.60000E+00	9.95710E-01	1.42000E-03
2.45900E+00	9.94820E-01	3.21000E-03	2.60000E+00	9.96180E-01	1.42000E-03
2.45900E+00	9.94940E-01	3.21000E-03	2.60000E+00	9.95340E-01	1.52000E-03
2.45900E+00	9.95140E-01	3.21000E-03	2.60000E+00	9.95470E-01	1.52000E-03
2.45900E+00	9.95640E-01	3.21000E-03	2.60000E+00	9.96910E-01	1.42000E-03
2.45900E+00	9.95080E-01	3.21000E-03	2.60000E+00	9.96140E-01	1.42000E-03
2.45900E+00	9.95260E-01	3.21000E-03	2.60000E+00	9.95890E-01	1.42000E-03
2.45900E+00	9.95200E-01	3.21000E-03	2.60000E+00	9.96240E-01	1.62000E-03
4.30600E+00	9.94020E-01	1.92000E-03	2.60000E+00	9.96670E-01	1.52000E-03
4.30600E+00	9.94460E-01	1.91000E-03	2.60000E+00	9.96760E-01	1.62000E-03
4.30600E+00	9.93550E-01	1.91000E-03	2.60000E+00	9.96370E-01	1.62000E-03
4.30600E+00	9.94010E-01	1.91000E-03	2.60000E+00	9.96430E-01	1.72000E-03
4.30600E+00	9.92810E-01	3.27000E-03	2.60000E+00	9.97010E-01	1.62000E-03
4.30600E+00	9.94960E-01	1.91000E-03	2.60000E+00	9.96500E-01	1.62000E-03
4.30600E+00	9.93780E-01	1.90000E-03	2.60000E+00	9.96340E-01	1.62000E-03
4.30600E+00	9.96680E-01	1.95000E-03	2.60000E+00	9.96580E-01	1.71000E-03
4.30600E+00	9.85950E-01	7.71000E-03	2.60000E+00	9.96450E-01	1.62000E-03
2.35000E+00	9.94940E-01	3.54000E-03			

```
chi =  2.5464 (upper bound = 9.49). The data tests normal.
      Output from statistical treatment
keff vs enrichment
 Number of data points (n)                          183
 Linear regression, k(X)                            0.9950 + (-1.5719E-04)*X
 Confidence on fit (1-gamma) [input]                 95.0%
 Confidence on proportion (alpha) [input]            95.0%
 Proportion of population falling above
 lower tolerance interval (rho) [input]              99.5%
 Minimum value of X                                  2.3500E+00
 Maximum value of X                                  4.7380E+00
 Average value of X                                  3.0597E+00
 Average value of k                                   0.99453
 Minimum value of k                                   0.98595
 Variance of fit, s(k,X)^2                           5.0408E-06
 Within variance, s(w)^2                             7.8633E-06
 Pooled variance, s(p)^2                             1.2904E-05
 Pooled std. deviation, s(p)                         3.5922E-03
 C(alpha,rho)*s(p)                                   1.5554E-02
 student-t @ (n-2,1-gamma)                           1.64500E+00
 Confidence band width, W                            5.9793E-03
 Minimum margin of subcriticality, C*s(p)-W          9.5746E-03

 Upper subcritical limits: (  2.3500     <= X <=    4.7380    )
 ***** *********** *******
 USL Method 1 (Confidence Band with
 Administrative Margin)            USL1 = 0.9390 + (-1.5719E-04)*X
 USL Method 2 (Single-Sided Uniform
 Width Closed Interval Approach)   USL2 = 0.9795 + (-1.5719E-04)*X
 USLs Evaluated Over Range of Parameter X:
 **** ********* **** ***** ** ********* **

 X:  2.35E+0  2.69E+0  3.03E+0  3.37E+0  3.71E+0  4.06E+0  4.40E+0  4.74E+0
     ---------------------------------------------------------------------
USL-1:  0.9387   0.9386   0.9386   0.9385   0.9385   0.9384   0.9383   0.9383
USL-2:  0.9791   0.9790   0.9790   0.9789   0.9789   0.9788   0.9788   0.9787
     ---------------------------------------------------------------------
```

Table 6.5.4-1 LEU Range of Applicability for Complete Set of 186 Benchmark Experiments

Parameter	Minimum	Maximum
Enrichment (wt % ^{235}U)	2.350%	4.738%
Fuel rod pitch (cm)	1.30	2.54
Fuel pellet outer diameter (cm)	0.790	1.265
Fuel rod diameter (cm)	0.9400	1.4172
$H/^{235}U$ atom ratio	72.7	403.9
Soluble boron (ppm by weight)	0	4986
Cluster gap (cm)	1.206	13.750
Boron (^{10}B) plate loading (g/cm^2)	0.0000	0.0670
Energy of average neutron lethargy causing fission (eV)	0.09781	0.77219

Table 6.5.4-2 LEU Correlation Coefficients and USLs for Benchmark Experiments

Variable	R^2	R	Range of Applicability	USLSTATS Correlation	USL Low	USL High
Enrichment (wt % ^{235}U)	0.00410	0.064	2.35<=X<=4.738	0.9390-1.57E-04X	0.9382	0.9386
Fuel rod pitch (cm)	0.00150	0.039	1.3<=X<=2.54	0.9380+2.64E-04X	0.9383	0.9386
Fuel pellet outer diameter (cm)	0.00260	0.051	0.79<=X<=1.265	0.9376+8.25E-04X	0.9382	0.9386
Fuel rod diameter (cm)	0.00380	0.062	0.94<=X<=1.4172	0.9372+1.01E-03X	0.9381	0.9386
$H/^{235}U$ atom ratio	3.00E-06	0.002	106.2<=X<=403.9	0.9386-4.74E-08X	0.9385	0.9385
Soluble boron (ppm by weight)	0.01730	0.132	0<=X<=4986	0.9379+3.96E-07X	0.9379	0.9398
Cluster gap (cm)	0.01940	0.139	1.2<=X<=13.8	0.9375+9.82E-05X	0.9376	0.9388
Boron (^{10}B) plate loading (g/cm^2)	0.00006	0.008	0<=X<=0.067	0.9382-1.37E-03X	0.9381	0.9382
Energy of average neutron lethargy causing fission (eV)	0.00900	0.095	0.09781<=X<=0.3447	0.9379+3.45E-03X	0.9382	0.9390

6.5.4.3 LEU (Maximum 5 wt % ^{235}U in UO_2) Criticality Benchmarks

From the International Handbook of Evaluated Criticality Safety Benchmark Experiments, 186 experiments are selected as the basis of the MCNP benchmarking. Experiments were selected for compatibility of materials and geometry with the spent fuel casks. Of particular interest are benchmarks with arrays of low enriched uranium oxide fuel rods.

MCNP benchmark cases represent a collection of files composed of inputs directly obtained from references (with cross-section sets adjusted to those used in the cask analysis), NAC modified input files representing unique geometries based on reference input files, and input files constructed from the experimental material and geometry information. All cases were reviewed on a "preparer/checker" principle for modeling consistency with the cask models and the choice of code options. Due to large variations in the benchmark complexities, not all options employed in the cask models are reflected in each of the benchmarks (e.g., UNIVERSE structure). A review of the criticality results did not indicate any result trend due to particular modeling choices (e.g., using the UNIVERSE structure versus a single universe, or employing KSRC versus SDEF sampling).

Key system parameters, the experimental uncertainty, and calculated k_{eff} and σ for each experiment are shown in Table 6.5.4-3. Stochastic Monte Carlo error is kept within ±0.2% and each output is checked to assure that the MCNP built-in statistical checks on the results are passed and that all fissile material is sampled.

Scatter plots of k_{eff} versus system parameters for 183 data point sets (full set minus three high lethargy points above 0.35 eV) are created (see Figure 6.5.4-2 through Figure 6.5.4-10). Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (R^2) to statistically indicate any trend or lack thereof. Scatter plates are created for k_{eff} versus the following.

- Enrichment in ^{235}U (wt % ^{235}U)
- Fuel rod pitch (cm)
- Fuel pellet outer diameter (cm)
- Fuel rod outer diameter (cm)
- Hydrogen/uranium (^{235}U) atom ratio
- Soluble boron (ppm by weight)
- Cluster gap spacing (spacing between assemblies in cm)
- Boron (^{10}B) plate loading (g/cm^2)
- Energy of average neutron lethargy causing fission (eV)

Figure 6.5.4-2 k_{eff} versus Fuel Enrichment (LEU)

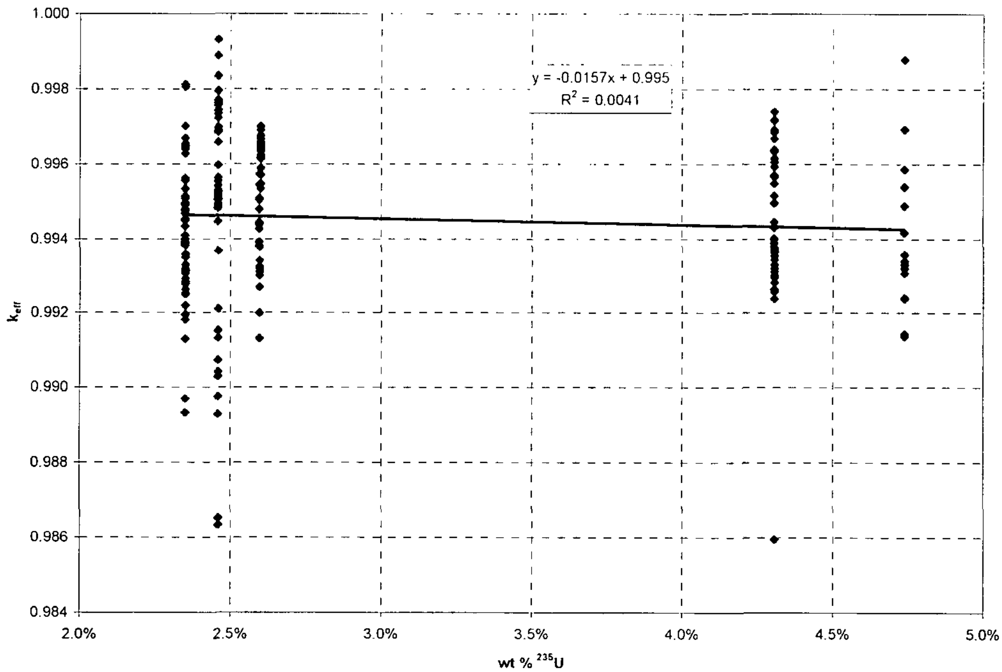


Figure 6.5.4-3 k_{eff} versus Rod Pitch (LEU)

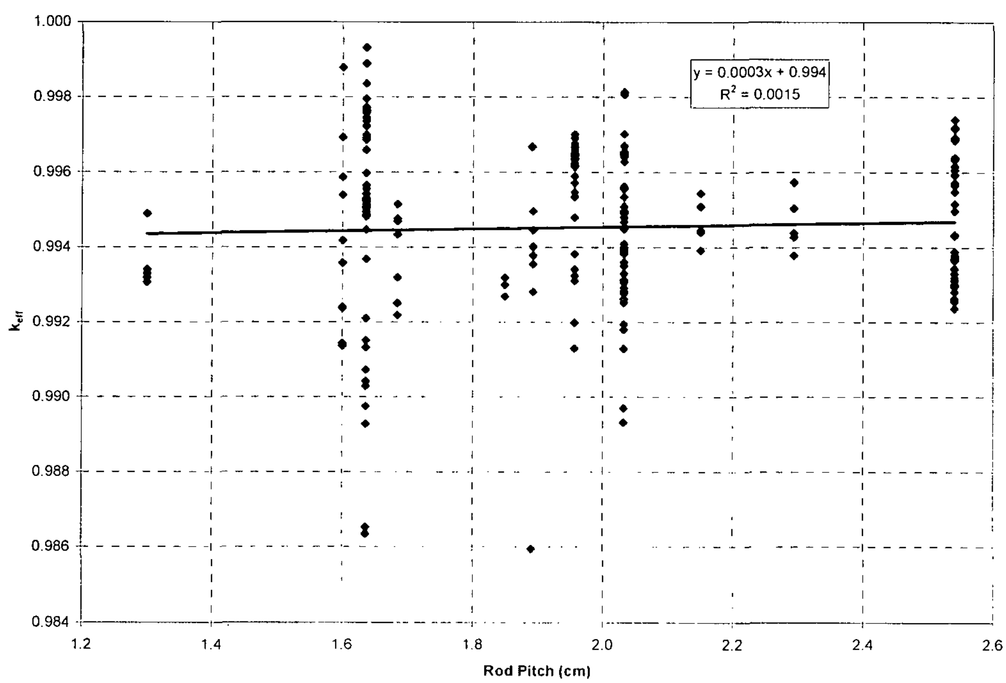


Figure 6.5.4-4 k_{eff} versus Fuel Pellet Diameter (LEU)

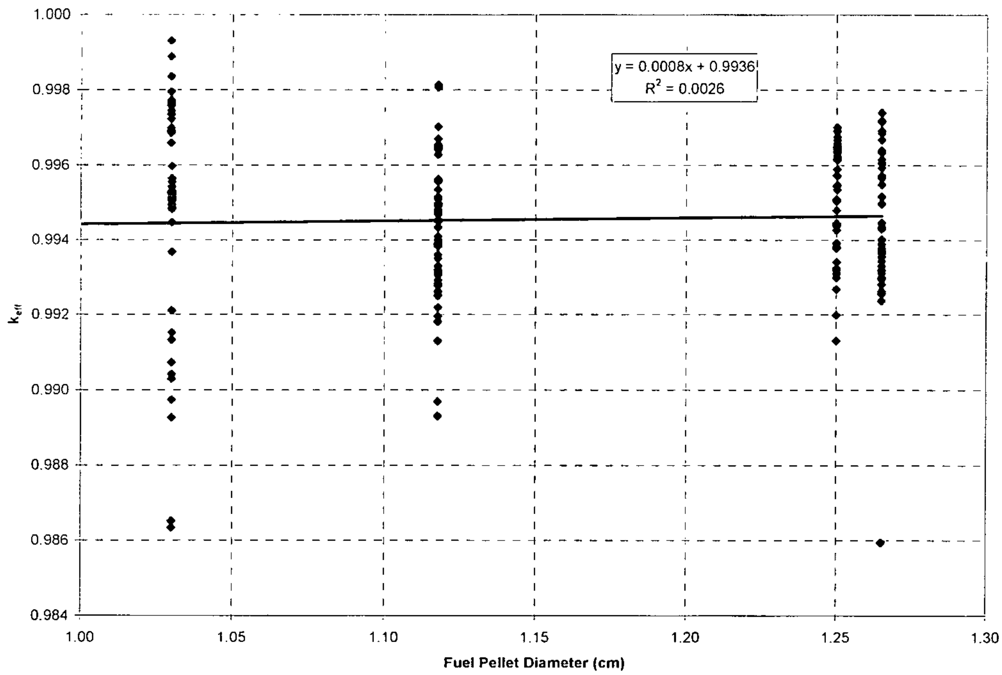


Figure 6.5.4-5 k_{eff} versus Fuel Rod Outside Diameter (LEU)

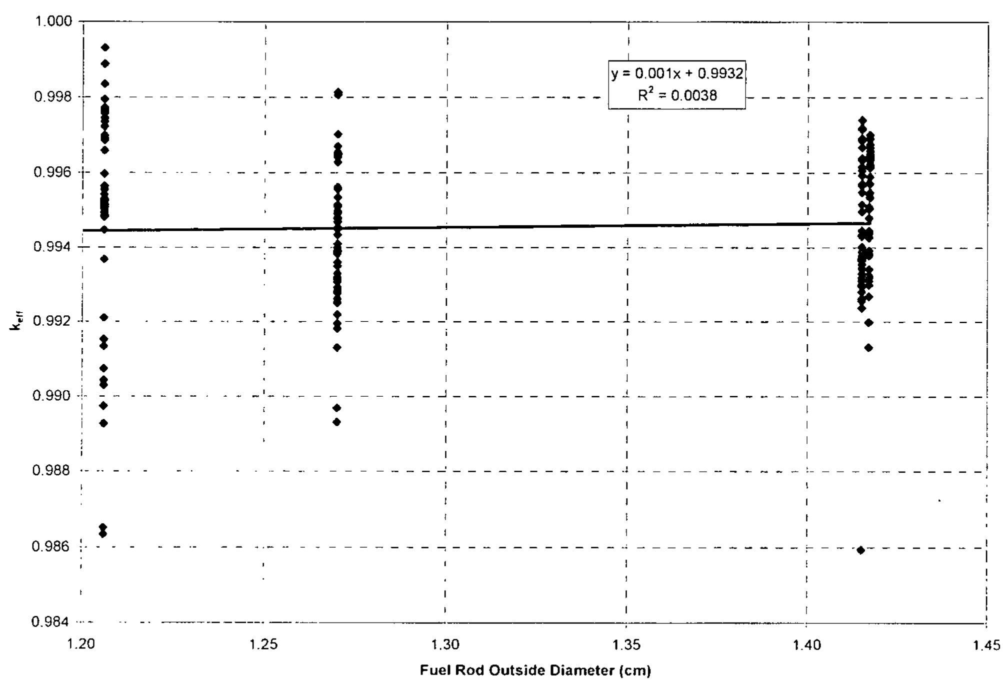


Figure 6.5.4-6 k_{eff} versus Hydrogen/^{235}U Atom Ratio (LEU)

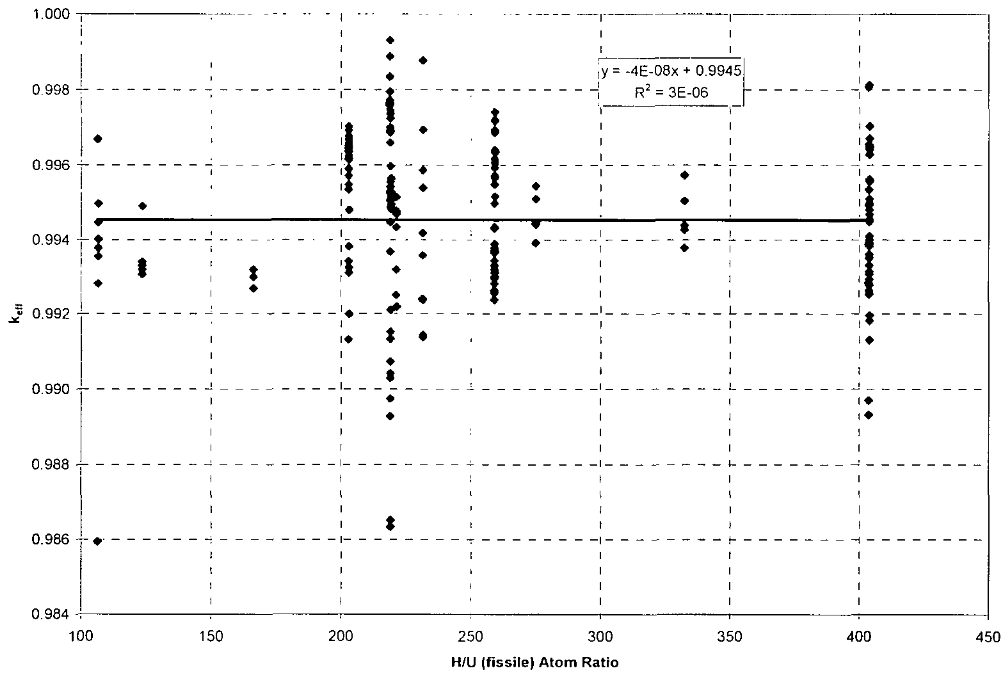


Figure 6.5.4-7 k_{eff} versus Soluble Boron Concentration (LEU)

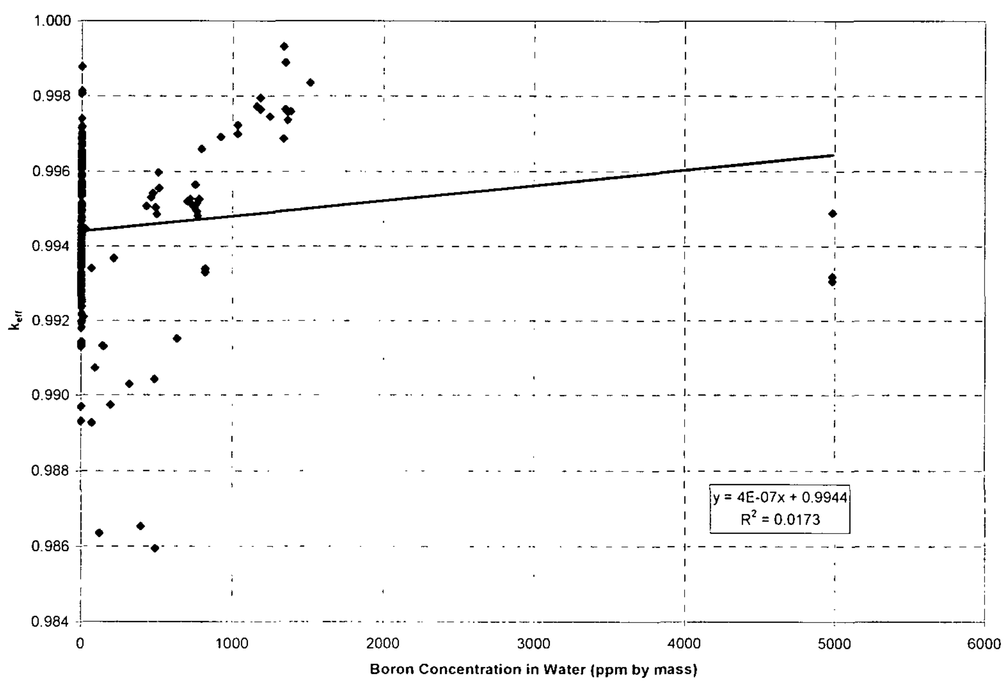


Figure 6.5.4-8 k_{eff} versus Cluster Gap Thickness (LEU)

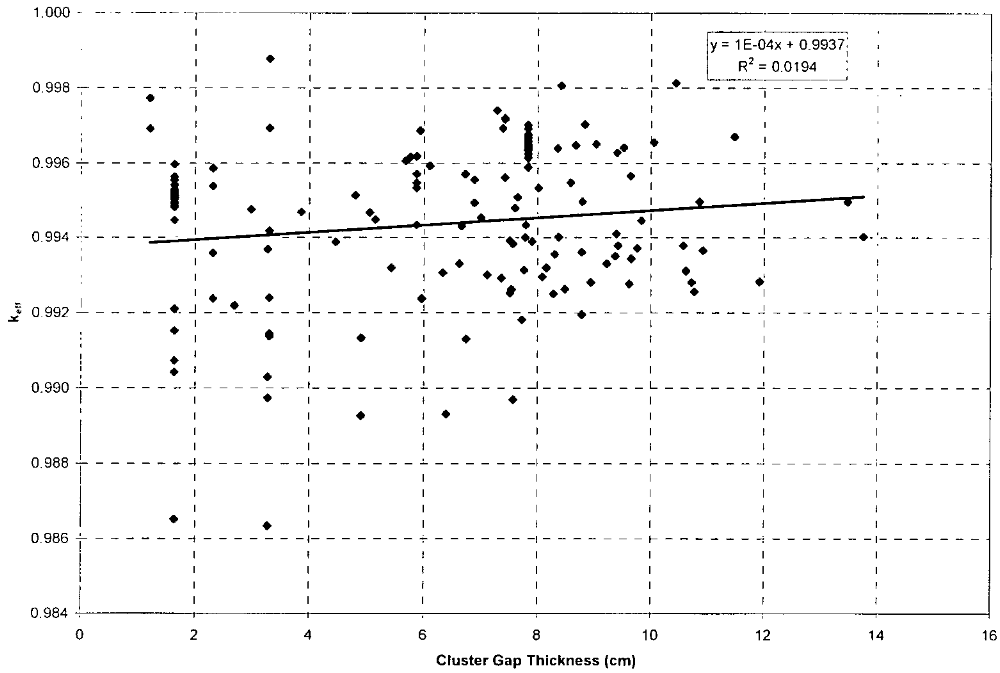


Figure 6.5.4-9 k_{eff} versus ^{10}B Plate Loading (LEU)

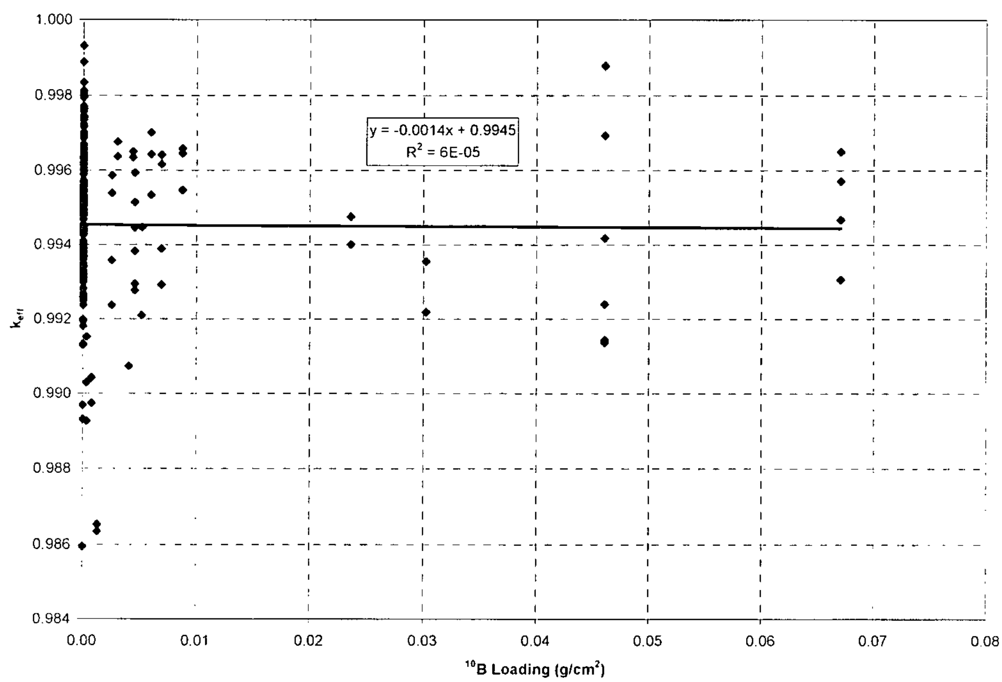


Figure 6.5.4-10 k_{eff} versus Energy of Average Neutron Lethargy Causing Fission (LEU)

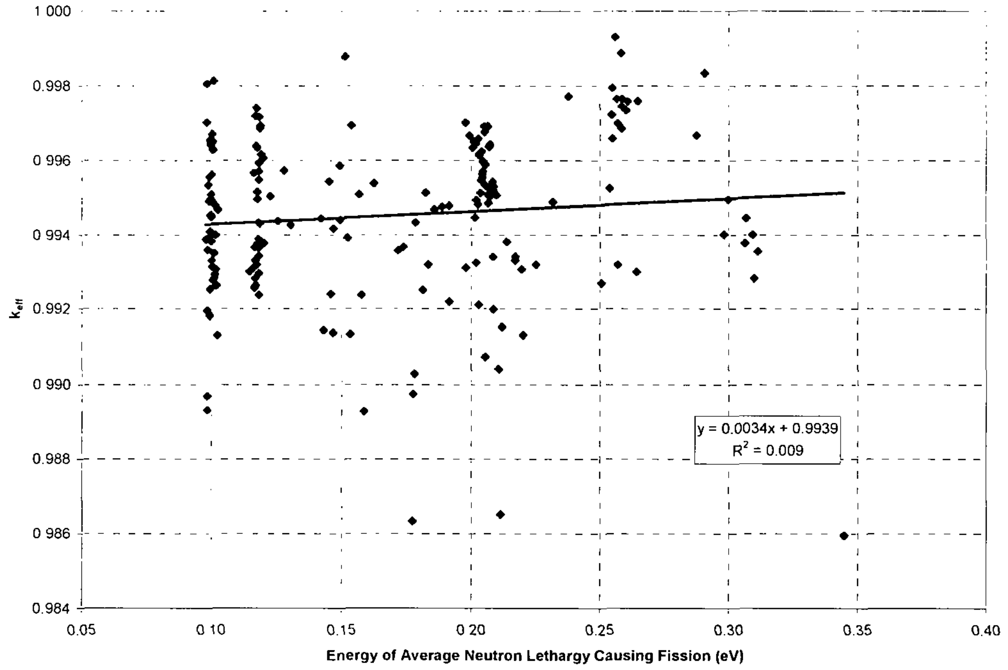


Table 6.5.4-3 LEU MCNP Validation Statistics

Case	1.01	1.02	1.03	1.04	1.05	1.06	1.07	1.08
Clusters	1	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%
Pitch (cm)	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032
Fuel OD (cm)	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118
Clad OD (cm)	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	404	404	404	404	404	404	404	404
Soluble B (ppm)	--	--	--	--	--	--	--	--
Absorber Type	--	--	--	--	--	--	--	--
Cluster Gap (cm)	--	11.9	8.4	10.1	6.4	8.0	4.5	7.6
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--
EALCF (MeV)	9.916E-8	1.010E-7	9.838E-8	9.933E-8	9.837E-8	9.874E-8	9.781E-8	9.826E-8
Exp. σ	0.0030	0.0030	0.0030	0.0030	0.0030	0.0030	0.0031	0.0030
k_{eff}	0.99491	0.99283	0.99806	0.99655	0.98931	0.99534	0.99388	0.98969
σ	0.00165	0.00155	0.00155	0.00165	0.00169	0.00162	0.00150	0.00152

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	2.01	2.02	2.03	2.04	2.05
Clusters	1	1	1	3	3
Enrichment (wt % ^{235}U)	4.31%	4.31%	4.31%	4.31%	4.31%
Pitch (cm)	2.540	2.540	2.540	2.540	2.540
Fuel OD (cm)	1.265	1.265	1.265	1.265	1.265
Clad OD (cm)	1.415	1.415	1.415	1.415	1.415
Clad Material	Al	Al	Al	Al	Al
H/U (fissile)	259	259	259	259	259
Soluble B (ppm)	--	--	--	--	--
Absorber Type	--	--	--	--	--
Cluster Gap (cm)	--	--	--	10.6	7.1
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--
EALCF (MeV)	1.177E-7	1.164E-7	1.175E-7	1.161E-7	1.146E-7
Exp. σ	0.0020	0.0020	0.0020	0.0018	0.0019
k_{eff}	0.99516	0.99367	0.99634	0.99311	0.99300
σ	0.00195	0.00157	0.00190	0.00193	0.00161

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	6.01	6.02	6.03	6.04	6.05	6.06	6.07	6.08	6.09
Clusters	1	1	1	1	1	1	1	1	1
Enrichment (wt % ^{235}U)	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%
Pitch (cm)	1.849	1.849	1.849	1.956	1.956	1.956	1.956	1.956	2.150
Fuel OD (cm)	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250
Clad OD (cm)	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	166	166	166	203	203	203	203	203	275
Soluble B (ppm)	--	--	--	--	--	--	--	--	--
Absorber Type	--	--	--	--	--	--	--	--	--
Cluster Gap (cm)	--	--	--	--	--	--	--	--	--
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--	--
EALCF (MeV)	2.506E-7	2.568E-7	2.642E-7	1.915E-7	1.978E-7	2.018E-7	2.085E-7	2.136E-7	1.422E-7
Exp. σ	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020
k_{eff}	0.99268	0.99319	0.99299	0.99479	0.99310	0.99324	0.99199	0.99382	0.99445
σ	0.00065	0.00076	0.00074	0.00074	0.00069	0.00070	0.00071	0.00071	0.00069

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	6.10	6.11	6.12	6.13	6.14	6.15	6.16	6.17	6.18
Clusters	1	1	1	1	1	1	1	1	1
Enrichment (wt % ^{235}U)	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%
Pitch (cm)	2.150	2.150	2.150	2.150	2.293	2.293	2.293	2.293	2.293
Fuel OD (cm)	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250
Clad OD (cm)	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	275	275	275	275	332	332	332	332	332
Soluble B (ppm)	--	--	--	--	--	--	--	--	--
Absorber Type	--	--	--	--	--	--	--	--	--
Cluster Gap (cm)	--	--	--	--	--	--	--	--	--
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--	--
EALCF (MeV)	1.453E-7	1.496E-7	1.523E-7	1.568E-7	1.202E-7	1.227E-7	1.257E-7	1.280E-7	1.306E-7
Exp. σ	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020	0.0020
k_{eff}	0.99544	0.99441	0.99392	0.99509	0.99378	0.99504	0.99438	0.99573	0.99427
σ	0.00073	0.00071	0.00078	0.00076	0.00070	0.00075	0.00067	0.00070	0.00076

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	8.01	8.02	8.03	8.04	8.05	8.06	8.07	8.08
Clusters	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3
Enrichment (wt % ^{235}U)	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%
Pitch (cm)	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636
Fuel OD (cm)	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
Clad OD (cm)	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	219	219	219	219	219	219	219	219
Soluble B (ppm)	1511	1336	1336	1182	1182	1033	1033	794
Absorber Type	--	--	--	--	--	--	--	--
Cluster Gap (cm)	--	--	--	--	--	--	--	--
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--
EALCF (MeV)	2.907E-7	2.583E-7	2.559E-7	2.548E-7	2.566E-7	2.568E-7	2.544E-7	2.548E-7
Exp. σ	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012
k_{eff}	0.99835	0.99686	0.99931	0.99795	0.99765	0.99699	0.99723	0.99659
σ	0.00060	0.00063	0.00032	0.00063	0.00069	0.00061	0.00066	0.00073

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	8.09	8.10	8.11	8.12	8.13	8.14	8.15	8.16	8.17
Clusters	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	3 x 3	5	5 x 5
Enrichment (wt % ^{235}U)	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%
Pitch (cm)	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636
Fuel OD (cm)	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
Clad OD (cm)	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	219	219	219	219	219	219	219	219	219
Soluble B (ppm)	779	1245	1384	1348	1348	1363	1363	1158	921
Absorber Type	--	--	--	--	--	--	--	--	--
Cluster Gap (cm)	-	-	-	-	-	-	-	1.2	1.2
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--	--
EALCF (MeV)	2.538E-7	2.586E-7	2.647E-7	2.587E-7	2.582E-7	2.600E-7	2.609E-7	2.379E-7	2.063E-7
Exp. σ	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012
k_{eff}	0.99526	0.99745	0.99759	0.99765	0.99888	0.99735	0.99758	0.99772	0.99691
σ	0.00072	0.00065	0.00068	0.00065	0.00070	0.00067	0.00071	0.00070	0.00062

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	9.01	9.02	9.03	9.04	9.05	9.06	9.07	9.08	9.09	9.10	9.11	9.12	9.13
Clusters	3	3	3	3	3	3	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%
Pitch (cm)	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540
Fuel OD (cm)	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265
Clad OD (cm)	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	259	259	259	259	259	259	259	259	259	259	259	259	259
Soluble B (ppm)	-	-	-	-	-	-	-	-	-	-	-	-	-
Absorber Type	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (1.05% B)	304L SS (1.05% B)	304L SS (1.62% B)	304L SS (1.62% B)	Boral	Cu	Cu	Cu	Cu
Cluster Gap (cm)	8.6	9.7	9.2	9.8	6.1	8.1	5.8	7.9	6.7	8.2	9.4	8.5	9.6
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00000	0.00000	0.00000	0.00000	0.00455	0.00455	0.00690	0.00690	0.06704	-	-	-	-
EALCF(MeV)	1.183E-7	1.181E-7	1.168E-7	1.179E-7	1.182E-7	1.182E-7	1.191E-7	1.182E-7	1.183E-7	1.173E-7	1.176E-7	1.169E-7	1.163E-7
Exp. σ	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021
k_{eff}	0.99548	0.99343	0.99330	0.99371	0.99593	0.99295	0.99616	0.99389	0.99571	0.99319	0.99378	0.99263	0.99566
σ	0.00191	0.00182	0.00187	0.00192	0.00174	0.00193	0.00198	0.00175	0.00209	0.00153	0.00178	0.00191	0.00177

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	9.14	9.15	9.16	9.17	9.18	9.19	9.20	9.21	9.22	9.23	9.24	9.25	9.26	9.27
Clusters	3	3	3	3	3	3	3	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%
Pitch (cm)	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540	2.540
Fuel OD (cm)	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265	1.265
Clad OD (cm)	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415	1.415
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	259	259	259	259	259	259	259	259	259	259	259	259	259	259
Soluble B (ppm)	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Absorber Type	Cu (0.989 wt % Cd)	Cu (0.989 wt % Cd)	Cd	Cd	Cd	Cd	Cd	Cd	Cd	Cd	Al (no B)	Al (no B)	Zircaloy-4	Zircaloy-4
Cluster Gap (cm)	6.7	8.4	5.9	7.4	6.0	7.4	5.9	7.4	5.7	7.3	10.7	10.8	10.9	10.9
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--	--	--	--	0.00000	0.00000	--	--
EALCF(MeV)	1.186E-7	1.171E-7	1.186E-7	1.183E-7	1.183E-7	1.168E-7	1.182E-7	1.187E-7	1.199E-7	1.173E-7	1.167E-7	1.165E-7	1.181E-7	1.177E-7
Exp. σ	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021	0.0021
k_{eff}	0.99431	0.99639	0.99686	0.99716	0.99237	0.99719	0.99434	0.99692	0.99606	0.99740	0.99281	0.99256	0.99365	0.99497
σ	0.00188	0.00207	0.00183	0.00166	0.00194	0.00187	0.00179	0.00183	0.00189	0.00206	0.00168	0.00197	0.00197	0.00193

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	11.03	11.04	11.05	11.06	11.07	11.08	11.09
Clusters	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%
Pitch (cm)	1.636	1.636	1.636	1.636	1.636	1.636	1.636
Fuel OD (cm)	1.030	1.030	1.030	1.030	1.030	1.030	1.030
Clad OD (cm)	1.206	1.206	1.206	1.206	1.206	1.206	1.206
Clad Material	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	219	219	219	219	219	219	219
Soluble B (ppm)	769	764	762	753	739	721	702
Absorber Type	--	--	--	--	--	--	--
Cluster Gap (cm)	1.6	1.6	1.6	1.6	1.6	1.6	1.6
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	--	--
EALCF [MeV]	2.027E-7	2.020E-7	2.035E-7	2.044E-7	2.065E-7	2.068E-7	2.085E-7
Exp. σ	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032
k_{eff}	0.99482	0.99494	0.99514	0.99564	0.99508	0.99526	0.99520
σ	0.00031	0.00030	0.00030	0.00030	0.00031	0.00030	0.00031

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	13.01	13.02	13.03	13.04	13.05	13.06	13.07
Clusters	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%	4.31%
Pitch (cm)	1.892	1.892	1.892	1.892	1.892	1.892	1.892
Fuel OD (cm)	1.265	1.265	1.265	1.265	1.265	1.265	1.265
Clad OD (cm)	1.415	1.415	1.415	1.415	1.415	1.415	1.415
Clad Material	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	107	107	107	107	107	107	107
Soluble B (ppm)	--	--	--	--	--	--	--
Absorber Type	304L SS (no B)	304L SS (1.05% B)	Boral B	Boroflex	Cd	Cu	Cu (0.989 wt % Cd)
Cluster Gap (cm)	13.8	9.8	8.3	8.4	8.9	13.5	10.6
Reflector	Steel	Steel	Steel	Steel	Steel	Steel	Steel
Plate Loading (g $^{10}B/cm^2$)	0.00000	0.00455	0.03022	0.02361	--	--	--
EALCF (MeV)	2.982E-7	3.068E-7	3.111E-7	3.094E-7	3.097E-7	2.998E-7	3.061E-7
Exp. σ	0.0018	0.0018	0.0018	0.0018	0.0032	0.0018	0.0018
k_{eff}	0.99402	0.99446	0.99355	0.99401	0.99281	0.99496	0.99378
σ	0.00068	0.00064	0.00064	0.00064	0.00066	0.00063	0.00062

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	14.01	14.02	14.05	14.06	14.07
Clusters	1	1	1	1	1
Enrichment (wt % ^{235}U)	4.31%	4.31%	4.31%	4.31%	4.31%
Pitch (cm)	1.890	1.890	1.890	1.715	1.715
Fuel OD (cm)	1.265	1.265	1.265	1.265	1.265
Clad OD (cm)	1.415	1.415	1.415	1.415	1.415
Clad Material	Al	Al	Al	Al	Al
H/U (fissile)	106	106	106	73	73
Soluble B (ppm)	0	491	2539	0	1030
Absorber Type	--	--	--	--	--
Cluster Gap (cm)	--	--	--	--	--
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--
EALCF (MeV)	2.873E-7	3.447E-7	6.003E-7	5.175E-7	7.722E-7
Exp. σ	0.0019	0.0077	0.0069	0.0033	0.0051
k_{eff}	0.99668	0.98595	1.00221	1.00245	0.99973
σ	0.00044	0.00045	0.00043	0.00045	0.00044

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	16.01	16.02	16.03	16.04	16.05	16.06	16.07	16.08	16.09	16.10
Clusters	3	3	3	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%
Pitch (cm)	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032
Fuel OD (cm)	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118
Clad OD (cm)	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	404	404	404	404	404	404	404	404	404	404
Soluble B (ppm)	--	--	--	--	--	--	--	--	--	--
Absorber Type	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (no B)	304L SS (1.05% B)	304L SS (1.05% B)	304L SS (1.62% B)
Cluster Gap (cm)	6.9	7.6	7.5	7.4	7.8	10.4	11.5	7.6	9.6	7.4
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00455	0.00455	0.00690
EALCF (MeV)	1.000E-7	9.983E-8	9.947E-8	1.001E-7	1.002E-7	1.009E-7	1.001E-7	9.993E-8	1.004E-7	1.012E-7
Exp. σ	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031
k_{eff}	0.99494	0.99509	0.99252	0.99562	0.99313	0.99813	0.99670	0.99383	0.99277	0.99292
σ	0.00171	0.00153	0.00157	0.00162	0.00173	0.00179	0.00175	0.00172	0.00157	0.00162

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	16.11	16.12	16.13	16.14	16.15	16.16	16.17	16.18	16.19	16.20	16.21	16.22
Clusters	3	3	3	3	3	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%
Pitch(cm)	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032
Fuel OD (cm)	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118
Clad OD (cm)	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	404	404	404	404	404	404	404	404	404	404	404	404
Soluble B (ppm)	--	--	--	--	--	--	--	--	--	--	--	--
Absorber Type	304L SS (1.62% B)	Boral	Boral	Boral	Cu	Cu	Cu	Cu	Cu	Cu (0.989 wt % Cd)	Cd	Cd
Cluster Gap (cm)	9.5	6.3	9.0	5.1	6.6	7.7	7.5	6.9	7.0	5.2	6.7	7.6
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00690	0.06704	0.06704	0.06704	--	--	--	--	--	--	--	--
EALCF (MeV)	9.962E-8	1.016E-7	1.006E-7	1.025E-7	1.000E-7	9.944E-8	9.904E-8	9.919E-8	9.971E-8	1.001E-7	1.024E-7	1.014E-7
Exp. σ	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031
k_{eff}	0.99641	0.99306	0.99650	0.99468	0.99330	0.99181	0.99392	0.99556	0.99454	0.99449	0.99130	0.99480
σ	0.00154	0.00161	0.00152	0.00162	0.00157	0.00153	0.00155	0.00172	0.00165	0.00155	0.00166	0.00157

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	16.23	16.24	16.25	16.26	16.27	16.28	16.29	16.30	16.31	16.32
Clusters	3	3	3	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%
Pitch(cm)	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032	2.032
Fuel OD (cm)	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118	1.118
Clad OD (cm)	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270	1.270
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	404	404	404	404	404	404	404	404	404	404
Soluble B (ppm)	--	--	--	--	--	--	--	--	--	--
Absorber Type	Cd	Cd	Cd	Cd	Cd	Al (no B)	Al (no B)	Al (no B)	Zircaloy-4	Zircaloy-4
Cluster Gap cm)	9.4	7.8	9.4	7.5	9.4	8.7	8.8	8.8	8.8	8.8
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--	0.00000	0.00000	0.00000	--	--
EALCF (MeV)	1.010E-7	1.018E-7	1.006E-7	1.019E-7	9.948E-8	9.991E-8	9.843E-8	9.807E-8	9.964E-8	9.834E-8
Exp. σ	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031	0.0031
k_{eff}	0.99350	0.99400	0.99628	0.99262	0.99410	0.99647	0.99360	0.99702	0.99497	0.99195
σ	0.00184	0.00152	0.00169	0.00151	0.00168	0.00166	0.00157	0.00160	0.00163	0.00172

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	35.01	35.02	40.01	40.02	40.03	40.04	40.05	40.06	40.07	40.08	40.09	40.10
Clusters	1	1	4	4	4	4	4	4	4	4	4	4
Enrichment (wt % ^{235}U)	2.60%	2.60%	4.74%	4.74%	4.74%	4.74%	4.74%	4.74%	4.74%	4.74%	4.74%	4.74%
Pitch (cm)	1.956	1.956	1.600	1.600	1.600	1.600	1.600	1.600	1.600	1.600	1.600	1.600
Fuel OD (cm)	1.250	1.250	0.790	0.790	0.790	0.790	0.790	0.790	0.790	0.790	0.790	0.790
Clad OD (cm)	1.417	1.417	0.940	0.940	0.940	0.940	0.940	0.940	0.940	0.940	0.940	0.940
Clad Material	Al	Al	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy
H/U (fissile)	203	203	231	231	231	231	231	231	231	231	231	231
Soluble B (ppm)	70	148	--	--	--	--	--	--	--	--	--	--
Absorber Type	--	--	Z2 CN18/10 SS (1.10% B)	Z2 CN18/10 SS (1.10% B)	Z2 CN18/10 SS (1.10% B)	Z2 CN18/10 SS (1.10% B)	Boral	Boral	Boral	Boral	Boral	Boral
Cluster Gap (cm)	--	--	2.3	2.3	2.3	2.3	3.3	3.3	3.3	3.3	3.3	3.3
Reflector	H_2O	H_2O	H_2O	Lead	Lead	Lead	H_2O	Lead	Lead	Lead	Steel	Steel
Plate Loading (g $^{10}B/cm^2$)	--	--	0.00252	0.00252	0.00252	0.00252	0.04608	0.04608	0.04608	0.04608	0.04608	0.04608
EALCF (MeV)	2.170E-7	2.202E-7	1.493E-7	1.717E-7	1.625E-7	1.576E-7	1.432E-7	1.515E-7	1.470E-7	1.459E-7	1.537E-7	1.469E-7
Exp. σ	0.0018	0.0019	0.0039	0.0041	0.0041	0.0041	0.0042	0.0044	0.0044	0.0044	0.0046	0.0046
k_{eff}	0.99341	0.99131	0.99586	0.99358	0.99539	0.99237	0.99144	0.99878	0.99418	0.99240	0.99693	0.99137
σ	0.00070	0.00078	0.00195	0.00192	0.00203	0.00194	0.00193	0.00196	0.00224	0.00216	0.00190	0.00208

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	42.01	42.02	42.03	42.04	42.05	42.06	42.07
Clusters	3	3	3	3	3	3	3
Enrichment (wt % ^{235}U)	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%	2.35%
Pitch (cm)	1.684	1.684	1.684	1.684	1.684	1.684	1.684
Fuel OD (cm)	1.118	1.118	1.118	1.118	1.118	1.118	1.118
Clad OD (cm)	1.270	1.270	1.270	1.270	1.270	1.270	1.270
Clad Material	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	221	221	221	221	221	221	221
Soluble B (ppm)	--	--	--	--	--	--	--
Absorber Type	304L SS (no B)	304L SS (1.05% B)	Boral B	Boroflex	Cd	Cu	Cu-Cd
Cluster Gap (cm)	8.3	4.8	2.7	3.0	3.9	7.8	5.4
Reflector	Steel	Steel	Steel	Steel	Steel	Steel	Steel
Plate Loading (g $^{10}B/cm^2$)	0.00000	0.00455	0.03022	0.02361	--	--	--
EALCF (MeV)	1.813E-7	1.824E-7	1.915E-7	1.887E-7	1.857E-7	1.786E-7	1.833E-7
Exp. σ	0.0016	0.0016	0.0016	0.0017	0.0033	0.0016	0.0018
k_{eff}	0.99250	0.99514	0.99219	0.99476	0.99469	0.99434	0.99319
σ	0.00171	0.00183	0.00169	0.00169	0.00161	0.00191	0.00157

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	50.03	50.03	50.03	50.03	50.03
Clusters	1	1	1	1	1
Enrichment (wt % ^{235}U)	4.74%	4.74%	4.74%	4.74%	4.74%
Pitch (cm)	1.300	1.300	1.300	1.300	1.300
Fuel OD (cm)	0.790	0.790	0.790	0.790	0.790
Clad OD (cm)	0.940	0.940	0.940	0.940	0.940
Clad Material	Al alloy	Al alloy	Al alloy	Al alloy	Al alloy
H/U (fissile)	124	124	124	124	124
Soluble B (ppm)	821	821	4986	4986	4986
Absorber Type	--	--	--	--	--
Cluster Gap (cm)	--	--	--	--	--
Reflector	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O
Plate Loading (g $^{10}B/cm^2$)	--	--	--	--	--
EALCF (MeV)	2.170E-7	2.083E-7	2.318E-7	2.252E-7	2.195E-7
Exp. σ	0.0010	0.0010	0.0010	0.0010	0.0010
k_{eff}	0.99330	0.99340	0.99489	0.99319	0.99306
σ	0.00080	0.00071	0.00075	0.00075	0.00080

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	51.01	51.02	51.03	51.04	51.05	51.06	51.07	51.08	51.09
Clusters	9	9	9	9	9	9	9	9	9
Enrichment (wt % ^{235}U)	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%
Pitch (cm)	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636
Fuel OD (cm)	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
Clad OD (cm)	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	219	219	219	219	219	219	219	219	219
Soluble B (ppm)	143	510	514	501	493	474	462	432	217
Absorber Type	none	SS	SS	SS	SS	SS	SS	SS	SS
Cluster Gap (cm)	4.9	1.6	1.6	1.6	1.6	1.6	1.6	1.6	3.3
Reflector	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00000	--	--	--	--	--	--	--	--
EALCF (MeV)	1.535E-7	2.045E-7	2.043E-7	2.067E-7	2.074E-7	2.083E-7	2.085E-7	2.098E-7	1.737E-7
Exp. σ	0.0020	0.0024	0.0024	0.0024	0.0024	0.0024	0.0024	0.0024	0.0019
k_{eff}	0.99133	0.99597	0.99555	0.99486	0.99504	0.99542	0.99530	0.99507	0.99368
σ	0.00033	0.00035	0.00033	0.00034	0.00034	0.00034	0.00034	0.00034	0.00033

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	51.10	51.11	51.12	51.13	51.14	51.15	51.16	51.17	51.18	51.19
Clusters	9	9	9	9	9	9	9	9	9	9
Enrichment (wt % ^{235}U)	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%	2.46%
Pitch (cm)	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636	1.636
Fuel OD (cm)	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030	1.030
Clad OD (cm)	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206	1.206
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	219	219	219	219	219	219	219	219	219	219
Soluble B (ppm)	15	28	92	395	121	487	197	634	320	72
Absorber Type	B/Al Set 5	B/Al Set 5A	B/Al Set 4	B/Al Set 3	B/Al Set 3	B/Al Set 2	B/Al Set 2	B/Al Set 1	B/Al Set 1	B/Al Set 1
Cluster Gap (cm)	1.6	1.6	1.6	1.6	3.3	1.6	3.3	1.6	3.3	4.9
Reflector	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O	Borated H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00517	0.00519	0.00403	0.00128	0.00128	0.00078	0.00078	0.00032	0.00032	0.00032
EALCF (MeV)	2.029E-7	2.015E-7	2.056E-7	2.112E-7	1.773E-7	2.106E-7	1.775E-7	2.119E-7	1.780E-7	1.587E-7
p. σ	0.0019	0.0019	0.0019	0.0022	0.0019	0.0024	0.0020	0.0027	0.0021	0.0019
k_{eff}	0.99210	0.99447	0.99073	0.98652	0.98634	0.99042	0.98974	0.99152	0.99029	0.98927
σ	0.00034	0.00034	0.00034	0.00034	0.00034	0.00034	0.00034	0.00034	0.00035	0.00035

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	65.01	65.02	65.03	65.04	65.05	65.06	65.07	65.08
Clusters	2	2	2	2	2	2	2	2
Enrichment (wt % ^{235}U)	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%
Pitch (cm)	1.956	1.956	1.956	1.956	1.956	1.956	1.956	1.956
Fuel OD (cm)	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250
Clad OD (cm)	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	203	203	203	203	203	203	203	203
Soluble B (ppm)	--	--	--	--	--	--	--	--
Absorber Type	none	304L SS (No B)	304L SS (0.67% B)	304L SS (0.98% B)	none	304L SS (No B)	304L SS (No B)	304L SS (No B)
Cluster Gap (cm)	5.9	5.9	5.9	5.9	7.8	7.8	7.8	7.8
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	--	0.00000	0.00599	0.00875	--	0.00000	0.00000	0.00000
EALCF [MeV]	2.045E-7	2.030E-7	2.054E-7	2.038E-7	2.049E-7	2.030E-7	2.055E-7	2.040E-7
Exp. σ	0.0014	0.0014	0.0015	0.0015	0.0014	0.0014	0.0014	0.0016
k_{eff}	0.99571	0.99618	0.99534	0.99547	0.99691	0.99614	0.99589	0.99624
σ	0.00023	0.00022	0.00023	0.00023	0.00023	0.00023	0.00023	0.00023

Table 6.5.4-3 LEU MCNP Validation Statistics (cont'd)

Case	65.09	65.10	65.11	65.12	65.13	65.14	65.15	65.16	65.17
Clusters	2	2	2	2	2	2	2	2	2
Enrichment (wt % ^{235}U)	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%	2.60%
Pitch (cm)	1.956	1.956	1.956	1.956	1.956	1.956	1.956	1.956	1.956
Fuel OD (cm)	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250	1.250
Clad OD (cm)	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417	1.417
Clad Material	Al	Al	Al	Al	Al	Al	Al	Al	Al
H/U (fissile)	203	203	203	203	203	203	203	203	203
Soluble B (ppm)	--	--	--	--	--	--	--	--	--
Absorber Type	304L SS (No B)	304L SS (0.67% B)	304L SS (0.67% B)	304L SS (0.67% B)	304L SS (0.67% B)	304L SS (0.98% B)	304L SS (0.98% B)	304L SS (0.98% B)	304L SS (0.98% B)
Cluster Gap (cm)	7.8	7.8	7.8	7.8	7.8	7.8	7.8	7.8	7.8
Reflector	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Plate Loading (g $^{10}B/cm^2$)	0.00000	0.00299	0.00299	0.00599	0.00599	0.00438	0.00438	0.00875	0.00875
EALCF [MeV]	1.993E-7	2.050E-7	2.069E-7	2.072E-7	1.977E-7	2.010E-7	2.004E-7	2.027E-7	2.017E-7
Exp. σ	0.0015	0.0016	0.0016	0.0017	0.0016	0.0016	0.0016	0.0017	0.0016
k_{eff}	0.99667	0.99676	0.99637	0.99643	0.99701	0.99650	0.99634	0.99658	0.99645
σ	0.00022	0.00022	0.00023	0.00023	0.00022	0.00023	0.00023	0.00022	0.00023

6.5.4.4 MOX (Plutonium Oxide/Uranium Oxide Mix) Results of Benchmark Calculations

The range of parameters included in the MOX benchmarks is shown in Table 6.5.4-4. Experiments are chosen to reflect the fuel evaluated for shipment. This includes the use of arrays of MOX rods (<10 wt % Pu) with light water moderation. Trending in k_{eff} was evaluated for the following independent variables: isotope weight percent as a function of ^{238}U fraction, moderator to fuel volume ratio, and energy of the average neutron lethargy causing fission (EALCF). No statistically significant trends were found for any of the system parameters. USLs are, therefore, generated for each of the independent variables. A minimum USL covering the range of applicability of the benchmark set is determined.

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. The linear correlation coefficient, R, is calculated by taking the square root of the R^2 value. In particular, the correlation coefficient, R, is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If R is +1, a perfect linear relationship with a positive slope is indicated. If R is -1, a perfect linear relationship with a negative slope is indicated. When R is 0, no linear relationship is indicated.

Table 6.5.4-5 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter. Linear fits and correlation constants are based on the 59 data-point evaluation sets plotted in Section 6.5.4.5.

As there is no significant correlation to any of the independent variables, the USL for each independent variable is calculated and shown with its range of applicability in Table 6.5.4-2. A sample output for EALCF is shown in Figure 6.5.4-11. Uncertainties included in the USLSTATS evaluation are the Monte Carlo uncertainty associated with the reactivity calculation and experimental uncertainty that was provided in the literature for each of the cases.

The $^{242}Pu/^{238}U$ ratio had the strongest correlation and the water-to-fuel volume ratio produced the minimum USL for all the independent variables correlated. Upper subcritical limits (USLs) are generated based on minimum margins of subcriticality (MMS), also referred to as administrative margin of 5%. The resulting minimum USLSTATS derived USL is 0.9331.

Figure 6.5.4-11 PWR MOX USLSTATS Output for Water to Fuel Volume Ratio

```
          uslstats: a utility to calculate upper subcritical
                    limits for criticality safety applications

****************************************************************************
                       Version 1.4, April 23, 2003
                       Oak Ridge National Laboratory
****************************************************************************


  Input to statistical treatment from file:w2fvr5.in

Title: keff vs Water-to-Fuel Volume Ratio

  Proportion of the population = .995
  Confidence of fit            = .950
  Confidence on proportion     = .950
  Number of observations       =   59
  Minimum value of closed band =    0.00
  Maximum value of closed band =    0.00
  Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
1.19000E+00	9.91550E-01	5.96000E-03	1.52000E+00	9.88170E-01	5.14000E-03
1.19000E+00	9.95580E-01	4.59000E-03	2.49000E+00	9.93450E-01	3.65000E-03
2.52000E+00	9.93540E-01	3.02000E-03	3.52000E+00	9.87870E-01	3.65000E-03
2.52000E+00	1.00039E+00	2.26000E-03	4.40000E+00	9.93650E-01	4.44000E-03
3.64000E+00	9.93960E-01	2.34000E-03	6.28000E+00	9.96480E-01	5.43000E-03
3.64000E+00	1.00264E+00	2.53000E-03	7.05000E+00	9.93170E-01	5.13000E-03
1.68000E+00	9.93170E-01	7.16000E-03	2.49000E+00	9.96100E-01	3.53000E-03
2.16000E+00	9.91940E-01	5.77000E-03	3.52000E+00	9.91630E-01	3.93000E-03
2.16000E+00	9.96120E-01	5.28000E-03	4.40000E+00	9.94460E-01	4.62000E-03
4.71000E+00	9.94700E-01	2.95000E-03	6.28000E+00	9.95150E-01	5.72000E-03
5.67000E+00	9.94430E-01	2.56000E-03	7.05000E+00	9.91400E-01	6.11000E-03
1.08000E+01	9.99990E-01	2.16000E-03	1.52000E+00	9.90720E-01	3.27000E-03
2.42000E+00	9.90540E-01	4.71000E-03	2.49000E+00	9.91190E-01	3.05000E-03
2.42000E+00	9.95620E-01	4.72000E-03	3.52000E+00	9.91770E-01	3.84000E-03
2.42000E+00	1.00016E+00	4.72000E-03	4.40000E+00	9.97310E-01	4.73000E-03
2.98000E+00	9.92510E-01	4.05000E-03	6.28000E+00	9.97440E-01	5.62000E-03
2.98000E+00	9.95970E-01	4.02000E-03	7.05000E+00	9.96110E-01	6.52000E-03
2.98000E+00	1.00453E+00	4.02000E-03	1.10000E+00	9.93110E-01	5.43000E-03
4.24000E+00	9.94610E-01	4.12000E-03	1.56000E+00	9.89370E-01	4.93000E-03
4.24000E+00	9.98940E-01	4.14000E-03	2.71000E+00	9.88960E-01	5.03000E-03
4.24000E+00	9.99980E-01	4.12000E-03	3.79000E+00	9.89000E-01	6.22000E-03
5.55000E+00	9.94400E-01	5.18000E-03	5.14000E+00	9.89770E-01	7.42000E-03
5.55000E+00	9.97870E-01	5.19000E-03	5.58000E+00	9.90580E-01	8.01000E-03
1.93000E+00	9.93380E-01	2.27000E-03	1.14000E+01	9.99200E-01	2.48000E-03
2.56000E+00	9.92740E-01	2.67000E-03	1.14000E+01	9.98350E-01	2.47000E-03
3.62000E+00	9.99510E-01	2.96000E-03	1.14000E+01	9.99230E-01	2.58000E-03
4.53000E+00	9.96420E-01	2.86000E-03	2.07000E+01	1.00066E+00	1.89000E-03
7.27000E+00	9.98140E-01	3.63000E-03	2.07000E+01	9.97920E-01	1.71000E-03
1.01000E+01	9.97360E-01	4.23000E-03	2.07000E+01	9.97870E-01	1.70000E-03
1.16000E+01	9.99430E-01	4.23000E-03			

```
  chi =  1.2542 (upper bound = 9.49). The data tests normal.


       Output from statistical treatment

  keff vs Water-to-Fuel Volume Ratio

   Number of data points (n)                     59
   Linear regression, k(X)                      0.9932 + ( 3.5528E-04)*X
   Confidence on fit (1-gamma) [input]           95.0%
   Confidence on proportion (alpha) [input]      95.0%
   Proportion of population falling above
   lower tolerance interval (rho) [input]        99.5%
   Minimum value of X                            1.1000E+00
   Maximum value of X                            2.0700E+01
   Average value of X                            5.3212E+00
   Average value of k                             0.99509
   Minimum value of k                             0.98787
   Variance of fit, s(k,X)^2                     1.2095E-05
```

Figure 6.5.4-11 PWR MOX USLSTATS Output for Water to Fuel Volume Ratio (cont'd)

```
Within variance, s(w)^2                          1.9626E-05
Pooled variance, s(p)^2                          3.1721E-05
Pooled std. deviation, s(p)                      5.6322E-03
C(alpha,rho)*s(p)                                2.2969E-02
student-t @ (n-2,1-gamma)                        1.67295E+00
Confidence band width, W                         1.0388E-02
Minimum margin of subcriticality, C*s(p)-W       1.2582E-02

Upper subcritical limits: (  1.1000     <= X <=   20.700    )
***** *********** *******

USL Method 1 (Confidence Band with
Administrative Margin)           USL1 = 0.9328 + ( 3.5528E-04)*X (X <     19.146    )
                                      = 0.9396                    (X >=    19.146    )

USL Method 2 (Single-Sided Uniform
Width Closed Interval Approach)  USL2 = 0.9702 + ( 3.5528E-04)*X (X <  1.91462E+01)
                                      = 0.9770                    (X >= 1.91462E+01)


USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** ********* **

   X:  1.10E+0  3.90E+0  6.70E+0  9.50E+0  1.23E+1  1.51E+1  1.79E+1  2.07E+1
      ---------------------------------------------------------------------------
USL-1:   0.9332   0.9342   0.9352   0.9362   0.9372   0.9382   0.9392   0.9396
USL-2:   0.9706   0.9716   0.9726   0.9736   0.9746   0.9756   0.9766   0.9770
      ---------------------------------------------------------------------------

  ****************************************************************************
```

Table 6.5.4-4 PWR MOX Range of Applicability for Complete Set of 59 Benchmark Experiments

Parameter	Minimum	Maximum
Energy of average neutron lethargy causing fission (eV)	8.10E-02	8.99E-01
Uranium-235/Uranium-238 Ratio	1.58E-03	1.51E+00
Plutonium-238/Uranium-238 Ratio	1.88E-06	1.54E-04
Plutonium-239/Uranium-238 Ratio	1.39E-02	7.77E-01
Plutonium-240/Uranium-238 Ratio	1.20E-03	8.48E-02
Plutonium-241/Uranium-238 Ratio	7.90E-05	7.59E-03
Plutonium-242/Uranium-238 Ratio	4.63E-06	6.38E-04
Water-to-fuel volume ratio	1.10E+00	2.07E+01

Table 6.5.4-5 PWR MOX Correlation Coefficients and USLs for Benchmark Experiments

Variable	R^2	R	Range of Applicability	USLSTATS Correlation	USL Low	USL High
Energy of average. neutron lethargy causing fission (eV)	0.0046	0.068	8.10E-02<=X<=8.99E-01	0.9372+-1.3542E-03X	0.9359	0.9370
U-235/U-238 Ratio	0.1148	0.339	1.58E-03<=X<=1.51E+00	0.9343+2.8147E-03X	0.9343	0.9385
Pu-238/U-238 Ratio	0.1030	0.321	1.88E-06<=X<=1.54E-04	0.9343+1.8044E+01X	0.9343	0.9370
Pu-239/U-238 Ratio	0.1202	0.347	1.39E-02<=X<=7.77E-01	0.9342+5.7481E-03X	0.9342	0.9386
Pu-240/U-238 Ratio	0.1409	0.375	1.20E-03<=X<=8.48E-02	0.9341+5.8088E-02X	0.9341	0.9390
Pu-241/U-238 Ratio	0.2160	0.465	7.90E-05<=X<=7.59E-03	0.9338+8.0681E-01X	0.9338	0.9399
Pu-242/U-238 Ratio	0.2440	0.494	4.63E-06<=X<=6.38E-04	0.9338+6.7569E+00X	0.9338	0.9381
Water-to-fuel volume ratio	0.1787	0.423	1.10E+00<=X<=2.07E+01	0.9328+3.5528E-04X	0.9331	0.9401

6.5.4.5 MOX (Plutonium Oxide/Uranium Oxide Mix) Criticality Benchmarks

From the International Handbook of Evaluated Criticality Safety Benchmark Experiments, 59 experiments are selected as the basis of the MCNP benchmarking. Experiments were selected for compatibility with LWR MOX rods evaluated for shipment. Of particular interest are benchmarks with rectangular arrays of MOX rods with plutonium weight percent less than 10%.

MCNP benchmark cases represent a collection of files composed of inputs directly obtained from references (with cross-section sets adjusted to those used in the cask analysis), NAC modified input files representing unique geometries based on reference input files, and input files constructed from the experimental material and geometry information. All cases were reviewed on a "preparer/checker" principle for modeling consistency with the cask models and the choice of code options. Due to large variations in the benchmark complexities, not all options employed in the cask models are reflected in each of the benchmarks (e.g., UNIVERSE structure). A review of the criticality results did not indicate any result trend due to particular modeling choices (e.g., using the UNIVERSE structure versus a single universe, or employing KSRC versus SDEF sampling).

Identifiers for the experiment, uncertainty and calculated k_{eff} and σ for each experiment are shown in Table 6.5.4-6. Stochastic Monte Carlo error is kept within ±0.2% and each output is checked to assure that the MCNP built-in statistical checks on the results are passed and that all fissile material is sampled.

Scatter plots of k_{eff} versus system parameters for 59 data point sets (see Figure 6.5.4-12 through Figure 6.5.4-19). Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (R^2) to statistically indicate any trend or lack thereof. Scatter plots are created for k_{eff} versus the following.

- Energy of average neutron lethargy causing fission
- $^{235}U/^{238}U$ ratio
- $^{238}Pu/^{238}U$ ratio
- $^{239}Pu/^{238}U$ ratio
- $^{240}Pu/^{238}U$ ratio
- $^{241}Pu/^{238}U$ ratio
- $^{242}Pu/^{238}U$ ratio
- Water-to-fuel volume ratio

Figure 6.5.4-12 Adjusted k_{eff} vs. Energy of Average Neutron Lethargy Causing Fission

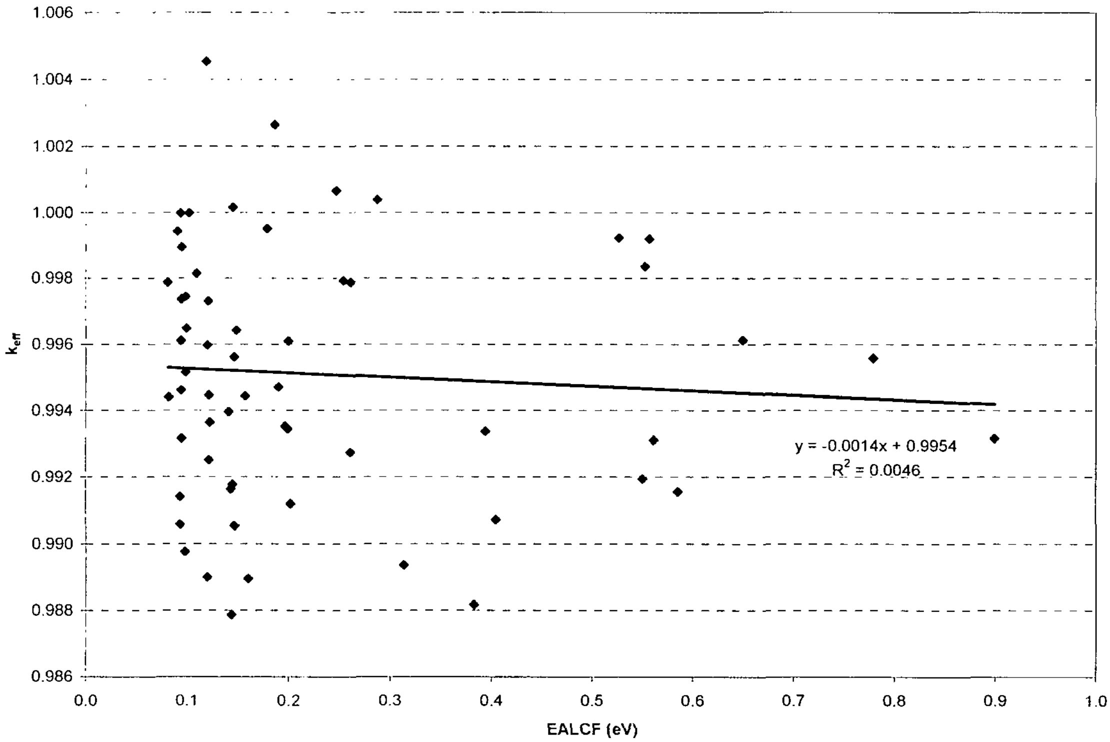


Figure 6.5.4-13 Adjusted k_{eff} vs. $^{235}U/^{238}U$ Ratio

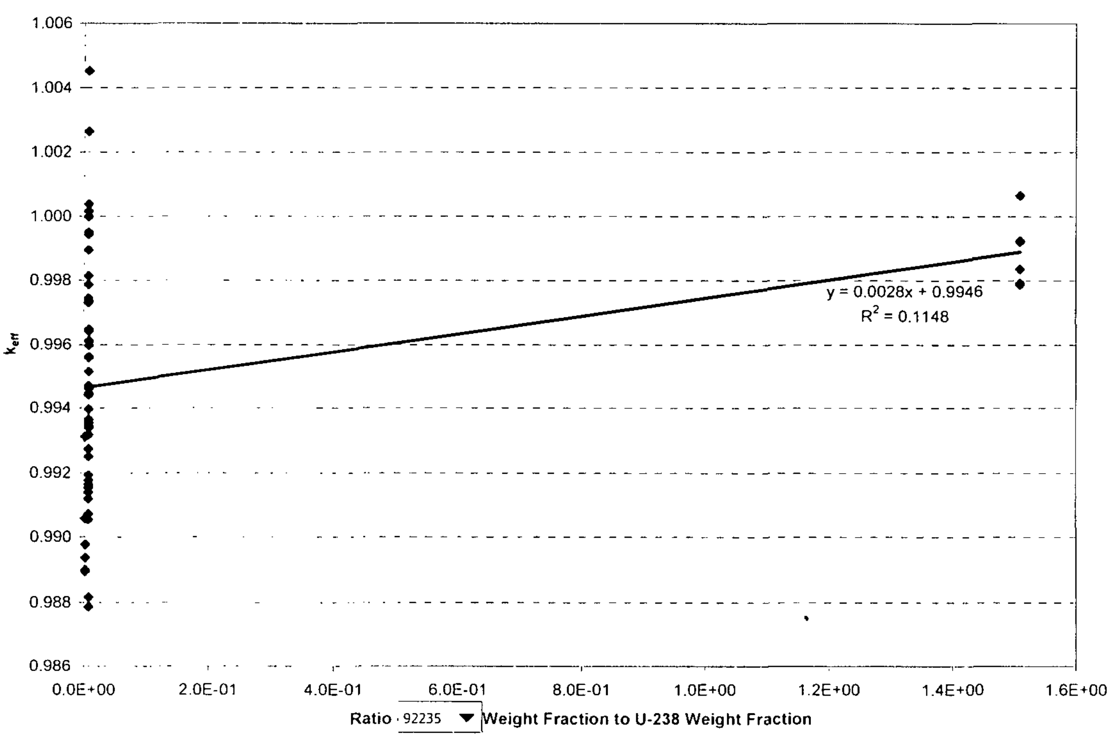


Figure 6.5.4-14 Adjusted k_{eff} vs. $^{238}Pu/^{238}U$ Ratio

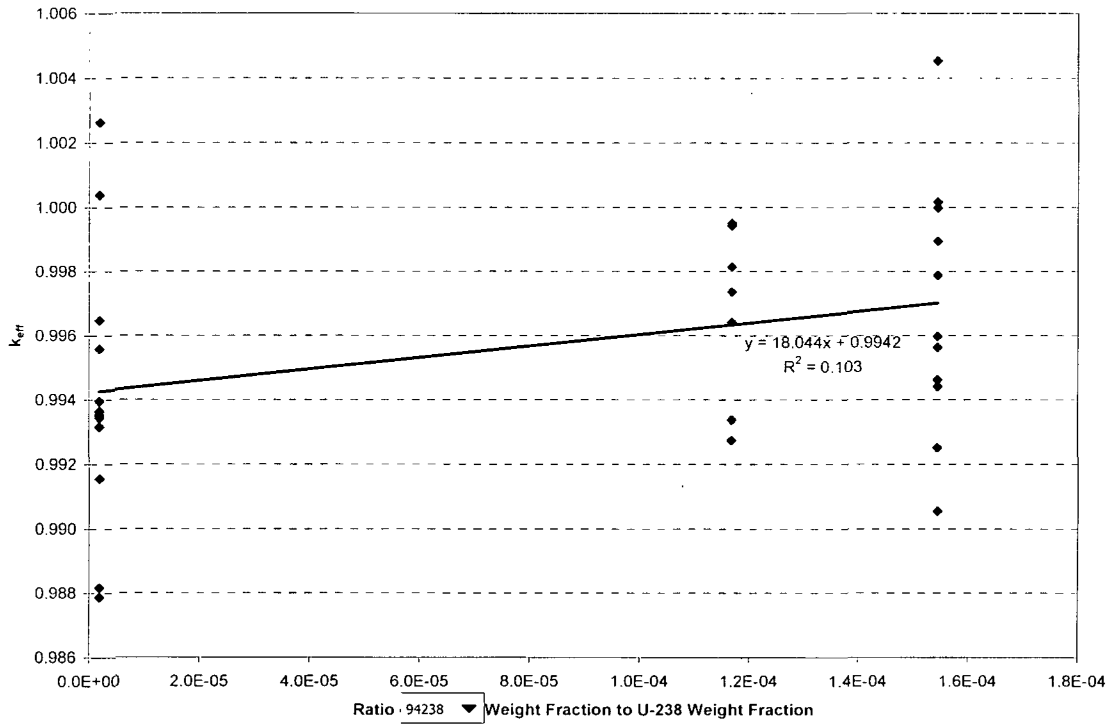


Figure 6.5.4-15 Adjusted k_{eff} vs. $^{239}Pu/^{238}U$ Ratio

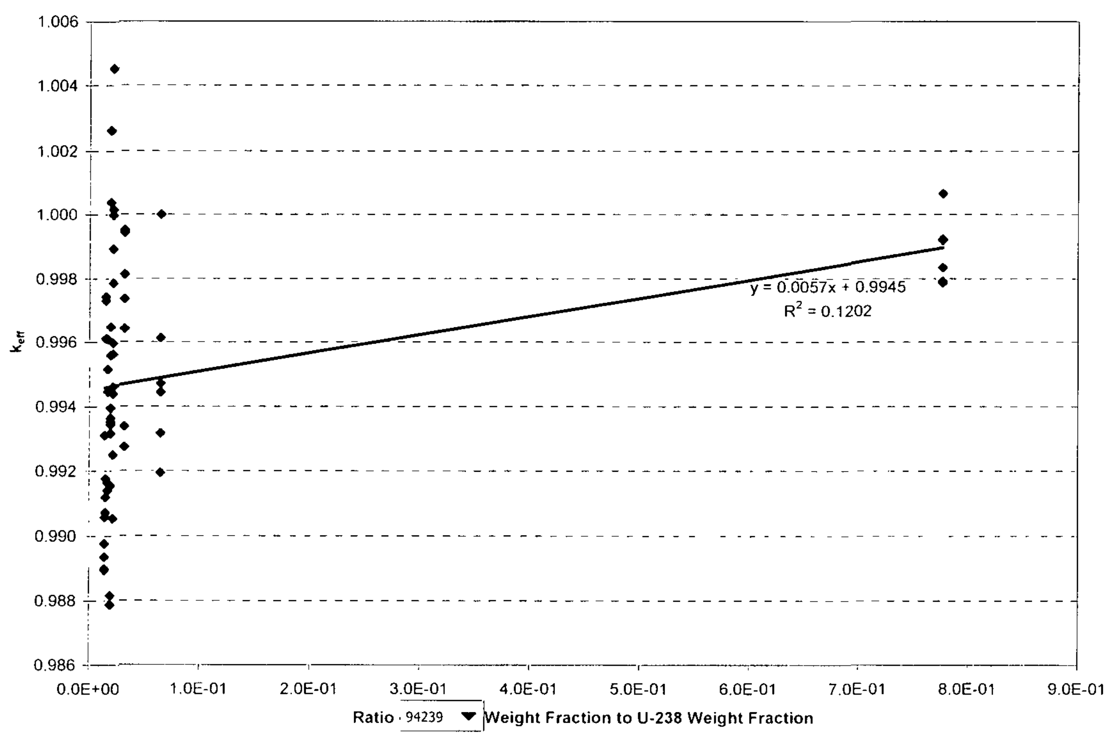


Figure 6.5.4-16 Adjusted k_{eff} vs. $^{240}Pu/^{238}U$ Ratio

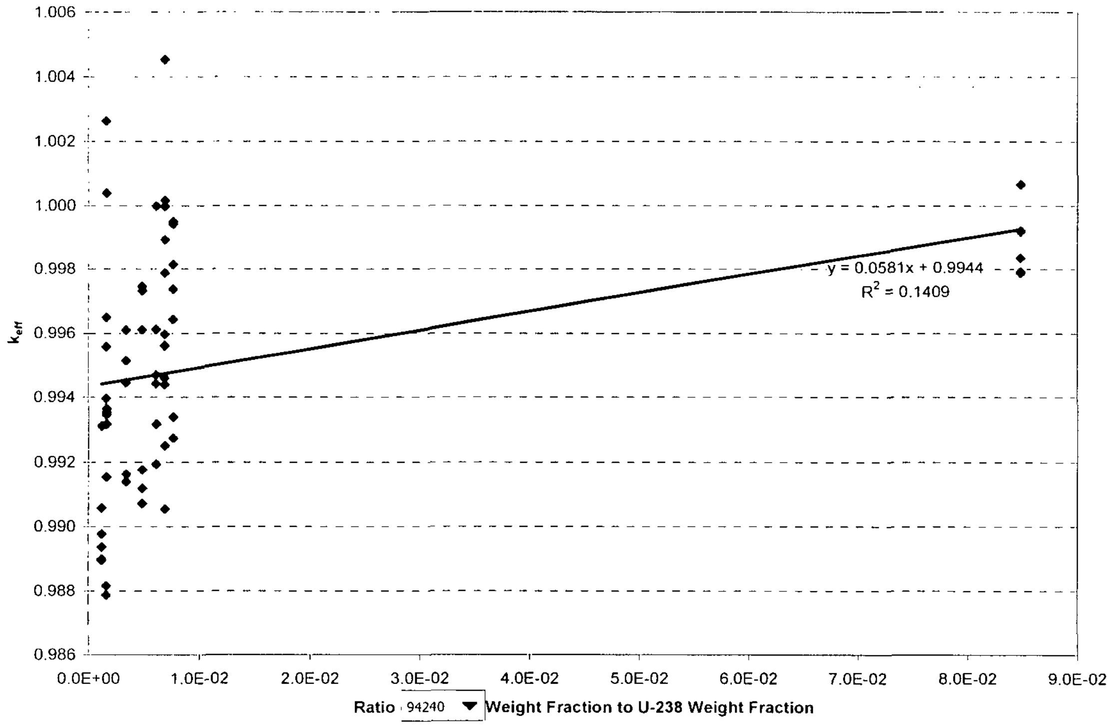


Figure 6.5.4-17 Adjusted k_{eff} vs. $^{241}Pu/^{238}U$ Ratio

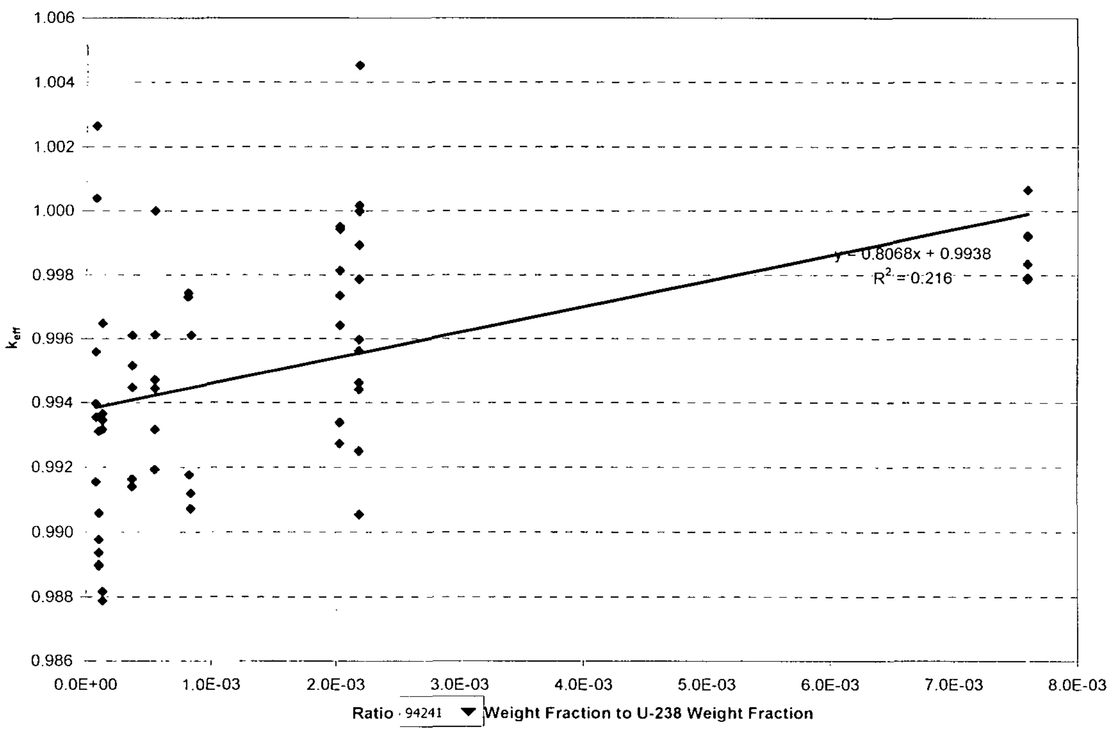


Figure 6.5.4-18 Adjusted k_{eff} vs. $^{242}Pu/^{238}U$ Ratio

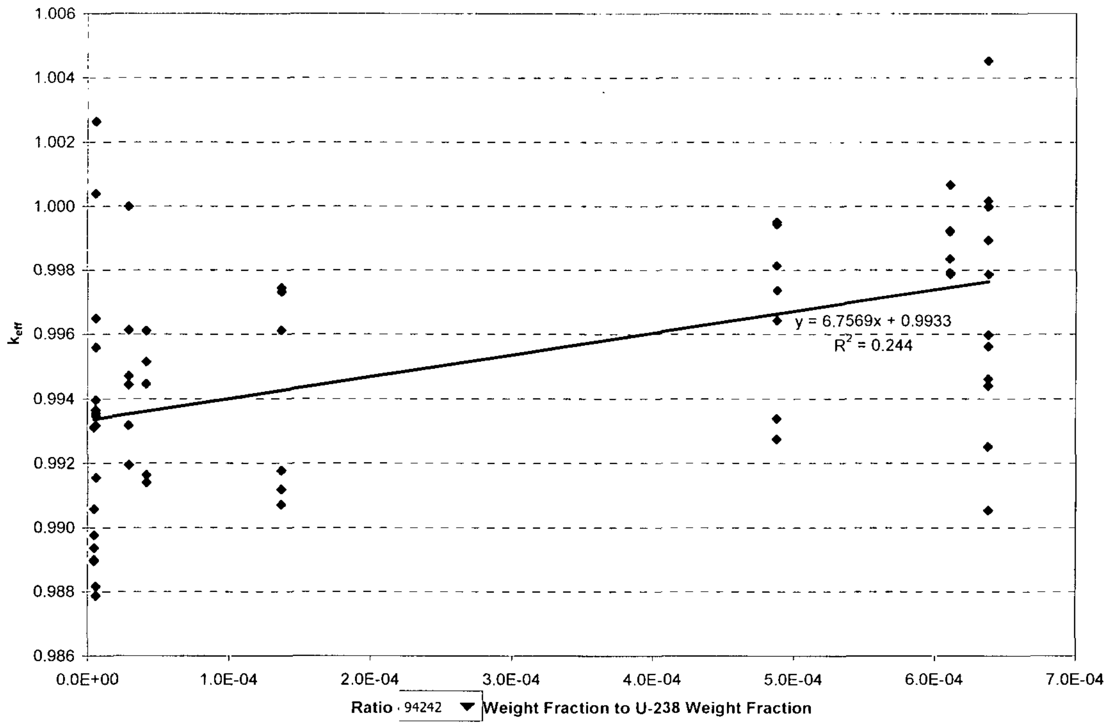


Figure 6.5.4-19 Adjusted k_{eff} vs. Water-to-Fuel Volume Ratio

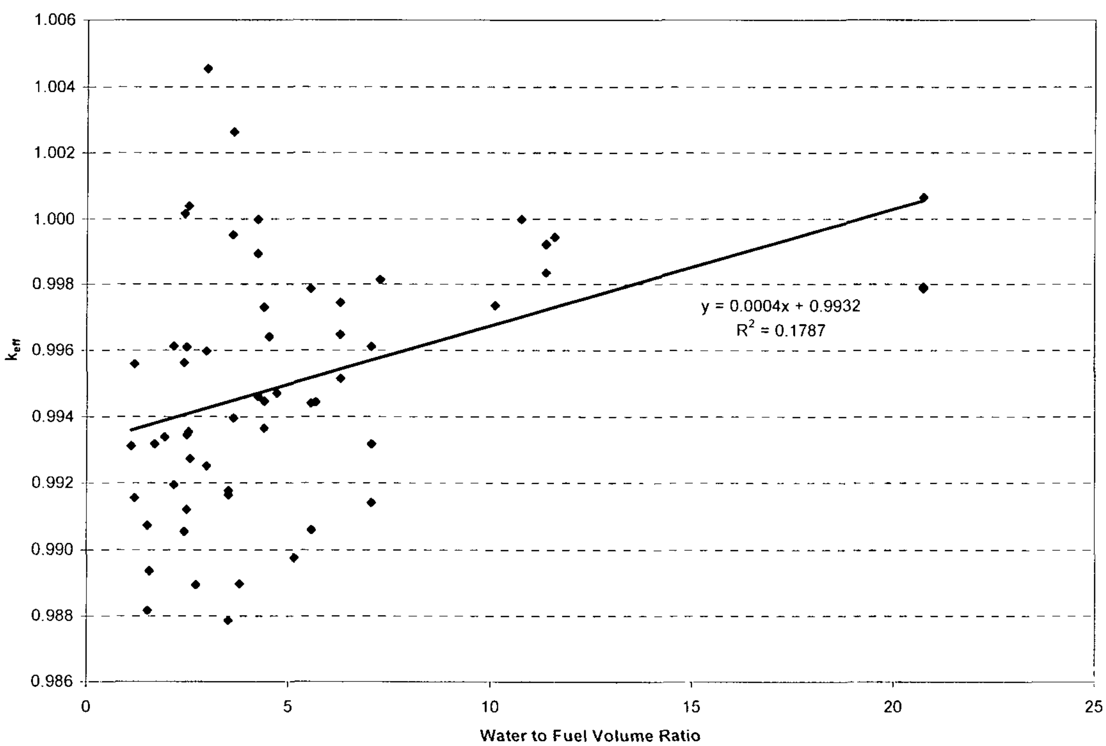


Table 6.5.4-6 MCNP Validation Statistics

		Benchmark			MCNP5 v1.30			
Identification	Water-to-Fuel Ratio	k_{eff}	σ	Δk	k_{eff}	σ	Adj. k_{eff}	EALCF (eV)
MIXCT-002-01	1.2	1.0010	0.0059	-0.0010	0.99255	0.00082	0.99155	0.58526
MIXCT-002-02	1.2	1.0009	0.0045	-0.0009	0.99648	0.00088	0.99558	0.77857
MIXCT-002-03	2.5	1.0024	0.0029	-0.0024	0.99594	0.00083	0.99354	0.19618
MIXCT-002-04	2.5	1.0024	0.0021	-0.0024	1.00279	0.00083	1.00039	0.28661
MIXCT-002-05	3.6	1.0038	0.0022	-0.0038	0.99776	0.00081	0.99396	0.14069
MIXCT-002-06	3.6	1.0029	0.0024	-0.0029	1.00554	0.00080	1.00264	0.18588
MIXCT-003-01	1.7	1.0000	0.0071	0.0000	0.99317	0.00092	0.99317	0.89937
MIXCT-003-02	2.2	1.0000	0.0057	0.0000	0.99194	0.00090	0.99194	0.54987
MIXCT-003-03	2.2	1.0000	0.0052	0.0000	0.99612	0.00093	0.99612	0.64918
MIXCT-003-04	4.7	1.0000	0.0028	0.0000	0.99470	0.00092	0.99470	0.18983
MIXCT-003-05	5.7	1.0000	0.0024	0.0000	0.99443	0.00088	0.99443	0.15707
MIXCT-003-06	10.8	1.0000	0.0020	0.0000	0.99999	0.00082	0.99999	0.10167
MIXCT-004-01	2.4	1.0000	0.0046	0.0000	0.99054	0.00099	0.99054	0.14647
MIXCT-004-02	2.4	1.0000	0.0046	0.0000	0.99562	0.00106	0.99562	0.14615
MIXCT-004-03	2.4	1.0000	0.0046	0.0000	1.00016	0.00107	1.00016	0.14470
MIXCT-004-04	3.0	1.0000	0.0039	0.0000	0.99251	0.00111	0.99251	0.12146
MIXCT-004-05	3.0	1.0000	0.0039	0.0000	0.99597	0.00098	0.99597	0.11971
MIXCT-004-06	3.0	1.0000	0.0039	0.0000	1.00453	0.00097	1.00453	0.11898
MIXCT-004-07	4.2	1.0000	0.0040	0.0000	0.99461	0.00100	0.99461	0.09396
MIXCT-004-08	4.2	1.0000	0.0040	0.0000	0.99894	0.00106	0.99894	0.09454
MIXCT-004-09	4.2	1.0000	0.0040	0.0000	0.99998	0.00097	0.99998	0.09350
MIXCT-004-10	5.6	1.0000	0.0051	0.0000	0.99440	0.00092	0.99440	0.08152
MIXCT-004-11	5.6	1.0000	0.0051	0.0000	0.99787	0.00094	0.99787	0.08098
MIXCT-005-01	1.9	1.0008	0.0022	-0.0008	0.99418	0.00057	0.99338	0.39380
MIXCT-005-02	2.6	1.0011	0.0026	-0.0011	0.99384	0.00059	0.99274	0.26049
MIXCT-005-03	3.6	1.0016	0.0029	-0.0016	1.00111	0.00058	0.99951	0.17856
MIXCT-005-04	4.5	1.0021	0.0028	-0.0021	0.99852	0.00057	0.99642	0.14820
MIXCT-005-05	7.3	1.0026	0.0036	-0.0026	1.00074	0.00049	0.99814	0.10925
MIXCT-005-06	10.1	1.0033	0.0042	-0.0033	1.00066	0.00048	0.99736	0.09455
MIXCT-005-07	11.6	1.0035	0.0042	-0.0035	1.00293	0.00046	0.99943	0.09019

Table 6.5.4-6 MCNP Validation Statistics (cont'd)

		Benchmark			MCNP5 v1.30			
Identification	Water-to-Fuel Ratio	k_{eff}	σ	Δk	k_{eff}	σ	Adj. k_{eff}	EALCF (eV)
MIXCT-006-01	1.5	1.0016	0.0051	-0.0016	0.98977	0.00060	0.98817	0.38252
MIXCT-006-02	2.5	1.0017	0.0036	-0.0017	0.99515	0.00061	0.99345	0.19894
MIXCT-006-03	3.5	1.0026	0.0036	-0.0026	0.99047	0.00059	0.98787	0.14393
MIXCT-006-04	4.4	1.0051	0.0044	-0.0051	0.99875	0.00058	0.99365	0.12231
MIXCT-006-05	6.3	1.0040	0.0054	-0.0040	1.00048	0.00055	0.99648	0.09904
MIXCT-006-06	7.1	1.0055	0.0051	-0.0055	0.99867	0.00052	0.99317	0.09438
MIXCT-007-01	2.5	1.0023	0.0035	-0.0023	0.99840	0.00049	0.99610	0.19918
MIXCT-007-02	3.5	1.0024	0.0039	-0.0024	0.99403	0.00050	0.99163	0.14313
MIXCT-007-03	4.4	1.0036	0.0046	-0.0036	0.99806	0.00048	0.99446	0.12122
MIXCT-007-04	6.3	1.0037	0.0057	-0.0037	0.99885	0.00043	0.99515	0.09818
MIXCT-007-05	7.1	1.0044	0.0061	-0.0044	0.99580	0.00042	0.99140	0.09330
MIXCT-008-01	1.5	0.9997	0.0032	0.0003	0.99042	0.00066	0.99072	0.40443
MIXCT-008-02	2.5	1.0008	0.0030	-0.0008	0.99199	0.00057	0.99119	0.20199
MIXCT-008-03	3.5	1.0023	0.0038	-0.0023	0.99407	0.00056	0.99177	0.14500
MIXCT-008-04	4.4	1.0015	0.0047	-0.0015	0.99881	0.00057	0.99731	0.12108
MIXCT-008-05	6.3	1.0022	0.0056	-0.0022	0.99964	0.00050	0.99744	0.09872
MIXCT-008-06	7.1	1.0028	0.0065	-0.0028	0.99891	0.00049	0.99611	0.09362
MIXCT-009-01	1.1	1.0003	0.0054	-0.0003	0.99341	0.00059	0.99311	0.56093
MIXCT-009-02	1.6	1.0020	0.0049	-0.0020	0.99137	0.00058	0.98937	0.31308
MIXCT-009-03	2.7	1.0035	0.0050	-0.0035	0.99246	0.00056	0.98896	0.16022
MIXCT-009-04	3.8	1.0046	0.0062	-0.0046	0.99360	0.00054	0.98900	0.11987
MIXCT-009-05	5.1	1.0059	0.0074	-0.0059	0.99567	0.00049	0.98977	0.09776
MIXCT-009-06	5.6	1.0067	0.0080	-0.0067	0.99728	0.00047	0.99058	0.09313
MIXCT-011-01	11.4	1.0000	0.0024	0.0000	0.99920	0.00063	0.99920	0.55670
MIXCT-011-02	11.4	1.0000	0.0024	0.0000	0.99835	0.00060	0.99835	0.55254
MIXCT-011-03	11.4	1.0000	0.0025	0.0000	0.99923	0.00062	0.99923	0.52644
MIXCT-011-04	20.7	1.0000	0.0018	0.0000	1.00066	0.00056	1.00066	0.24638
MIXCT-011-05	20.7	1.0000	0.0016	0.0000	0.99792	0.00059	0.99792	0.25390
MIXCT-011-06	20.7	1.0000	0.0016	0.0000	0.99787	0.00058	0.99787	0.26082

6.5.5 MCNP Criticality Benchmarks for Research Reactor Fuels

The results of the criticality analyses presented in this chapter must be compared to the upper subcritical limit (USL). The USL accounts for bias and uncertainty resulting from the method using information obtained from the analysis of criticality benchmark experimental data. Code bias calculated in this section is applicable to research reactor fuel (e.g., MTR, DIDO, and SLOWPOKE) with the exception of TRIGA (U-ZrH) elements.

Criticality code validation is performed for the Monte Carlo evaluation code and neutron cross-section libraries. Criticality validation is required by the criticality safety standard ANSI/ANS-8.1.

6.5.5.1 Benchmark Experiments and Applicability Discussion

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," provides a guide to LWR criticality benchmark calculations and the determination of bias and subcritical limits in criticality safety evaluations. In Section 2 of the NUREG, a series of LWR criticality experiments is described in sufficient detail for independent modeling. In Section 3, the criticality experiments are modeled, and the results (k_{eff} values) are presented. The method utilized in the NUREG is KENO-Va with the 44-group ENDF/B-V cross-section library embedded in SCALE 4.3. In Section 4, a guide for the determination of bias and subcritical safety limits is provided based on ANSI/ANS-8.1 and statistical analysis of the trending in the bias. Finally, guidelines for experiment selection and applicability are presented in Section 5. The approach outlined in Section 4 of the NUREG is described in detail herein and is implemented for MCNP5 with continuous energy ENDF/B-VI cross-sections.

NUREG/CR-6361 implements ANSI/ANS-8.1 criticality safety criterion as follows.

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad \text{(Equation 1)}$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from a calculation of benchmark criticality experiments using a particular calculation method. If the calculated k_{eff} values for the criticality experiments exhibit a trend with an independent parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

Δk_s = allowance for the following:

- statistical or convergence uncertainties, or both, in computation of k_s
- material and fabrication tolerances
- geometric or material representations used in computational method

Δk_c = margin for uncertainty in k_c, which includes allowance for the following:

- uncertainties in criticality experiments
- statistical or convergence uncertainties, or both, in computation of k_c
- uncertainties resulting from extrapolation of k_c outside range of experimental data
- uncertainties resulting from limitations in geometrical or material representations used in the computational method

Δk_m = arbitrary administrative margin to ensure subcriticality of k_s

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as shown.

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1 - k_c) - \Delta k_c \quad \text{(Equation 2)}$$

Noting that the definition of the bias is $\beta = 1 - k_c$, Equation 2 can be written as shown.

$$k_s + \Delta k_s \leq 1 - \Delta km - \beta - \Delta\beta \quad \text{(Equation 3)}$$

where:

$$\Delta\beta = \Delta k_c$$

Thus, the maximum allowable value for k_{eff} plus uncertainties in the system being analyzed must be below 1 minus an administrative margin (typically 0.05), which includes the bias and the uncertainty in the bias. This can also be written as shown.

$$k_s + \Delta k_s \leq \text{Upper Subcritical Limit (USL)} \quad \text{(Equation 4)}$$

where:

$$USL \equiv 1 - \Delta k_m - \beta - \Delta\beta \quad \text{(Equation 5)}$$

This is the USL criterion as described in Section 4 of NUREG/CR-6361. Two methods are prescribed for the statistical determination of the USL. The "Confidence Band with Administrative Margin (USL-1)" approach is implemented here and is referred to generically as USL. A $\Delta k_m = 0.05$ and a lower confidence band are specified based on a linear regression of k_{eff} as a function of some system parameter.

Subsequent sections contain the the list of critical benchmarks employed in the validation of MCNP with its continuous energy neutron cross-section libraries and the processing of the experimental results into the USL. Also included are linear fits of reactivity (k_{eff}) to each of the

system parameters evaluated. Experiments were chosen to reflect the in-cask fuel geometry and materials as closely as available.

6.5.5.2 HEU and IEU Criticality Benchmarks

From the International Handbook of Evaluated Criticality Safety Benchmark Experiments intermediate and high enriched thermal neutron experiments are selected as the basis of the MCNP benchmarking. Materials selected were compounds and metallic fuels. Experiments were selected for compatibility of materials and geometry with the spent fuel casks. As such experiments containing significant neutron absorber in the form of plates, rods, or soluble poison are eliminated. Also removed are experiments containing non light water moderator. Further review eliminated experiments if they did not contain a primarily thermal neutron spectrum causing fission.

All cases were reviewed on a "preparer/checker" principle for modeling consistency with the cask models and the choice of code options. Case inputs from the handbook where modified as necessary to match experimental data or to correct modeling errors. Due to large variations in the benchmark complexities, not all options employed in the cask models are reflected in each of the benchmarks (e.g., UNIVERSE structure). A review of the criticality results did not indicate any result trend due to particular modeling choices (e.g., using the UNIVERSE structure versus a single universe, or employing KSRC versus SDEF sampling).

Key system parameters for the experiments are listed in Table 6.5.5-1. Table 6.5.5-2 lists the benchmark k_{eff} and experimental uncertainty. The NAC calculated MCNP k_{eff}'s and Monte Carlo uncertainties are shown in Table 6.5.5-3. Stochastic Monte Carlo error is kept within ±0.2% and each output is checked to assure that the MCNP built-in statistical checks on the results are passed and that all fissile material is sampled. Also included in the table are the combined Monte Carlo and experimental uncertainty and the results of the initial processing of the data indicating the minimum and average bias of each experiment.

Scatter plots of k_{eff} versus enrichment and average lethargy of neutrons causing fission are shown in Figure 6.5.5-1 and Figure 6.5.5-2. Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (R^2) to statistically indicate any trend or lack thereof.

Table 6.5.5-1 MCNP Benchmark Configurations for Research Reactor Fuel Benchmarks

Identification	%U235	Clusters	Config.	Lattice	Fuel Elements	Fuel Shape	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Fuel Mat'l	Clad Mat'l
HCT-003-01	79.47	1	--	hex	1409	cross	1.22/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-02	79.47	1	--	hex	1243	cross	1.22/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-03	79.47	1	--	hex	1018	cross	1.22/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-04	79.47	1	--	hex	776	cross	1.22/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-05	79.47	1	--	hex	596	cross	1.22/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-06	79.47	1	--	hex	1266	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-07	79.47	1	--	hex	1043	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-08	79.47	1	--	hex	1006	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-09	79.47	1	--	hex	764	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-10	79.47	1	--	hex	589	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-11	79.47	1	--	hex	500	cross	0.61/1.22	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-12	79.47	1	--	hex	1106	cross	1.83/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-13	79.47	1	--	hex	727	cross	1.83/0.61	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-14	79.47	1	--	hex	949	cross	0.61/1.82	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-003-15	79.47	1	--	hex	662	cross	0.61/1.83	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-006-01	79.47	1	--	hex	1819	cross	0.56	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-006-02	79.47	1	--	hex	457	cross	1	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS
HCT-006-03	79.47	1	--	hex	554	cross	2.113	0.475 side to side	varies, 0.2 on sides	UO_2+Cu	SS

Note: Some wt%U235 values were converted from atom fractions

Table 6.5.5-1 MCNP Benchmark Configurations for Research Reactor Fuel Benchmarks (cont'd)

Identification	%U235	Clusters	Config.	Lattice	Fuel Elements	Fuel Shape	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Fuel Mat'l	Clad Mat'l
HCT-011-01	79.19	1	21x21	square	1764	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-011-02	79.19	1	21x21	square	1764	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-011-03	79.19	1	21x21	square	1764	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-012-01	79.24	4	18x18	square	1296	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-012-02	79.24	4	18x18	square	1296	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-013-01	79.19	9	14x14	square	1764	cylinder	1.4	0.7	1 (0.05thick)	UO_2+Al	SS
HCT-013-02	79.19	9	14x14	square	1764	cylinder	1.4	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-014-01	79.19	9	10x10	square	900	cylinder	$1.4\sqrt{2}$	0.7	1 (0.05 thick)	UO_2+Al	SS
HCT-014-02	79.19	9	10x10	square	900	cylinder	$1.4\sqrt{2}$	0.7	1 (0.05thick)	UO_2+Al	SS
HCT-022-01	93.2	13	4/5/4	square	494	plate	0.40132	.0508 thick	0.0762 thick	UO_2+SS	SS
HCT-022-02	93.2	14	5/5/4	square	532	plate	0.40132	.0508 thick	0.0762 thick	UO_2+SS	SS
HCT-022-03	93.2	15	5x3	square	570	plate	0.40132	.0508 thick	0.0762 thick	UO_2+SS	SS
HCT-022-04	93.2	15	5x3	square	570	plate	0.40132	.0508 thick	0.0762 thick	UO_2+SS	SS
HCT-022-05	93.2	24	4x3&4x3	square	912	plate	0.40132	.0508 thick	0.0762 thick	UO_2+SS	SS

Note: Some wt%U235 values were converted from atom fractions

Table 6.5.5-1 MCNP Benchmark Configurations for Research Reactor Fuel Benchmarks (cont'd)

Identification	%U235	Clusters	Config.	Lattice	Fuel Elements	Fuel Shape	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Fuel Mat'l	Clad Mat'l
HMT-006-01	93.17	1	4x4	square	194	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-02	93.17	1	4x4	square	188	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-03	93.17	1	4x4	square	221	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-04	93.17	1	4x4	square	255	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-05	93.17	1	4x4	square	520	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-06	93.17	1	4x4	square	286	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-07	93.17	1	5x5	square	233	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-08	93.17	1	6x6	square	232	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-09	93.17	1	7x7	square	296	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-10	93.17	1	4x4	square	365	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-11	93.17	1	3x4	square	1487	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-12	93.17	1	4x4	square	829	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-14	93.17	1	16x3	square	420	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-15	93.17	1	16x4	square	452	plate	0.316411	.0508 thick	0.1524	U+Al	AL
HMT-006-16	93.17	2	16x4	square	488	plate	0.316411	.0508 thick	0.1524	U+Al	AL

Note: Some wt%U235 values were converted from atom fractions

Table 6.5.5-1 MCNP Benchmark Configurations for Research Reactor Fuel Benchmarks (cont'd)

Identification	%U235	Clusters	Config.	Lattice	Fuel Elements	Fuel Shape	Pitch (cm)	Fuel OD (cm)	Clad OD (cm)	Fuel Mat'l	Clad Mat'l
ICT-002-01	17	1	--	hex	34	tube	6.8	4.18	4.12	UO_2	SS
ICT-002-02	17	1	--	hex	34	tube	6.8	4.18	4.12	UO_2	SS
ICT-002-03	17	1	--	hex	74	tube	6.8	4.18	4.12	UO_2	SS
ICT-002-04	17	1	--	hex	74	tube	6.8	4.18	4.12	UO_2	SS
ICT-002-05	17	1	--	hex	68	tube	6.8	4.18	4.12	UO_2	SS
ICT-002-06	17	1	--	hex	68	tube	6.8	4.18	4.12	UO_2	SS
ICT-014	19.77	1	2/1-c-1-c-1/5/1-c-1-c-1/3	square	360	plate	3.21	0.05066	0.04917	U_3Si_2	Al

Note: Some wt%U235 values were converted from atom fractions

Table 6.5.5-2 Research Reactor Fuel Benchmark K_{eff}'s and Uncertainties

Identification	k_{eff}	σ	Identification	k_{eff}	σ
HCT-003-01	1.0000	0.0044	HCT-022-01	1.0000	0.0081
HCT-003-02	1.0000	0.0044	HCT-022-02	1.0000	0.0081
HCT-003-03	1.0000	0.0044	HCT-022-03	1.0000	0.0081
HCT-003-04	1.0000	0.0044	HCT-022-04	1.0000	0.0081
HCT-003-05	1.0000	0.0044	HCT-022-05	1.0000	0.0081
HCT-003-06	1.0000	0.0044	HMT-006-01	1.0000	0.0044
HCT-003-07	1.0000	0.0044	HMT-006-02	1.0000	0.0040
HCT-003-08	1.0000	0.0044	HMT-006-03	1.0000	0.0040
HCT-003-09	1.0000	0.0044	HMT-006-04	1.0000	0.0040
HCT-003-10	1.0000	0.0044	HMT-006-05	1.0000	0.0040
HCT-003-11	1.0000	0.0044	HMT-006-06	1.0000	0.0040
HCT-003-12	1.0000	0.0044	HMT-006-07	1.0000	0.0040
HCT-003-13	1.0000	0.0044	HMT-006-08	1.0000	0.0040
HCT-003-14	1.0000	0.0044	HMT-006-09	1.0000	0.0040
HCT-003-15	1.0000	0.0044	HMT-006-10	1.0000	0.0040
HCT-006-01	1.0000	0.0058	HMT-006-11	1.0000	0.0040
HCT-006-02	1.0000	0.0020	HMT-006-12	1.0000	0.0040
HCT-006-03	1.0000	0.0048	HMT-006-14	1.0000	0.0040
HCT-011-01	0.9988	0.0042	HMT-006-15	1.0000	0.0040
HCT-011-02	0.9988	0.0042	HMT-006-16	1.0000	0.0040
HCT-011-03	0.9988	0.0042	ICT-002-01	1.0014	0.0039
HCT-012-01	0.9987	0.0032	ICT-002-02	1.0019	0.0040
HCT-012-02	0.9987	0.0034	ICT-002-03	1.0017	0.0044
HCT-013-01	0.9988	0.0042	ICT-002-04	1.0019	0.0044
HCT-013-02	0.9988	0.0043	ICT-002-05	1.0014	0.0043
HCT-014-01	0.9986	0.0048	ICT-002-06	1.0016	0.0044
HCT-014-02	0.9986	0.0049	ICT-014	1.0016	0.0014

Table 6.5.5-3 MCNP Criticality Results Research Reactor Fuel Benchmarks

	MCNP5 v1.30					Bias		
Identification	k_{eff}	σ	Adj. k_{eff}	EALCF (eV)	Total σ	Δk		
HCT-003-01	0.98696	0.00151	0.98696	0.38377	0.0047	-0.01304		
HCT-003-02	0.98944	0.00159	0.98944	0.28104	0.0047	-0.01056		
HCT-003-03	0.99199	0.00153	0.99199	0.18151	0.0047	-0.00801		
HCT-003-04	0.99330	0.00141	0.99330	0.12114	0.0046	-0.00670		
HCT-003-05	0.99699	0.00142	0.99699	0.08968	0.0046	-0.00301		
HCT-003-06	0.99991	0.00149	0.99991	0.29388	0.0046	-0.00009		
HCT-003-07	1.00396	0.00153	1.00396	0.21027	0.0047	0.00396		
HCT-003-08	1.00527	0.00140	1.00527	0.19407	0.0046	0.00527		
HCT-003-09	1.00402	0.00136	1.00402	0.13126	0.0046	0.00402		
HCT-003-10	1.00843	0.00152	1.00843	0.09467	0.0047	0.00843		
HCT-003-11	1.00758	0.00145	1.00758	0.07995	0.0046	0.00758		
HCT-003-12	0.98542	0.00141	0.98542	0.11823	0.0046	-0.01458		
HCT-003-13	0.99411	0.00133	0.99411	0.06127	0.0046	-0.00589		
HCT-003-14	0.99881	0.00147	0.99881	0.16116	0.0046	-0.00119	average	**-0.00217**
HCT-003-15	1.00119	0.00118	1.00119	0.07980	0.0046	0.00119	min	-0.01458
HCT-006-01	0.98580	0.00136	0.98580	1.11980	0.0060	-0.01420		
HCT-006-02	1.00162	0.00124	1.00162	0.10599	0.0024	0.00162	average	**-0.00625**
HCT-006-03	0.99382	0.00095	0.99382	0.05029	0.0049	-0.00618	min	-0.0142
HCT-011-01	0.98261	0.00088	0.98381	0.71500	0.0043	-0.01619		
HCT-011-02	0.98564	0.00095	0.98684	0.55072	0.0043	-0.01316	average	**-0.01408**
HCT-011-03	0.98592	0.00085	0.98712	0.43403	0.0043	-0.01288	min	-0.01619
HCT-012-01	0.98245	0.00094	0.98375	0.60393	0.0033	-0.01625	average	**-0.01535**
HCT-012-02	0.98426	0.00080	0.98556	0.45884	0.0035	-0.01444	min	-0.01625
HCT-013-01	0.98665	0.00092	0.98785	0.45717	0.0043	-0.01215	average	**-0.01189**
HCT-013-02	0.98717	0.00090	0.98837	0.31622	0.0044	-0.01163	min	-0.01215
HCT-014-01	0.99374	0.00086	0.99514	0.11840	0.0049	-0.00486	average	**-0.00486**
HCT-014-02	0.99374	0.00081	0.99514	0.09841	0.0050	-0.00486	min	-0.00486
HCT-022-01	0.98924	0.00030	0.98924	0.09658	0.0081	-0.01076		
HCT-022-02	0.98970	0.00031	0.98970	0.09820	0.0081	-0.01030		
HCT-022-03	0.98878	0.00031	0.98878	0.09950	0.0081	-0.01122		
HCT-022-04	0.98917	0.00031	0.98917	0.10060	0.0081	-0.01083	average	**-0.01104**
HCT-022-05	0.98789	0.00030	0.98789	0.09664	0.0081	-0.01211	min	-0.01211

Table 6.5.5-3 MCNP Criticality Results Research Reactor Fuel Benchmarks (cont'd)

	MCNP5 v1.30					Bias		
Identification	k_{eff}	σ	Adj. k_{eff}	EALCF (eV)	Total σ	Δk		
HMT-006-01	0.99188	0.00087	0.99188	0.08574	0.0045	-0.00812		
HMT-006-02	0.99219	0.00087	0.99219	0.07130	0.0041	-0.00781		
HMT-006-03	0.99800	0.00081	0.99800	0.06400	0.0041	-0.00200		
HMT-006-04	0.99212	0.00082	0.99212	0.06253	0.0041	-0.00788		
HMT-006-05	0.99021	0.00077	0.99021	0.05917	0.0041	-0.00979		
HMT-006-06	0.98823	0.00079	0.98823	0.05672	0.0041	-0.01177		
HMT-006-07	0.98709	0.00072	0.98709	0.05502	0.0041	-0.01291		
HMT-006-08	0.98534	0.00068	0.98534	0.05301	0.0041	-0.01466		
HMT-006-09	0.98793	0.00068	0.98793	0.05271	0.0041	-0.01207		
HMT-006-10	0.99813	0.00088	0.99813	0.08343	0.0041	-0.00187		
HMT-006-11	0.99327	0.00078	0.99327	0.06311	0.0041	-0.00673		
HMT-006-12	0.99536	0.00069	0.99536	0.05494	0.0041	-0.00464		
HMT-006-14	0.98707	0.00077	0.98707	0.05788	0.0041	-0.01293		
HMT-006-15	0.98398	0.00077	0.98398	0.05720	0.0041	-0.01602	average	**-0.00912**
HMT-006-16	0.99243	0.00078	0.99243	0.06375	0.0041	-0.00757	min	-0.01602
ICT-002-01	0.99359	0.00070	0.99219	0.09130	0.0040	-0.00781		
ICT-002-02	0.99748	0.00077	0.99558	0.13470	0.0041	-0.00442		
ICT-002-03	0.99883	0.00074	0.99713	0.10068	0.0045	-0.00287		
ICT-002-04	0.99865	0.00074	0.99675	0.12575	0.0045	-0.00325		
ICT-002-05	0.99351	0.00072	0.99211	0.10065	0.0044	-0.00789	average	**-0.00596**
ICT-002-06	0.99208	0.00076	0.99048	0.12574	0.0045	-0.00952	min	-0.00952
ICT-014	0.99791	0.00009	0.99631	0.08780	0.0014	-0.00369		

Figure 6.5.5-1 k_{eff} versus Fuel Enrichment (MCNP – Research Reactor Fuel)

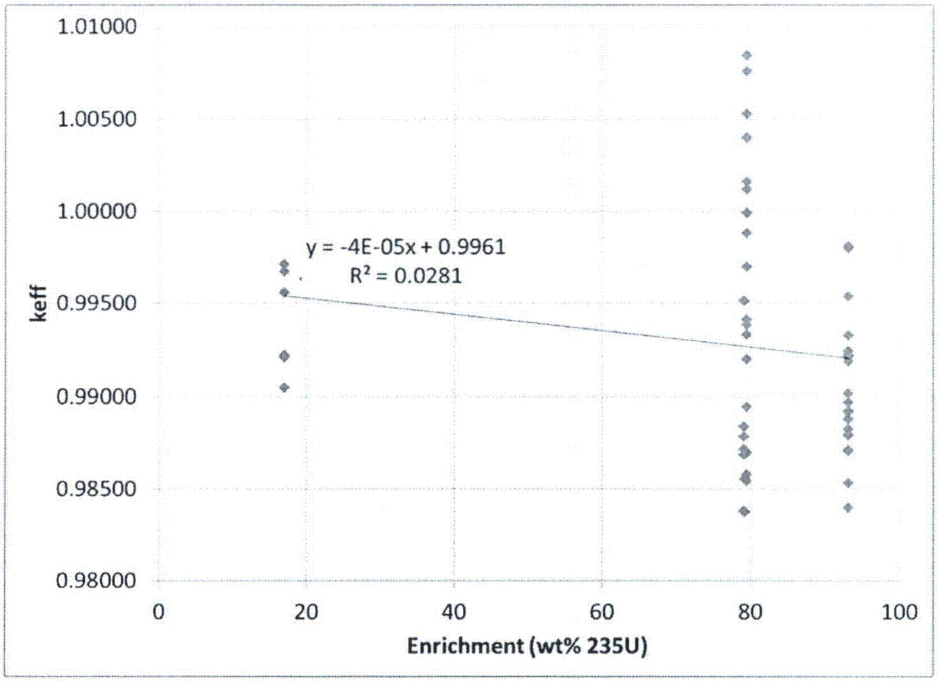
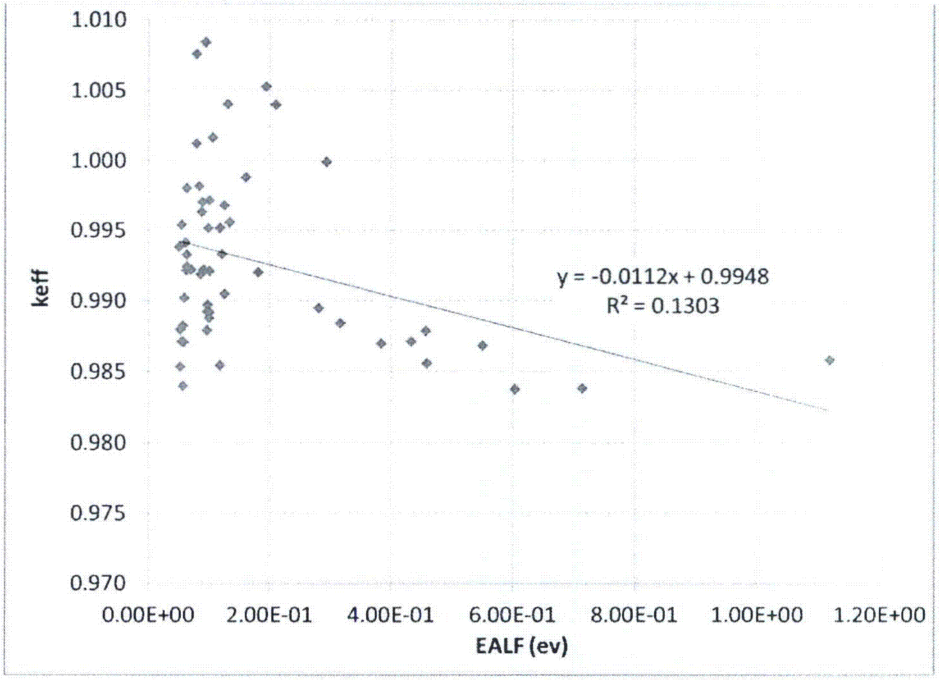


Figure 6.5.5-2 k_{eff} versus Energy of Average Neutron Lethargy Causing Fission (MCNP – Research Reactor Fuel)



6.5.5.3 Results of MCNP Research Reactor Benchmark Calculations

Trending in k_{eff} was evaluated for wt % ^{235}U and energy of the average neutron lethargy causing fission (EALCF), the two parameters most likely to show cross section effects. No statistically significant trends were found for any of the system parameters. USLs are generated for each of the independent variables. A minimum USL covering the range of applicability of the benchmark set is determined.

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. The linear correlation coefficient, R, is calculated by taking the square root of the R^2 value. In particular, the correlation coefficient, R, is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If R is +1, a perfect linear relationship with a positive slope is indicated. If R is -1, a perfect linear relationship with a negative slope is indicated. When R is 0, no linear relationship is indicated.

Table 6.5.5-4 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter. Linear fits and correlation constants are based on full 54 data-point evaluation sets plotted in the previous section.

The USL for each independent variable is calculated and shown with its range of applicability in Table 6.5.5-5. The USLSTATS output of keff versus EALCF and enrichment are shown in Figure 6.5.5-3 and Figure 6.5.5-4. Uncertainties included in the USLSTATS evaluation are the Monte Carlo uncertainty associated with the reactivity calculation and experimental uncertainty that was provided in the literature for each of the cases.

Based on all the independent variable correlations, a lower limit constant USL of 0.9171 may be applied. The range of applicability (area of applicability) of this limit may be extended to lower enrichment as the correlation shows an increase in USL as a function of reduced enrichment.

Figure 6.5.5-3 MCNP Research Reactor Fuel USLSTATS Output for EALCF

```
Proportion of the population = .995
Confidence of fit            = .950
Confidence on proportion     = .950
Number of observations       =   54
Minimum value of closed band =    0.00
Maximum value of closed band =    0.00
Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
3.83770E-01	9.86960E-01	4.65000E-03	9.65800E-02	9.89240E-01	8.11000E-03
2.81040E-01	9.89440E-01	4.68000E-03	9.82040E-02	9.89700E-01	8.11000E-03
1.81510E-01	9.91990E-01	4.66000E-03	9.95040E-02	9.88780E-01	8.11000E-03
1.21140E-01	9.93300E-01	4.62000E-03	1.00600E-01	9.89170E-01	8.11000E-03
8.96800E-02	9.96990E-01	4.62000E-03	9.66350E-02	9.87890E-01	8.11000E-03
2.93880E-01	9.99910E-01	4.65000E-03	8.57350E-02	9.91880E-01	4.49000E-03
2.10270E-01	1.00396E+00	4.66000E-03	7.12960E-02	9.92190E-01	4.09000E-03
1.94070E-01	1.00527E+00	4.62000E-03	6.39970E-02	9.98000E-01	4.08000E-03
1.31260E-01	1.00402E+00	4.61000E-03	6.25280E-02	9.92120E-01	4.08000E-03
9.46680E-02	1.00843E+00	4.66000E-03	5.91670E-02	9.90210E-01	4.07000E-03
7.99520E-02	1.00758E+00	4.63000E-03	5.67170E-02	9.88230E-01	4.08000E-03
1.18230E-01	9.85420E-01	4.62000E-03	5.50180E-02	9.87090E-01	4.06000E-03
6.12710E-02	9.94110E-01	4.60000E-03	5.30120E-02	9.85340E-01	4.06000E-03
1.61160E-01	9.98810E-01	4.64000E-03	5.27070E-02	9.87930E-01	4.06000E-03
7.97960E-02	1.00119E+00	4.56000E-03	8.34310E-02	9.98130E-01	4.10000E-03
1.11980E+00	9.85800E-01	5.96000E-03	6.31130E-02	9.93270E-01	4.08000E-03
1.05990E-01	1.00162E+00	2.35000E-03	5.49410E-02	9.95360E-01	4.06000E-03
5.02850E-02	9.93820E-01	4.89000E-03	5.78770E-02	9.87070E-01	4.07000E-03
7.15000E-01	9.83810E-01	4.29000E-03	5.71950E-02	9.83980E-01	4.07000E-03
5.50720E-01	9.86840E-01	4.31000E-03	6.37490E-02	9.92430E-01	4.08000E-03
4.34030E-01	9.87120E-01	4.29000E-03	9.13040E-02	9.92190E-01	3.96000E-03
6.03930E-01	9.83750E-01	3.34000E-03	1.34700E-01	9.95580E-01	4.07000E-03
4.58840E-01	9.85560E-01	3.49000E-03	1.00680E-01	9.97130E-01	4.46000E-03
4.57170E-01	9.87850E-01	4.30000E-03	1.25750E-01	9.96750E-01	4.46000E-03
3.16220E-01	9.88370E-01	4.39000E-03	1.00650E-01	9.92110E-01	4.36000E-03
1.18400E-01	9.95140E-01	4.88000E-03	1.25740E-01	9.90480E-01	4.47000E-03
9.84130E-02	9.95140E-01	4.97000E-03	8.78000E-02	9.96310E-01	1.40000E-03

```
chi =  0.6296 (upper bound = 9.49). The data tests normal.

      Output from statistical treatment

keff vs EALCF

 Number of data points (n)                       54
 Linear regression, k(X)                        0.9948 + (-1.1154E-02)*X
 Confidence on fit (1-gamma) [input]             95.0%
 Confidence on proportion (alpha) [input]        95.0%
 Proportion of population falling above
 lower tolerance interval (rho) [input]          99.5%
 Minimum value of X                              5.0285E-02
 Maximum value of X                              1.1198E+00
 Average value of X                              1.7980E-01
 Average value of k                               0.99279
 Minimum value of k                               0.98375
 Variance of fit, s(k,X)^2                       3.4489E-05
 Within variance, s(w)^2                         2.3078E-05
 Pooled variance, s(p)^2                         5.7567E-05
 Pooled std. deviation, s(p)                     7.5873E-03
 C(alpha,rho)*s(p)                               3.4132E-02
 student-t @ (n-2,1-gamma)                       1.67620E+00
 Confidence band width, W                        1.5195E-02
 Minimum margin of subcriticality, C*s(p)-W      1.8937E-02

 Upper subcritical limits: ( 5.02850E-02 <= X <=   1.1198     )
 ***** *********** *******

 USL Method 1 (Confidence Band with
 Administrative Margin)             USL1 = 0.9296 + (-1.1154E-02)*X

 USL Method 2 (Single-Sided Uniform
 Width Closed Interval Approach)    USL2 = 0.9607 + (-1.1154E-02)*X

 USLs Evaluated Over Range of Parameter X:
 **** ********* **** ***** ** ********* **

 X:  5.03E-2  2.03E-1  3.56E-1  5.09E-1  6.61E-1  8.14E-1  9.67E-1  1.12E+0
    ---------------------------------------------------------------------
USL-1:    0.9290   0.9273   0.9256   0.9239   0.9222   0.9205   0.9188   0.9171
USL-2:    0.9601   0.9584   0.9567   0.9550   0.9533   0.9516   0.9499   0.9482
```

Figure 6.5.5-4 MCNP Research Reactor Fuel USLSTATS Output for wt% ^{235}U

Proportion of the population = .995
Confidence of fit = .950
Confidence on proportion = .950
Number of observations = 54
Minimum value of closed band = 0.00
Maximum value of closed band = 0.00
Administrative margin = 0.05

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
7.94700E+01	9.86960E-01	4.65000E-03	9.32000E+01	9.89240E-01	8.11000E-03
7.94700E+01	9.89440E-01	4.68000E-03	9.32000E+01	9.89700E-01	8.11000E-03
7.94700E+01	9.91990E-01	4.66000E-03	9.32000E+01	9.88780E-01	8.11000E-03
7.94700E+01	9.93300E-01	4.62000E-03	9.32000E+01	9.89170E-01	8.11000E-03
7.94700E+01	9.96990E-01	4.62000E-03	9.32000E+01	9.87890E-01	8.11000E-03
7.94700E+01	9.99910E-01	4.65000E-03	9.31700E+01	9.91680E-01	4.49000E-03
7.94700E+01	1.00396E+00	4.66000E-03	9.31700E+01	9.92190E-01	4.09000E-03
7.94700E+01	1.00527E+00	4.62000E-03	9.31700E+01	9.98000E-01	4.08000E-03
7.94700E+01	1.00402E+00	4.61000E-03	9.31700E+01	9.92120E-01	4.08000E-03
7.94700E+01	1.00843E+00	4.66000E-03	9.31700E+01	9.90210E-01	4.07000E-03
7.94700E+01	1.00758E+00	4.63000E-03	9.31700E+01	9.88230E-01	4.08000E-03
7.94700E+01	9.85420E-01	4.62000E-03	9.31700E+01	9.87090E-01	4.06000E-03
7.94700E+01	9.94110E-01	4.60000E-03	9.31700E+01	9.85340E-01	4.06000E-03
7.94700E+01	9.98810E-01	4.64000E-03	9.31700E+01	9.87930E-01	4.06000E-03
7.94700E+01	1.00119E+00	4.56000E-03	9.31700E+01	9.98130E-01	4.10000E-03
7.94700E+01	9.85800E-01	5.96000E-03	9.31700E+01	9.93270E-01	4.08000E-03
7.94700E+01	1.00162E+00	2.35000E-03	9.31700E+01	9.95360E-01	4.06000E-03
7.94700E+01	9.93820E-01	4.89000E-03	9.31700E+01	9.87070E-01	4.07000E-03
7.91900E+01	9.83810E-01	4.29000E-03	9.31700E+01	9.83980E-01	4.07000E-03
7.91900E+01	9.86840E-01	4.31000E-03	9.31700E+01	9.92430E-01	4.08000E-03
7.91900E+01	9.87120E-01	4.29000E-03	1.70000E+01	9.92190E-01	3.96000E-03
7.92400E+01	9.83750E-01	3.34000E-03	1.70000E+01	9.95580E-01	4.07000E-03
7.92400E+01	9.85560E-01	3.49000E-03	1.70000E+01	9.97130E-01	4.46000E-03
7.91900E+01	9.87850E-01	4.30000E-03	1.70000E+01	9.96750E-01	4.46000E-03
7.91900E+01	9.88370E-01	4.39000E-03	1.70000E+01	9.92110E-01	4.36000E-03
7.91900E+01	9.95140E-01	4.88000E-03	1.70000E+01	9.90480E-01	4.47000E-03
7.91900E+01	9.95140E-01	4.97000E-03	1.97700E+01	9.96310E-01	1.40000E-03

chi = 0.6296 (upper bound = 9.49). The data tests normal.

Output from statistical treatment

keff vs enrichment

Number of data points (n)	54
Linear regression, k(X)	0.9961 + (-4.3735E-05)*X
Confidence on fit (1-gamma) [input]	95.0%
Confidence on proportion (alpha) [input]	95.0%
Proportion of population falling above lower tolerance interval (rho) [input]	99.5%
Minimum value of X	1.7000E+01
Maximum value of X	9.3200E+01
Average value of X	7.6455E+01
Average value of k	0.99279
Minimum value of k	0.98375
Variance of fit, s(k,X)^2	3.8545E-05
Within variance, s(w)^2	2.3078E-05
Pooled variance, s(p)^2	6.1623E-05
Pooled std. deviation, s(p)	7.8500E-03
C(alpha,rho)*s(p)	3.0787E-02
student-t @ (n-2,1-gamma)	1.67620E+00
Confidence band width, W	1.4021E-02
Minimum margin of subcriticality, C*s(p)-W	1.6767E-02

Upper subcritical limits: (17.000 <= X <= 93.200)
***** *********** *******

USL Method 1 (Confidence Band with Administrative Margin) USL1 = 0.9321 + (-4.3735E-05)*X

USL Method 2 (Single-Sided Uniform Width Closed Interval Approach) USL2 = 0.9653 + (-4.3735E-05)*X

USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** ********* **

USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** ********* **

X:	1.70E+1	2.79E+1	3.88E+1	4.97E+1	6.05E+1	7.14E+1	8.23E+1	9.32E+1
USL-1:	0.9314	0.9309	0.9304	0.9299	0.9295	0.9290	0.9285	0.9280
USL-2:	0.9646	0.9641	0.9637	0.9632	0.9627	0.9622	0.9617	0.9613

Table 6.5.5-4 Range of Applicability and Excel Generated Correlation Coefficients of Research Reactor Fuel Benchmarks

Correlation	R^2	Minimum	Maximum
Enrichment (wt% ^{235}U)	0.0281	17	93.2
Energy of average neutron lethargy causing fission (eV)	0.1303	0.0503	1.12

Table 6.5.5-5 MCNP Research Reactor Fuel USLSTATS Generated USLs for Benchmark Experiments

Variable	Enrichment (wt% ^{235}U)	EALCF (eV)
# Points	54	54
AOA Range	$17 \leq X \leq 93.2$	$0.050285 \leq X \leq 1.1198$
USL Limit	0.9321 -4.3735E-05 X	0.9296 -1.1154E-02 X
USL Low	0.9280	0.9171
USL High	0.9314	0.9290

Note: USL increases as a function of decreasing enrichment. The USL determined from this data may therefore be applied to low enriched research reactor fuels (<17 wt% ^{235}U).

6.7 Payload Specific Details

This section contains NAC-LWT cask payload specific evaluation detail.

6.7.1 PWR Mixed Oxide Fuel Rods

This section includes input, analysis method, results, and criticality benchmark evaluations for the NAC-LWT cask containing a payload of up to 16 PWR rods. The PWR rods may be composed of uranium oxide fuel pellets or mixed oxide fuel pellets (depleted or natural uranium oxide with plutonium oxide contributing the primary quantity of fissile material).

6.7.1.1 Package Fuel Loading

The NAC-LWT cask may transport up to 16 undamaged PWR fuel rods in a fuel rod holder. To bound all PWR MOX rods that may be transported in the NAC-LWT cask, UO_2 rods are evaluated with enrichments up to 5.0 wt % ^{235}U, while MOX rods were evaluated up to 7 wt % fissile plutonium. Characteristics of the design basis PWR rods are presented in Table 6.7.1-12 and Table 6.7.1-13. Given a fixed fissile material density, defined by a maximum UO_2 enrichment or fissile plutonium weight percent, the most reactive rod has the greatest fissile mass, i.e., the rod with the largest pellet radius. Therefore, the CE 14×14 pellet diameter of 0.3765 inch is chosen as the base radius for the most reactive PWR fuel rod evaluated here. A conservative maximum fissile material length of 153.5 inches is also applied. As a maximum reactivity fuel pitch is established, and the zirconium alloy is essentially transparent to neutrons, the clad thickness has no significant effect on the analysis.

6.7.1.2 Criticality Model Specifications

This section describes the models that are used in the criticality analyses for the NAC-LWT cask containing up to 16 PWR rods. PWR rods are either low enriched uranium oxide (maximum 5 wt % ^{235}U) fueled or MOX fueled. The models are analyzed separately under normal conditions and hypothetical accident conditions to ensure that all possible configurations are subcritical.

The model uses the MCNP5 code package with the ENDF/B-VI cross-section set. No cross-section pre-processing is required prior to MCNP implementation. MCNP uses the Monte Carlo technique to calculate the k_{eff} of a system. In these analyses, approximately 530 cycles with 1,000 neutron histories per cycle are tracked through the system.

Description of Calculational Models

The MCNP model of the NAC-LWT cask with 16 undamaged PWR rods includes triangular and square lattice formation of design basis rods centered in the cask cavity. No credit is taken for geometry control provided by the rod holder. The fuel rods, cask cavity and radial shields are explicitly modeled as shown in the Figure 6.7.1-1 model sketch.

The model of the NAC-LWT cask takes advantage of the universe structure of MCNP. Each universe defines an infinite space, bounded after its insertion into a containing cell. Four universes are employed herein. The "0" universe defines the cask universe. The remaining universes are discussed in the following sections. Each universe is developed independently as surfaces and cells. Fuel rod array surfaces and cells are configured to place the rods in either a rectangular (square) pitch array or a hex (triangular) pitch array. The rod pitch is a variable input into the model and is modified to achieve a maximum reactivity configuration. In the basket universe, the rod array is placed into a square (RPP) body that allows the moderator density outside the rod array to be adjusted independently.

The modeled accident condition completely removes the neutron shielding, the neutron shield tank and the cask impact limiters. In the normal conditions model, the impact limiter diameter is modeled as identical to the neutron shield tank diameter. This allows for closer packing for the cask array than physically possible.

VISED sketches of the assembled geometry are shown in Figure 6.7.1-2 through Figure 6.7.1-4. The cask outer surface is surrounded by a rectangular body with reflecting boundary conditions. The boundary conditions are imposed on the sides, top and bottom, which simulates an infinite array of casks.

Package Regional Densities

The composition densities (g/cc) and nuclide number densities (atm/b-cm) used in the subsequent criticality analyses are shown in Table 6.7.1-1. The various isotope weight compositions used in the MOX/UO_2 rod analysis are listed in Table 6.7.1-2.

6.7.2 SLOWPOKE Fuel Rods

This section includes input, analysis method, results, and criticality benchmark evaluations for the NAC-LWT cask containing a payload of up to 800 SLOWPOKE rods. SLOWPOKE rods contain highly enriched uranium in an aluminum matrix material.

6.7.2.1 Package Fuel Loading

The NAC-LWT cask may transport up to 800 undamaged or its equiavalent in damaged fuel rods in fuel canisters. The canisters are screened to prevent the release of gross particulate (note that based on the aluminum metal fuel material no significant release of fuel material from the rods is expected even under severe clad damage conditions). Characteristics of the design basis SLOWPOKE rods are presented in Table 6.7.2-1. Bounding characteristics applied to the model are listed in Table 6.7.2-2. The two characetristics most controlling in system reactivity are increased enrichment and ^{235}U mass per rod. This provides maximum fissile material mass while reducing parasitic absorption in ^{238}U.

6.7.2.2 Criticality Model Specifications

This section describes the models that are used in the criticality analyses for the NAC-LWT cask containing up to 800 SLOWPOKE rods. The models are analyzed separately under normal conditions and hypothetical accident conditions to ensure that all possible configurations are subcritical.

Each model uses the MCNP5 code package with the ENDF/B-VI cross-section set. No cross-section pre-processing is required prior to MCNP implementation. MCNP uses the Monte Carlo technique to calculate the k_{eff} of a system.

Description of Calculational Models

The base MCNP model of the NAC-LWT cask for the analysis of the SLOWPOKE payload is built based on the canister containing an axial stack of four canister inserts each containing a 5x5 fuel tube array (100 rods per canister). The canister is placed into the outer four openings of the NAC-LWT MTR-28 basket configuration. Only the top two baskets are loaded with the bottom two baskets acting as spacers. This allows the loading of up to 800 SLOWPOKE rods (or the equivalent material). Only the 5x5 array is evaluated. The 4x4 array, while allowing increased moderation, removes 36% of the fissile material from the system and will therefore be substantially less reactive. The SLOWPOKE fuel rod is depicted in Figure 6.7.2-1. The fuel rods, cask cavity, and radial shields are explicitly modeled as shown in Figure 6.7.2-2 model sketch.

The model of the NAC-LWT cask takes advantage of the universe structure of MCNP. Each universe defines an infinite space, bounded after its insertion into a containing cell. Four universes are employed herein. The "0" universe defines the cask universe. The remaining universes are discussed in the following sections. Each universe is developed independently as surfaces and cells. The canister interior material is defined separately from the cask cavity material to allow preferential flooding to be evaluated (i.e., different density water in the cask and canister cavity). This option is also exercised when modeling damaged fuel in which the distinct fuel rods are replaced and the canister cavity filled by a homogenized water/fuel mixture.

The modeled accident condition completely removes the neutron shielding, the neutron shield tank, and the cask impact limiters. In the normal conditions model, the impact limiter diameter is modeled as identical to the neutron shield tank diameter. This allows for closer packing for the cask array than physically possible.

VISED sketches of the assembled geometry are shown in Figure 6.7.2-3 through Figure 6.7.2-4. The cask outer surface is surrounded by a cylindrical body with the option of applying reflecting boundary conditions. The reflecing boundary conditions are imposed on the sides, top and bottom, which simulates an infinite array of casks. For single cask analysis (10 CFR 71.55), the cask is surrounded by 20 cm of water to apply full water reflection.

Package Regional Densities

The composition densities (g/cc) and nuclide number densities (atm/b-cm) used in the subsequent criticality analyses are shown in Table 6.7.2-3.

Figure Withheld Under 10 CFR 2.390

Figure 6.7.2-2 MCNP Model Sketch of the NAC-LWT Cask with SLOWPOKE Fuel Rods

(Dimensions in inches)

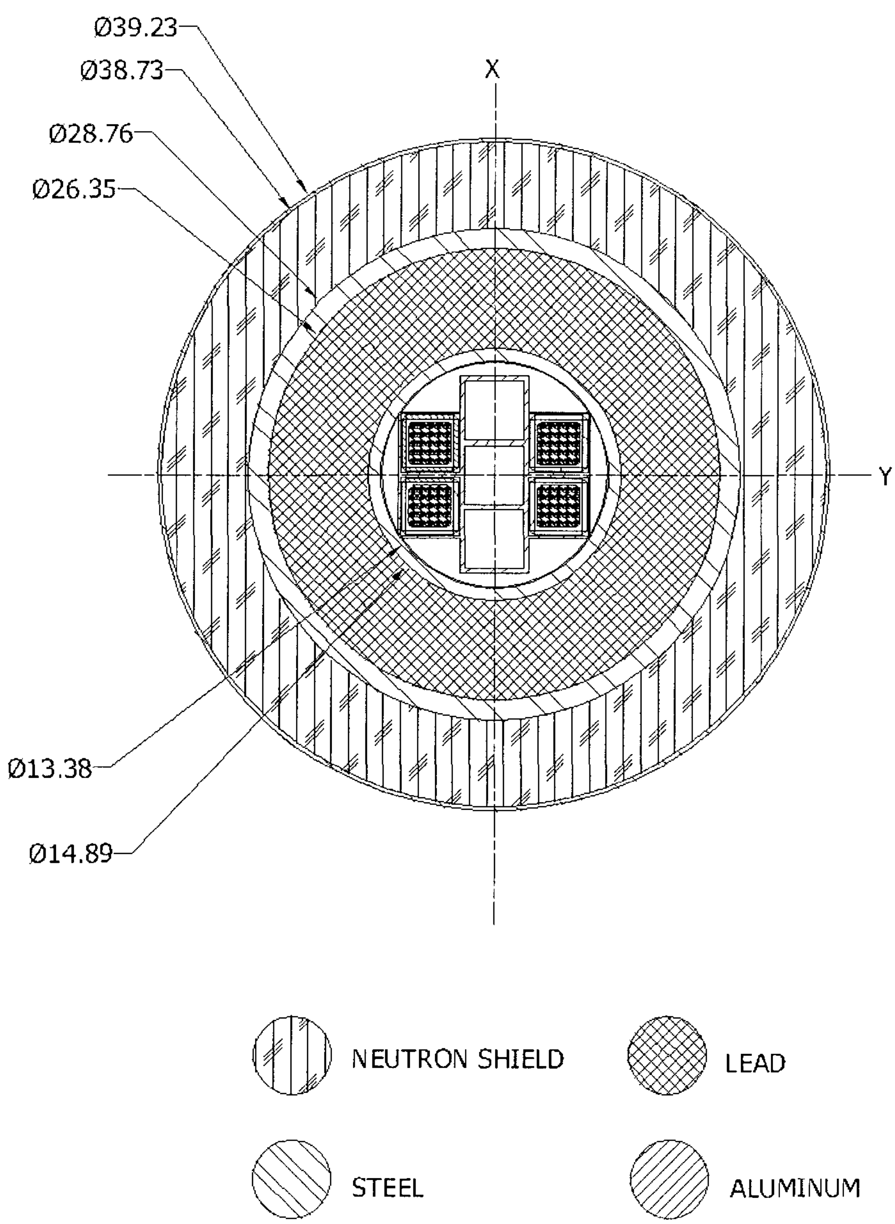


Figure 6.7.2-3 VISED Sketch of LWT Radial View – Undamaged Fuel

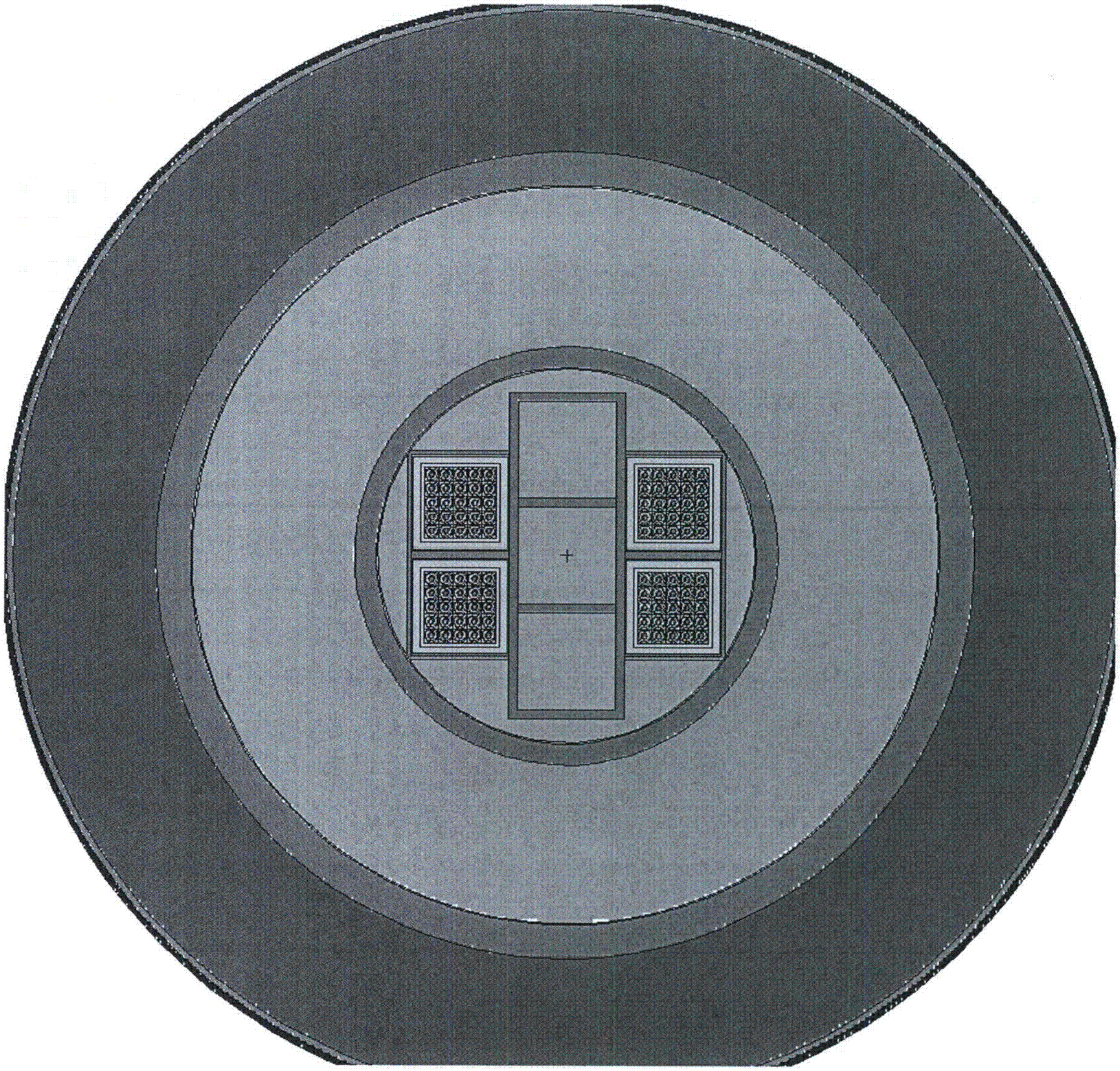
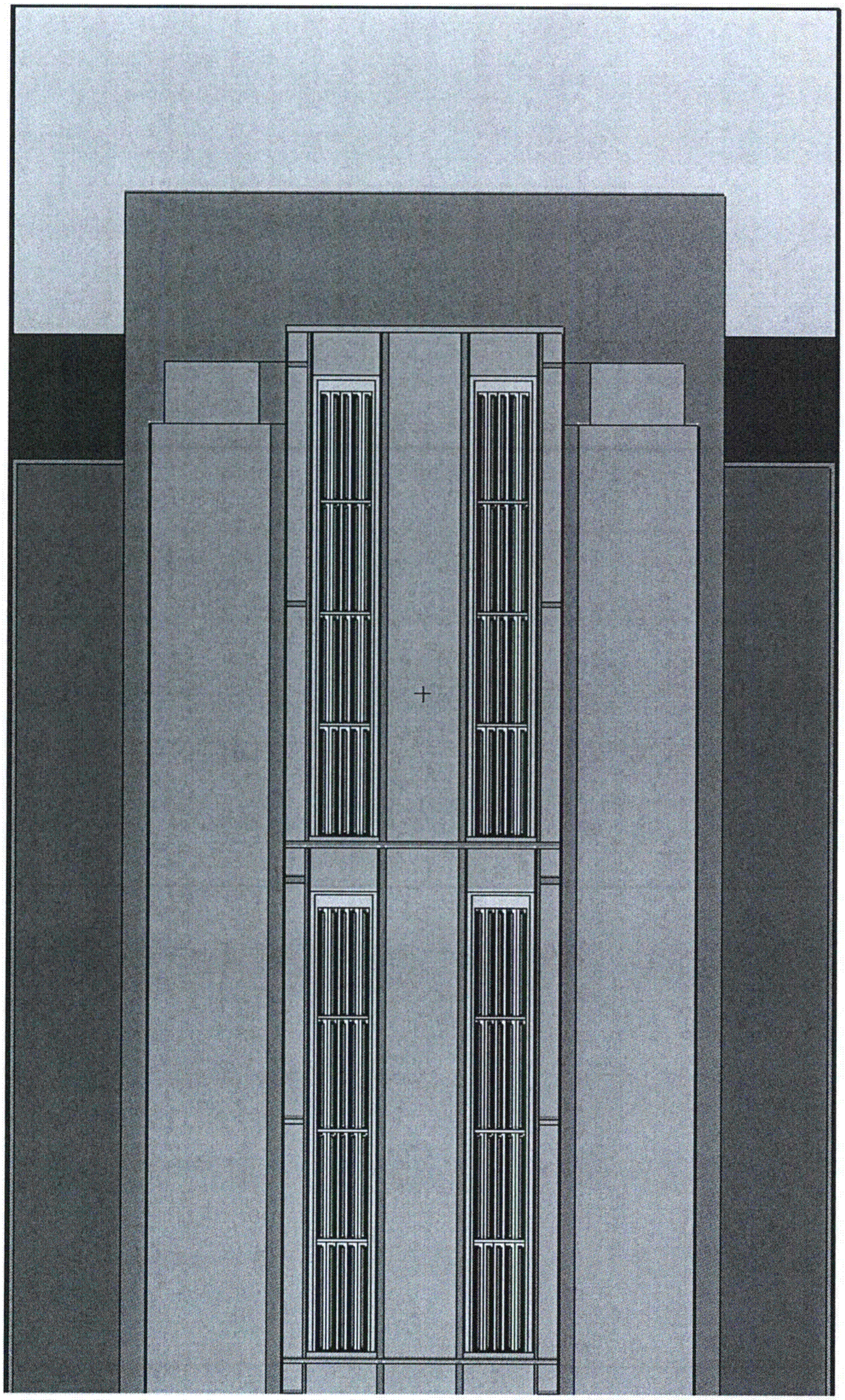


Figure 6.7.2-4 VISED Sketch of LWT Axial View – Undamaged Fuel – Normal Conditions



Note: Model extent shown is limited to loaded baskets. All four baskets modeled.

Table 6.7.2-1 SLOWPOKE Fuel Configuration

Parameter	Value
Fuel Matrix	U-Al Alloy
Clad Material	Al
Clad Thickness, cm	0.051
Fuel Thickness or Diameter, cm	0.422
Rod Length, cm	22.83
Active Fuel Length, cm	22.0
U-235 Enrichment, %	93
U-235 per Rod, g	2.79
U per Rod, g	2.99
Aluminum per Rod (Fuel), g	7.688
Total weight of Fuel per Rod, g	10.678
Rod End Cap Diameter, cm	0.61

Table 6.7.2-2 Modeled SLOWPOKE Fuel Configuration

Parameter	Value
U-235 Enrichment, %	95
U-235 per Rod, g	2.800
U per Rod, g	2.947
Aluminum per Rod (Fuel), g	7.688
Total Weight of Fuel per Rod, g	10.635
Active Fuel Length, cm	22.0
Fuel Thickness or Diameter, cm	0.422
Active Fuel Volume, cc	3.077
Rod Length, cm	22.83
Clad Thickness, cm	0.051
Clad Diameter, cm	0.5240
Rod End Cap Diameter, cm	0.61
Rod End Cap Height, cm	0.4150
Fuel Meat Density, g/cc (calculated)	3.457

Table 6.7.2-3 SLOWPOKE Analysis Compositions and Number Densities

Material	U-Al	H_2O	304 Stainless Steel	Pb	Al
Density, g/cc	3.457	0.9982	7.920	11.344	2.702
Density	atoms/b-cm				
Uranium-235	2.34E-03				
Uranium-238	1.14E-04				
Oxygen		3.338E-2			
Hydrogen		6.677E-2			
Zirconium Alloy					
Iron			5.936E-2		
Chromium			1.743E-2		
Nickel			7.721E-3		
Manganese			1.736E-3		
Lead				3.297E-2	
Aluminum	5.58E-02				6.031E-2

6.7.2.3 Criticality Calculations

The maximum reactivity configuration is determined by performing a series of studies. Based on preliminary calculations and results of other MTR basket payloads, it is initially assumed that accident conditions with close pitch and flooding of the canister interior provide the maximum reactivity configuration. Close pitch (surface reflected) models an infinite array of casks with void (no moderation/absorption) between cask surfaces. Other options for reflection include a cask separation of 40 cm (20 cm reflective surface from cask boundary) or no reflection modeled (single cask with 20-cm boundary at cask exterior conditions). Results of a preliminary study of reactivity versus flooding / configuration are shown in Table 6.7.2-4. A reflective condition of "none" and "wet" cask exterior conditions models a single cask that is fully water reflected. The cask condition of canister interior flooded with dry cask interior and exterior models a hypothetical preferential flooding configuration. The reactivity ($k_{eff} + 2\sigma$) of undamaged fuel is maximized for the preferential flooded configuration at a value of 0.5159 as seen in Table 6.7.2-4.

The maximum reacitivty geometry and moderation studies will be based on the maximum reactivity preliminary case:

- Cask accident conditions
- Close pitch (cask surface reflected)
- Canister flooded, cask interior and exterior dry (preferential flooding)
- Nominal tolerances and no geometry perturbations
- Undamaged fuel

6.7.2.3.1 Geometric Pertubration Study

To observe potential geometry perturbations, the reactivity changes due to fuel and canister shifts were calculated. The base configuration was applied for the perturbation study. The fuel rods were shifted inside their individual tubes towards or away from the center of the 5x5 tube array. The canisters were also shifted towards or away from the center of the MTR-28 basket module. The reactivity results are shown in Table 6.7.2-5. The fuel shifted away from the center of the 5x5 array and canisters shifted toward the center produced a statistical increase (defined as $\Delta k_{eff}/\sigma > 3$). The more reactive outward fuel shift is due to the 5x5 array being under-moderated as is discussed in later sections. The more reactive inward canister shift allows for more neutron interaction between 5x5 arrays (other canisters).

6.7.2.3.2 Material Tolerance Study

The primary purpose of the material tolerance study is to observe the absorption/scatter effects of materials. It is expected that the aluminum of the canister will have no significant effect on the

reactivity. The tolerance for the steel plates in the basket has a range from a low negative and to a large positive. Parasitic absorption for stainless steel is expected to reduce system reactivity at the plus tolerances. The low negative tolerances are expected to have no significant effect. To observe potential tolerance effects in the basket and canister, the densities of the basket and canister materials were modified. A -5% and +25% density changes reflect potential tolerance thickness effects in the basket components. Density changes of ±50% for the aluminum canister are assumed. The reactivity results are shown in Table 6.7.2-6. The -5% density reduction for the toleranced basket produces a statistically negligible effect on reactivity. The increase in modeled steel density (equivalent to plate thickness increase) significantly reduces reactivity. The aluminum canister material variations have no statistical effect on the reactivity results.

6.7.2.3.3 Moderator Density Study (Including Preferential Flooding)

A moderator density study was performed to determine the moderator conditions that result in the most reactive system. The base configuration was changed from close pitch (surface reflected) to a 20-cm reflection surface to observe the variable exterior moderator conditions. Included in the moderator density study is preferential flooding. Preferential flooding includes the hypothetical flooding of the canister, interior, or exterior simultaneously.

The canister interior flooded condition with void in the cask exterior and interior remained the bounding condition (as was assumed in the base configuration). This condition allows moderation within the tube arrays and neutronic coupling of the canisters and casks. The results are tabulated in Table 6.7.2-7 and plotted in Figure 6.7.2-5.

6.7.2.3.4 Maximum Reactivity Configuration for Undamaged Fuel

Based on the previous analyses, the following conditions are bounding for the maximum reactivity configuration:

- Accident conditions
- Close pitch (surface reflected)
- Canister flooded, interior and exterior dry
- Canister shifted toward the center of basket
- Fuel shifted away from the center of 5x5 array

Toleranced components were not included as the reactivity effects were statistically negligible.

6.7.2.3.5 Undamaged Fuel Case Matrix to Conform to 10 CFR 71.55 and 10 CFR 71.59 Requirements

Compliance with the NRC Code of Federal Regulations (CFRs) for the transport of fissile material packages is evaluated. 10 CFR 71.55 general requirements are satisfied by evaluating the following configurations for both normal and accident conditions:

- Single Cask (no MCNP modeled reflective surfaces)
- Close full reflection of the containment system by water on all sides (cask exterior flooded up to 20 cm boundary, fully water reflected)
- Most reactive credible configuration:
  - Canister shifted toward the center of basket
  - Fuel shifted away from the center of 5x5 array
  - Canister interior flooded, cask interior dry

10 CFR 71.59 standards are satisfied by evaluating the maximum reactivity configuration for both normal and accident conditions for an infinite array of casks at close pitch. These requirements are satisfied by modifying the cask and cask exterior configuration of the maximum reactivity configuration established by the preceeding studies. Satisfying 10 CFR 71.59 under an infinite array allows the CSI designation of 0 for the transport package. Both CFR requirements are satisfied and transport of the undamaged fuel is acceptable. The results for the configurations stipulated in the CFR requirements are shown inTable 6.7.2-8 for undamaged fuel. The maximum reactivity configuration produced a reactivity of 0.5222. Satisfying 10 CFR 71.59 under an infinite array allows the CSI designation of 0 for the transport package.

6.7.2.3.6 Damaged Fuel Case Matrix to Conform to 10 CFR 71.55 and 10 CFR 71.59 Requirements

In addition to undamaged fuel, the canister is designed to contain gross fuel material from damaged fuel. The undamaged fuel analysis demonstrated that a significantly higher reactivity is obtained from a dry cask cavity, flooded canister interior, configuration as this configuration allows neutronic interaction between the basket locations (which is reduced by the approximately 3.5 inches of water between the two sets of two canisters), between baskets, and between casks in the infinite array. This configuration is therefore adopted for the damaged fuel analysis. Also adopted is the shifted in-canister configuration which minimizes distance between fissile material components. As fuel rods are replaced by a water/fuel mixture the fuel rod shift configuration is not applicable to damaged fuel. A fuel/water mixture is analyzed for compliance with 10 CFR 71.55 and 10 CFR 71.59. The results are shown in Table 6.7.2-9. The damaged fuel resulted in a maximum reactivity increase of approximately 10% from 0.5222 to

0.5706. Both CFR requirements are satisfied and transport of the damaged fuel is acceptable. As an infinite cask array is evaluated the CSI for damaged fuel is 0.

The maximum reactivity EALCF of 0.08eV is within the area of applicability of the research reactor benchmark. The enrichment of the maximum reactivity case is slightly above the benchmark at 95wt% versus 93.2 wt% in the critical set. The USL of 0.9171 (per Section 6.5.5) is determined using the range of applicability for the EALCF. If the enrichment range of applicability for the USL determination was increased from 93.2wt% to 95wt%, the expected USL decrease determined by applying the enrichment trend functions would be from 0.9280 to 0.9279. The applied USL of 0.9171 based on the EALCF bounds this limit and is therefore acceptable.

Figure 6.7.2-5 SLOWPOKE Moderator Density Study (Percent Full Density Water)

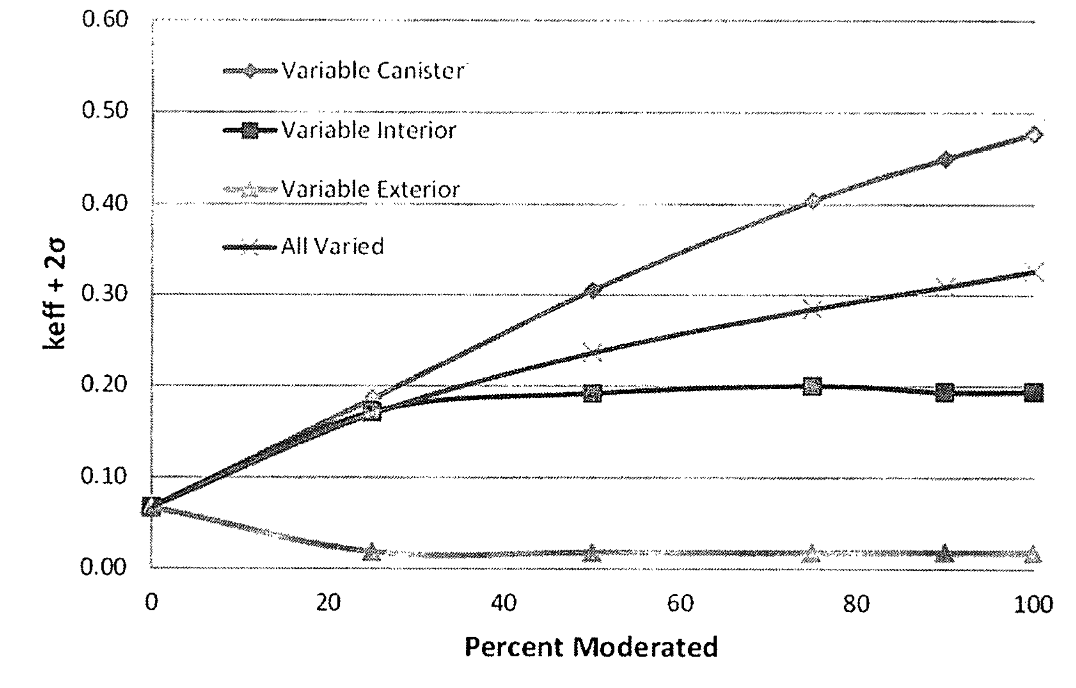


Table 6.7.2-4 Preliminary Reactivity Results for Undamged SLOWPOKE Fuel

Condition	1	2	3	4
Canister Interior	Wet	Wet	Wet	Wet
Cask Interior	Wet	Dry	Dry	Dry
Cask Exterior	N/A	N/A	Dry	Wet
Reflected	Surface	Surface	20cm	None
$k_{eff}+2\sigma$	0.3502	0.5159	0.4774	0.3482

Table 6.7.2-5 SLOWPOKE Component Shift Reactivity Study Results

Parameter		k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Fuel Shift	None	0.5146	0.0007	0.5159	--	--
	In	0.5041	0.0007	0.5055	-0.0104	-11.1
	Out	0.5181	0.0006	0.5193	0.0034	3.8
Can Shift	None	0.5146	0.0007	0.5159	--	--
	In	0.5173	0.0007	0.5186	0.0028	3.0
	Out	0.5122	0.0006	0.5135	-0.0024	-2.7

Table 6.7.2-6 SLOWPOKE Component Tolerance Reactivity Study Results

Parameter	Δ%	k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Basket Tolerance	0%	0.5146	0.0007	0.5159	--	--
	-5%	**0.5159**	**0.0006**	**0.5171**	**0.0013**	**1.37**
	+25%	0.5078	0.0006	0.5091	-0.0068	-7.39
Canister Shell Tolerance	0%	0.5146	0.0007	0.5159	--	--
	-50%	0.5128	0.0006	0.5141	-0.0018	-2.00
	50%	0.5141	0.0006	0.5154	-0.0005	-0.53

Table 6.7.2-7 SLOWPOKE Moderator Density Study

Percent Moderated			Density [g/cc]	k_{eff}	σ	$k_{eff} + 2\sigma$
Canister Interior	Cask Interior	Cask Exterior				
0	0	0	0.00	0.0657	0.0004	0.0666
25	0	0	0.25	0.1837	0.0011	0.1859
50	0	0	0.50	0.3031	0.0010	0.3051
75	0	0	0.75	0.4030	0.0006	0.4041
90	0	0	0.90	0.4486	0.0006	0.4498
100	0	0	0.9982	0.4762	0.0006	0.4774
0	0	0	0.00	0.0657	0.0004	0.0666
0	25	0	0.25	0.1683	0.0016	0.1714
0	50	0	0.50	0.1884	0.0019	0.1921
0	75	0	0.75	0.1969	0.0018	0.2004
0	90	0	0.90	0.1898	0.0019	0.1936
0	100	0	0.9982	0.1903	0.0019	0.1940
0	0	0	0.00	0.0657	0.0004	0.0666
0	0	25	0.25	0.0177	0.0002	0.0182
0	0	50	0.50	0.0175	0.0002	0.0179
0	0	75	0.75	0.0175	0.0002	0.0179
0	0	90	0.90	0.0179	0.0003	0.0184
0	0	100	0.9982	0.0176	0.0003	0.0181
0	0	0	0.00	0.0657	0.0004	0.0666
25	25	25	0.25	0.1667	0.0012	0.1691
50	50	50	0.50	0.2345	0.0012	0.2369
75	75	75	0.75	0.2835	0.0005	0.2846
90	90	90	0.90	0.3079	0.0006	0.3091
100	100	100	0.9982	0.3250	0.0006	0.3262
100	100	0	0.9982	0.3418	0.0006	0.3430
100	0	100	0.9982	0.3485	0.0006	0.3498
0	100	100	0.9982	0.1741	0.0015	0.1771

Table 6.7.2-8 SLOWPOKE Undamaged Fuel Maximum Reactivity Results

				Moderator			Shift		Toleranced		Reactivity Results		
Parameter		Cask	Pitch	Canister	Interior	Exterior	Can	Fuel	Basket	Canister	k_{eff}	σ	$k_{eff}+2\sigma$
CFR 71.59	Accident	Acc	Close	Wet	Dry	Dry	In	Out	No	No	0.5209	0.0006	0.5222
	Normal	Nrm	Close	Wet	Dry	Dry	In	Out	No	No	0.3559	0.0006	0.3571
CFR 71.55	Accident	Acc	Single	Wet	Dry	Wet	In	Out	No	No	0.3529	0.0006	0.3542
	Normal	Nrm	Single	Wet	Dry	Wet	In	Out	No	No	0.3529	0.0006	0.3541

Table 6.7.2-9 SLOWPOKE Damaged Fuel Maximum Reactivity Results

				Moderator			Shift		Toleranced		Reactivity Results		
Parameter		Cask	Pitch	Canister	Interior	Exterior	Can	Fuel	Basket	Canister	keff	σ	$k_{eff}+2\sigma$
CFR 71.59	Accident	Acc	Close	Wet	Dry	Dry	In	N/A	No	No	0.5694	0.0006	0.5706
	Normal	Nrm	Close	Wet	Dry	Dry	In	N/A	No	No	0.3832	0.0005	0.3842
CFR 71.55	Accident	Acc	Single	Wet	Dry	Wet	In	N/A	No	No	0.3803	0.0006	0.3814
	Normal	Nrm	Single	Wet	Dry	Wet	In	N/A	No	No	0.3807	0.0006	0.3818

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6.6.17 SLOWPOKE Fuel MCNP Input

This section contains sample input files from the evaluation of SLOWPOKE fuel elements in the LWT cask. The input files are shown in Figures 6.6.17-1 and 6.6.17-2.

Figure 6.6.17-1 Maximum Reactivity Input for Undamaged SLOWPOKE Fuel

```
NAC-LWT Cask - Accident Transport Conditions
C SlowPoke Fuel - Fuel in Aluminum Tubes
C Fuel Assembly Cells
1  1  -3.4570  -1  2  -3     u=8  $ Fuel Meat
2  2  -2.7000  -4  1  +2  -3    u=8  $ Clad
3  2  -2.7000  -5  3        u=8  $ Top Cap
4  2  -2.7000  -5  -2        u=8  $ Bottom Cap
5  3  -0.9982  #1 #2 #3 #4          u=8  $ Outside Rod
C Canister Related Cells
c Aluminum Tubes - Centerered
31  3  -0.9982  -31          u=7  $ Tube ID
32  5  -2.7000  -32  +31        u=7  $ Tube OD
33  3  -0.9982  32          u=7  $ Outside Tube
c Tube array
41  3  -0.9982  -33  +34  -35  +36      $ Tube Array
       trcl=(0 0 0.635)      lat=1  u=6  fill=-3:3 -3:3 0:0
       6  6  6  6  6  6  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  6  6  6  6  6  6
51  5  -2.7000  -38  -37          u=5  $ Base Plate for Array
52  3  -0.9982  -41  fill=8  trcl = ( -2.6834 2.6834 0.6351 )      u=5  $ Fuel Rod 1
53  like  52  but  fill=8  trcl = ( -1.4134 2.6834 0.6351 )      u=5  $ Fuel Rod 2
54  like  52  but  fill=8  trcl = ( 0.0000 2.7429 0.6351 )      u=5  $ Fuel Rod 3
55  like  52  but  fill=8  trcl = ( 1.4134 2.6834 0.6351 )      u=5  $ Fuel Rod 4
56  like  52  but  fill=8  trcl = ( 2.6834 2.6834 0.6351 )      u=5  $ Fuel Rod 5
57  like  52  but  fill=8  trcl = ( -2.6834 1.4134 0.6351 )      u=5  $ Fuel Rod 6
58  like  52  but  fill=8  trcl = ( -1.4134 1.4134 0.6351 )      u=5  $ Fuel Rod 7
59  like  52  but  fill=8  trcl = ( 0.0000 1.4729 0.6351 )      u=5  $ Fuel Rod 8
60  like  52  but  fill=8  trcl = ( 1.4134 1.4134 0.6351 )      u=5  $ Fuel Rod 9
61  like  52  but  fill=8  trcl = ( 2.6834 1.4134 0.6351 )      u=5  $ Fuel Rod 10
62  like  52  but  fill=8  trcl = ( -2.7429 0.0000 0.6351 )      u=5  $ Fuel Rod 11
63  like  52  but  fill=8  trcl = ( -1.4729 0.0000 0.6351 )      u=5  $ Fuel Rod 12
64  like  52  but  fill=8  trcl = ( 0.0000 0.0000 0.6351 )      u=5  $ Fuel Rod 13
65  like  52  but  fill=8  trcl = ( 1.4729 0.0000 0.6351 )      u=5  $ Fuel Rod 14
66  like  52  but  fill=8  trcl = ( 2.7429 0.0000 0.6351 )      u=5  $ Fuel Rod 15
67  like  52  but  fill=8  trcl = ( -2.6834 -1.4134 0.6351 )      u=5  $ Fuel Rod 16
68  like  52  but  fill=8  trcl = ( -1.4134 -1.4134 0.6351 )      u=5  $ Fuel Rod 17
69  like  52  but  fill=8  trcl = ( 0.0000 -1.4729 0.6351 )      u=5  $ Fuel Rod 18
70  like  52  but  fill=8  trcl = ( 1.4134 -1.4134 0.6351 )      u=5  $ Fuel Rod 19
71  like  52  but  fill=8  trcl = ( 2.6834 -1.4134 0.6351 )      u=5  $ Fuel Rod 20
72  like  52  but  fill=8  trcl = ( -2.6834 -2.6834 0.6351 )      u=5  $ Fuel Rod 21
73  like  52  but  fill=8  trcl = ( -1.4134 -2.6834 0.6351 )      u=5  $ Fuel Rod 22
74  like  52  but  fill=8  trcl = ( 0.0000 -2.7429 0.6351 )      u=5  $ Fuel Rod 23
75  like  52  but  fill=8  trcl = ( 1.4134 -2.6834 0.6351 )      u=5  $ Fuel Rod 24
76  like  52  but  fill=8  trcl = ( 2.6834 -2.6834 0.6351 )      u=5  $ Fuel Rod 25
77  3  -0.9982  -38  +37
              #52 #53 #54 #55 #56 #57 #58 #59 #60 #61 #62 #63 #64 #65
              #66 #67 #68 #69 #70 #71 #72 #73 #74 #75 #76
              fill=6          u=5  $ Tube Array Arround Fuel Rods
78  3  -0.9982  +38          u=5  $ Tube Array Exterior
c Canister for four tube arrays
81  5  -2.7000  -40  +39          u=4  $ Can Base and Shell
82  3  -0.9982  -38  fill=5  trcl = ( 0.0000 0.0000 0.9652 )      u=4  $ Tube Assy 1
83  like  82  but  fill=5  trcl = ( 0.0000 0.0000 25.0952 )      u=4  $ Tube Assy 2
84  like  82  but  fill=5  trcl = ( 0.0000 0.0000 49.2252 )      u=4  $ Tube Assy 3
85  like  82  but  fill=5  trcl = ( 0.0000 0.0000 73.3552 )      u=4  $ Tube Assy 4
86  3  -0.9982  -39  #82 #83 #84 #85          u=4  $ Canister Cavity
87  4  -0.0001  40          u=4  $ Canister Exterior
C Cells - MTR 7 Element Basket
91  6  -7.9400  -91  +94 +95 +96 +97 +98 +99 +100        u=3  $ Base plate
92  6  -7.9400  -92  +101 +105          u=3  $ Support plate
93  6  -7.9400  -93  +101 +105          u=3  $ Support plate
94  6  -7.9400  -101  +102  #91 #92 #93        u=3  $ Center column
95  6  -7.9400  -103  #91 #92 #93          u=3  $ Center divider upper
96  6  -7.9400  -104  #91 #92 #93          u=3  $ Center divider lower
97  6  -7.9400  -105  +106  +101  #91 #92 #93        u=3  $ Small side
98  6  -7.9400  -107  #91 #92 #93          u=3  $ Left divider
99  6  -7.9400  -108  #91 #92 #93          u=3  $ Right divider
100  4  -0.0001  #91 #92 #93 #94 #95 #96 #97 #98 #99          u=3  $ Cask Cavity Material
C Cells - Basket Cell Opening
151  4  -0.0001  -151  fill=4  trcl = ( -9.3503 4.5222 1.2700 )      u=2  $ UL
152  like  151  but    trcl = ( -9.3503 -4.5222 1.2700 )      u=2  $ LL
153  like  151  but    trcl = ( 9.3503 4.5222 1.2700 )      u=2  $ UR
154  like  151  but    trcl = ( 9.3503 -4.5222 1.2700 )      u=2  $ LR
155  4  -0.0001  #151 #152 #153 #154          fill=3  u=2  $ Basket Materials
C Cells - LWT Cavity
201  4  -0.0001  -201  fill=3  ( 0.0000 0.0000 3.8100 )      u=1
202  4  -0.0001  -202  fill=3  ( 0.0000 0.0000 115.5700 )      u=1
203  4  -0.0001  -203  fill=2  ( 0.0000 0.0000 227.3300 )      u=1
204  4  -0.0001  -204  fill=2  ( 0.0000 0.0000 339.0900 )      u=1
205  4  -0.0001  #201 #202 #203 #204          u=1
C Cells - LWT Cask Accident Conditions
301  7  -11.344  -304          $ BotPb
302  4  -0.0001  -303  fill=1          $ Cavity
303  6  -7.9400  -302  +304          $ Bottom
304  6  -7.9400  -301  +302  +306  +309  +303      $ OuterShell
305  6  -7.9400  -305  +308  +303          $ InnerShellTaper
306  6  -7.9400  -307  +303          $ InnerShell
307  7  -11.344  -308  +307          $ Lead
```

```
308  7  -11.344  -306  +305  +308        $ LeadTaper
309  0           -309  +308          $ LeadGap
310  9  -0.0001  +301  -310          $ Gap To Reflector
311  0           +310          $ Outside

C Fuel Assembly Surfaces
1  CZ  0.2110              $ Fuel Meat
2  PZ  0.4150              $ Lower Meat Elevation
3  PZ  22.4150              $ Upper Meat Elevation
4  CZ  0.2620              $ Clad
5  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  22.8300  0.3051  $ Rod Outline
C Canister Surfaces
c Aluminum Tubes
31  CZ  0.5080              $ Tube ID
32  CZ  0.6350              $ Tube OD
33  PX  0.6350
34  PX  -0.6350
35  PY  0.6350
36  PY  -0.6350              $ Tube Pitch Box
c Tube array base and outer envelope
37  PZ   0.6351              $ Base Plate Top Elevation
38  RPP  -3.1750  3.1750  -3.1750  3.1750  0.0000  24.1300    $ Tube Container
c Canister for four tube arrays
39  RPP  -3.5560  3.5560  -3.5560  3.5560  0.9652  100.1776    $ Can Cavity
40  RPP  -4.1910  4.1910  -4.1910  4.1910  0.0000  101.1428    $ Can Outer
41  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  22.8300  0.3050  $ Rod Outline
C Surfaces - MTR 7 Element Basket
91  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  1.2700  16.8466  $ Base plate
92  RCC  0.0000  0.0000  52.0700  0.0000  0.0000  1.2700  16.8466  $ Support plate
93  RCC  0.0000  0.0000  104.1400  0.0000  0.0000  1.2700  16.8466  $ Support plate
94  CZ  1.2700              $ Hole CC
95  C/Z  0.0000  9.5250  1.2700          $ Hole UC
96  C/Z  0.0000  -9.5250  1.2700          $ Hole LC
97  C/Z  -9.5250  4.6990  1.2700          $ Hole UL
98  C/Z  -9.5250  -4.6990  1.2700          $ Hole LL
99  C/Z  9.5250  4.6990  1.2700          $ Hole UR
100  C/Z  9.5250  -4.6990  1.2700          $ Hole LR
101  RPP  -5.1604  5.1604  -14.6939  14.6939  1.2700  111.7600    $ Center column outer
102  RPP  -4.3667  4.3667  -13.9002  13.9002  1.2700  111.7600    $ Center column inner
103  RPP  -4.3667  4.3667  4.3688  5.1626  1.2700  111.7600   $ Center divider upper
104  RPP  -4.3667  4.3667  -5.1626  -4.3688  1.2700  111.7600    $ Center divider lower
105  RPP  -14.1986  14.1986  -9.3599  9.3599  1.2700  111.7600    $ Small side outer
106  RPP  -13.8938  13.8938  -9.0551  9.0551  1.2700  111.7600    $ Small side inner
107  RPP  -13.8938  -5.1604  -0.3175  0.3175  1.2700  111.7600    $ Left divider
108  RPP  5.1604  13.8938  -0.3175  0.3175  1.2700  111.7600    $ Right divider
C Surfaces - Basket Cell Opening
151  RPP  -4.1909  4.1911  -4.1909  4.1911  1.2700  111.7600    $ Opening in Basket
C Surfaces - LWT Cavity
201  RCC  0.0000  0.0000  3.8100  0.0000  0.0000  111.7600  16.8467  $ Basket
202  RCC  0.0000  0.0000  115.5700  0.0000  0.0000  111.7600  16.8467  $ Basket
203  RCC  0.0000  0.0000  227.3300  0.0000  0.0000  111.7600  16.8467  $ Basket
204  RCC  0.0000  0.0000  339.0900  0.0000  0.0000  111.7600  16.8467  $ Basket
C Surfaces - LWT Cask Accident Conditions
301  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  507.3650  36.5189  $ Lwt
302  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  26.6700  36.5189  $ Bottom
303  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  452.1200  16.9863  $ Cavity
304  RCC  0.0000  0.0000  -17.7800  0.0000  0.0000  7.6200  26.3525  $ Bottom gamma shield
305  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  20.1740  $ Lead id - taper
306  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  31.5976  $ Lead od - taper
307  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  16.9103  $ Lead id
308  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.3271  $ Lead od
309  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.4645  $ Lead gap
*310  RCC  0.0000  0.0000  -27.1700  0.0000  0.0000  508.3650  37.0189  $ Container

C
C Materials List
C
C - U-Al
m1  92235  -2.6400E-01
       92238  -1.3000E-02
       13027  -7.2300E-01
C Aluminum / Clad
m2  13027  -1.0
C Canister Water
m3  1001  6.6667E-01  8016  3.3333E-01
mt3  lwtr.01
C Cask Cavity Water
m4  1001  6.6667E-01  8016  3.3333E-01
mt4  lwtr.01
C Aluminum
m5  13027  -1.0
C Stainless Steel 304
m6  26000  -0.695  24000  -0.190  28000  -0.095
       25055  -0.020
C Lead
m7  82000  -1.0
C Aluminum Honeycomb Impact Limiter
m8  13027  -1.0
C Water/Glycol - Cask Neutron Shield
m10  1001  -1.03651E-01  8016  -6.75619E-01  6000  -2.20730E-01
C Cask Exterior (Water at Various Densities)
m9  1001  6.6667E-01  8016  3.3333E-01
mt9  lwtr.01
C
C Cell Importances
imp:n 1 73r 0
c
```

```
c Criticality Controls
c
kcode  2000  1.00  30  530
c
c Source Distribution for Initial Generation
SDEF  CEL=  302:D4:D5:D6:D7:-1
       ERG=  D1
       POS=  0.0000  0.00  0.4150
       RAD=  D2
       AXS=  0.00  0.00  1.00
       EXT=  D3
C - Neutron Source Energy Source Distribution
#         SP1
       -3
C - Uniform Radial Distribution in Fuel Rod
#  SI2  SP2
       0.0000  -21
       0.2110  1
C - Axial Source Profile
#  SI3  SP3
       0  0.0
       22  1.0
C - Two Baskets With Fuel in Cask
#     SI4         SP4
          1         d
       203  1
       204  1
C - Four Openings in Basket with Fuel
#     SI5         SP5
          1         d
       151  1
       152  1
       153  1
       154  1
C - Four Tube Arrays per Canister
#     SI6         SP6
          1         d
       82  1
       83  1
       84  1
       85  1
C - 25 Fuel Rods
#     SI7         SP7
          1         d
       52  1
       53  1
       54  1
       55  1
       56  1
       57  1
       58  1
       59  1
       60  1
       61  1
       62  1
       63  1
       64  1
       65  1
       66  1
       67  1
       68  1
       69  1
       70  1
       71  1
       72  1
       73  1
       74  1
       75  1
       76  1
C Print Control
prdmp  -30 -60 1 2
print
C Random Number Generator
rand  gen=2  1.90735E+13  stride=152917  hist=1
```

Figure 6.6.17-2 Maximum Reactivity Input for Damaged SLOWPOKE Fuel

```
NAC-LWT Cask - Accident Transport Conditions
C SlowPoke Fuel - Fuel in Aluminum Tubes
C Canister Related Cells
c Aluminum Tubes - Centerered
31  1  -1.2795  -31           u=7  $ Tube ID
32  5  -2.7000  -32  +31         u=7  $ Tube OD
33  1  -1.2795  32          u=7  $ Outside Tube
c Tube array
41  1  -1.2795  -33  +34  -35  +36      $ Tube Array
       trcl=(0 0 0.635)      lat=1  u=6  fill=-3:3 -3:3 0:0
       6  6  6  6  6  6  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  7  7  7  7  7  6
       6  6  6  6  6  6  6
51  5  -2.7000  -38  -37         u=5  $ Base Plate for Array
52  1  -1.2795  -38  +37
            fill=6          u=5  $ Tube Array Arround Fuel Rods
53  1  -1.2795  +38           u=5  $ Tube Array Exterior
c Canister for four tube arrays
81  5  -2.7000  -40  +39         u=4  $ Can Base and Shell
82  1  -1.2795  -38  fill=5  trcl = ( 0.0000 0.0000 0.9652 )      u=4  $ Tube Assy 1
83  like  82  but  fill=5  trcl = ( 0.0000 0.0000 25.0952 )      u=4  $ Tube Assy 2
84  like  82  but  fill=5  trcl = ( 0.0000 0.0000 49.2252 )      u=4  $ Tube Assy 3
85  like  82  but  fill=5  trcl = ( 0.0000 0.0000 73.3552 )      u=4  $ Tube Assy 4
86  1  -1.2795  -39  #82 #83 #84 #85        u=4  $ Canister Cavity
87  4  -0.0001  40          u=4  $ Canister Exterior
C Cells - MTR 7 Element Basket
91  6  -7.9400  -91  +94 +95 +96 +97 +98 +99 +100        u=3  $ Base plate
92  6  -7.9400  -92  +101 +105        u=3  $ Support plate
93  6  -7.9400  -93  +101 +105        u=3  $ Support plate
94  6  -7.9400  -101  +102  #91 #92 #93        u=3  $ Center column
95  6  -7.9400  -103  #91 #92 #93        u=3  $ Center divider upper
96  6  -7.9400  -104  #91 #92 #93        u=3  $ Center divider lower
97  6  -7.9400  -105  +106  +101  #91 #92 #93     u=3  $ Small side
98  6  -7.9400  -107  #91 #92 #93        u=3  $ Left divider
99  6  -7.9400  -108  #91 #92 #93        u=3  $ Right divider
100  4  -0.0001  #91 #92 #93 #94 #95 #96 #97 #98 #99          u=3  $ Cask Cavity Material
C Cells - Basket Cell Opening
151  4  -0.0001  -151  fill=4  trcl = ( -9.3503 4.5222 1.2700 )      u=2  $ UL
152  like  151  but    trcl = ( -9.3503 -4.5222 1.2700 )      u=2  $ LL
153  like  151  but    trcl = ( 9.3503 4.5222 1.2700 )      u=2  $ UR
154  like  151  but    trcl = ( 9.3503 -4.5222 1.2700 )      u=2  $ LR
155  4  -0.0001  #151 #152 #153 #154          fill=3  u=2  $ Basket Materials
C Cells - LWT Cavity
201  4  -0.0001  -201  fill=3  ( 0.0000 0.0000 3.8100 )      u=1
202  4  -0.0001  -202  fill=3  ( 0.0000 0.0000 115.5700 )      u=1
203  4  -0.0001  -203  fill=2  ( 0.0000 0.0000 227.3300 )      u=1
204  4  -0.0001  -204  fill=2  ( 0.0000 0.0000 339.0900 )      u=1
205  4  -0.0001  #201 #202 #203 #204          u=1
C Cells - LWT Cask Accident Conditions
301  7  -11.344  -304            $ BotPb
302  4  -0.0001  -303  fill=1          $ Cavity
303  6  -7.9400  -302  +304          $ Bottom
304  6  -7.9400  -301  +302  +306  +309  +303     $ OuterShell
305  6  -7.9400  -305  +308  +303          $ InnerShellTaper
306  6  -7.9400  -307  +303          $ InnerShell
307  7  -11.344  -308  +307          $ Lead
308  7  -11.344  -306  +305  +308          $ LeadTaper
309  0           -309  +308          $ LeadGap
310  9  -0.0001  +301  -310          $ Gap To Reflector
311  0           +310          $ Outside

C Canister Surfaces
c Aluminum Tubes
31  CZ  0.5080             $ Tube ID
32  CZ  0.6350             $ Tube OD
33  PX  0.6350
34  PX  -0.6350
35  PY  0.6350
36  PY  -0.6350            $ Tube Pitch Box
c Tube array base and outer envelope
37  PZ   0.6351            $ Base Plate Top Elevation
38  RPP  -3.1750  3.1750  -3.1750  3.1750  0.0000  24.1300    $ Tube Container
c Canister for four tube arrays
39  RPP  -3.5560  3.5560  -3.5560  3.5560  0.9652  100.1776    $ Can Cavity
40  RPP  -4.1910  4.1910  -4.1910  4.1910  0.0000  101.1428    $ Can Outer
41  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  22.8300  0.3050  $ Rod Outline
C Surfaces - MTR 7 Element Basket
91  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  1.2700  16.8466  $ Base plate
92  RCC  0.0000  0.0000  52.0700  0.0000  0.0000  1.2700  16.8466  $ Support plate
93  RCC  0.0000  0.0000  104.1400  0.0000  0.0000  1.2700  16.8466  $ Support plate
94  CZ  1.2700             $ Hole CC
95  C/Z  0.0000  9.5250  1.2700         $ Hole UC
96  C/Z  0.0000  -9.5250  1.2700         $ Hole LC
97  C/Z  -9.5250  4.6990  1.2700         $ Hole UL
98  C/Z  -9.5250  -4.6990  1.2700         $ Hole LL
99  C/Z  9.5250  4.6990  1.2700         $ Hole UR
100  C/Z  9.5250  -4.6990  1.2700         $ Hole LR
101  RPP  -5.1604  5.1604  -14.6939  14.6939  1.2700  111.7600    $ Center column outer
102  RPP  -4.3667  4.3667  -13.9002  13.9002  1.2700  111.7600    $ Center column inner
```

```
103  RPP  -4.3667  4.3667  4.3688  5.1626  1.2700  111.7600    $ Center divider upper
104  RPP  -4.3667  4.3667  -5.1626  -4.3688  1.2700  111.7600    $ Center divider lower
105  RPP  -14.1986  14.1986  -9.3599  9.3599  1.2700  111.7600    $ Small side outer
106  RPP  -13.8936  13.8936  -9.0551  9.0551  1.2700  111.7600    $ Small side inner
107  RPP  -13.8936  -5.1604  -0.3175  0.3175  1.2700  111.7600    $ Left divider
108  RPP  5.1604  13.8936  -0.3175  0.3175  1.2700  111.7600    $ Right divider
C Surfaces - Basket Cell Opening
151  RPP  -4.1909  4.1911  -4.1909  4.1911  1.2700  111.7600    $ Opening in Basket
C Surfaces - LWT Cavity
201  RCC  0.0000  0.0000  3.8100  0.0000  0.0000  111.7600  16.8467  $ Basket
202  RCC  0.0000  0.0000  115.5700  0.0000  0.0000  111.7600  16.8467  $ Basket
203  RCC  0.0000  0.0000  227.3300  0.0000  0.0000  111.7600  16.8467  $ Basket
204  RCC  0.0000  0.0000  339.0900  0.0000  0.0000  111.7600  16.8467  $ Basket
C Surfaces - LWT Cask Accident Conditions
301  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  507.3650  36.5189  $ Lwt
302  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  26.6700  36.5189  $ Bottom
303  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  452.1200  16.9863  $ Cavity
304  RCC  0.0000  0.0000  -17.7800  0.0000  0.0000  7.6200  26.3525  $ Bottom gamma shield
305  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  20.1740  $ Lead id - taper
306  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  31.5976  $ Lead od - taper
307  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  18.9103  $ Lead id
308  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.3271  $ Lead od
309  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.4645  $ Lead gap
*310  RCC  0.0000  0.0000  -27.1700  0.0000  0.0000  508.3650  37.0189  $ Container

C
C Materials List
C
C - U-Al_H2O
m1  92235  -5.7000E-02  1001  -7.5000E-02
       92238  -3.0000E-03  8016  -6.0400E-01
       13027  -2.6100E-01
C Aluminum / Clad
m2  13027  -1.0
C Canister Water
m3  1001  6.6667E-01  8016  3.3333E-01
mt3  lwtr.01
C Cask Cavity Water
m4  1001  6.6667E-01  8016  3.3333E-01
mt4  lwtr.01
C Aluminum
m5  13027  -1.0
C Stainless Steel 304
m6  26000  -0.695  24000  -0.190  28000  -0.095
       25055  -0.020
C Lead
m7  82000  -1.0
C Aluminum Honeycomb Impact Limiter
m8  13027  -1.0
C Water/Glycol - Cask Neutron Shield
m10  1001  -1.03651E-01  8016  -6.75619E-01  6000  -2.20730E-01
C Cask Exterior (Water at Various Densities)
m9  1001  6.6667E-01  8016  3.3333E-01
mt9  lwtr.01
C
C Cell Importances
imp:n 1 43r 0
c
c Criticality Controls
c
kcode  2000  1.00  30  530
c
c Source Distribution for Initial Generation
SDEF  CEL=  302:D4:D5:D6
       ERG=  D1
       POS=  0.0000  0.00  0.4150
       RAD=  D2
       AXS=  0.00  0.00  1.00
       EXT=  D3
C - Neutron Source Energy Source Distribution
#          SP1
       -3
C - Uniform Radial Distribution in Fuel Rod
#  SI2  SP2
       0.0000  -21
       3.5560  1
C - Axial Source Profile
#  SI3  SP3
       0  0.0
       100  1.0
C - Two Baskets With Fuel in Cask
#     SI4         SP4
         1          d
       203  1
       204  1
C - Four Openings in Basket with Fuel
#     SI5         SP5
         1          d
       151  1
       152  1
       153  1
       154  1
C - Four Tube Arrays per Canister
#     SI6         SP6
         1          d
       82  1
       83  1
```

```
       84  1
       85  1
C Print Control
prdmp  -30 -60 1 2
print
C Random Number Generator
rand  gen=2  1.90735E+13  stride=152917  hist=1
```

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if the Alternate B port covers are used, the metallic O-ring seal will be replaced for each transport following component removal and the helium maintenance leakage rate test is required to be performed.

For cask loading operations performed under water or when water is introduced into the cask cavity, the cask cavity is required to be blown down to remove the cavity water, vacuum dried, verified as dry, and helium backfilled prior to final closure and leakage testing. The cavity is vacuum dried by attaching a vacuum pump to the vent and/or drain port and evacuating the cavity to a pressure of less than 10 torr (13 mbar), and continuing to vacuum pump for an additional 15 minutes. If the cavity pressure rise is less than 5 torr (6.7 mbar) during a 10-minute isolation and hold period, there is no free water in the cavity and the cask cavity is verified as dry. Final containment closure and leakage testing operations in preparation for transport can proceed. If the pressure rise is >5 torr (6.7 mbar), the vacuum drying will be continued until the dryness verification criteria are met. The successful performance of the dryness verification and backfilling the cavity with helium ensures that there is no free water in the cavity and oxidation of the cask's contents is precluded. When the cask is loaded in a dry cell or under other conditions where no water is introduced into the cask cavity, the procedure sequences for cavity blow down, vacuum drying and dryness verification can be eliminated and the loading sequence can proceed directly to final closure, containment boundary leakage testing and helium backfill operations.

7.1.1 Procedures for Wet Loading of LWR Fuel Assemblies and Canistered LWR Fuel Rods

The procedures for wet loading the NAC-LWT with LWR fuel are as follows:

1. Perform a receipt inspection of the empty cask and trailer/ISO container, inspecting for transport damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release brakes and remove the chocks when required to complete uprighting operations. If an ISO container is used, it may be removed from the trailer and secured in the unloading area.
3. Remove the personnel barrier or the roof and roof cross-members from the ISO container.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.
4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

Note: A receiving survey of the cask and transporter must be performed as soon as practicable after arrival at the site to assure compliance with 10 CFR 20, 10 CFR 71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.

5. Remove the top and bottom impact limiters.
6. Remove the cask tie-down strap.
7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to keep the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.
8. Place the cask in the cask preparation area or other designated location. Disengage the lifting yoke. Clean cask surfaces of road dirt as required for entry into the spent fuel pool.
9. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.
10. Remove the vent and drain valve port covers. Prior to reinstallation of the port covers, carefully inspect the valve port cover O-ring seals and, if the O-rings show any damage, replace them with approved spares. Ensure that the replacement O-rings are properly installed and seated. Visually inspect the valved quick-disconnect nipples and replace them, if necessary.

    Note: For Alternate B port covers, replace the metallic O-ring with an approved spare prior to reinstallation.
11. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid and, if it shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-rings are properly installed and seated. Inspect the lid bolts and replace any that are damaged.
12. Visually inspect the inner cavity for foreign material or damage. Install or verify the presence of the proper drain tube and basket assembly.
13. Fill the cask cavity with clean water.
14. Install lift yoke arm guides and remote actuation component on the cask lifting yoke.
15. Engage the cask lifting yoke with the cask lifting trunnions and pick up the cask. Carefully lower the cask to the bottom of the cask loading area. Rinse the cask surfaces with clean water to minimize cask surface contamination.
16. Disengage the lifting yoke from the cask and remove the yoke from the pool, if necessary, to provide fuel loading clearance.

release brakes and remove the chocks when required to complete uprighting operations. If an ISO container is used, it may be removed from the trailer and secured in the unloading area.

3. Remove the personnel barrier or the roof and roof cross-members from the ISO container.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.

4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

   Note: A receiving survey of the cask and transporter must be performed as soon as practicable after arrival at the site to assure compliance with 10 CFR 20, 10 CFR71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.

5. Remove the top and bottom impact limiters.

6. Remove the cask tie-down strap.

7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to keep the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.

8. Place the cask onto the dry loading station/stand. Disengage the lifting yoke and move clear.

9. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.

10. Remove the vent and drain valve port covers. Prior to reinstallation of the port covers, carefully inspect the O-ring seals and, if the O-rings show any damage, replace them with approved spares. Ensure that the replacement O-rings are properly installed and seated. Visually inspect the valved quick-disconnect nipples and replace them, if necessary.

    Note: For Alternate B port covers, replace the metallic O-ring with an approved spare prior to reinstallation.

11. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid and, if it shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-rings are properly installed and seated. Inspect the lid bolts and replace any that are damaged.

12. Visually inspect the inner cavity for foreign material or damage. Install or verify presence of a proper drain tube including drain tube alignment ring, as required.

13. Install the required dry transfer system components on the top of the cask.
14. Position the shielded transfer cask system components for fuel loading, as appropriate.
15. Identify the fuel to be loaded into each fuel basket module. Fuel elements loaded into each basket and/or module shall comply with the approved content conditions specified in Condition 5.(b)(1) and 5.(b)(2) of CoC No. 9225. Specific guidance on fuel selection, use of loading diagrams and preferential loading procedures is provided in Section 7.1.5. Perform an independent verification of the loading diagrams and fuel loading operations per Section 7.1.5.3. MTR plate canister loading shall be in accordance with Section 7.1.4.1 and ANSTO DFC loading shall be in accordance with Section 7.1.4.2.

    Note: If a basket module is to be loaded with a LEU MTR fuel element having ^{235}U content >470 g (>22 g ^{235}U per plate), cell black spacers, as shown on Drawing 315-40-085, shall be installed in basket module cell positions 1, 2 and 3 to prevent inadvertent loading of more than four LEU MTR fuel elements.

    Note: For the loading of HEU MTR fuel elements having ^{235}U content >380 g, a minimum of 2.0 cm of nonfuel hardware and /or spacer plates shall be provided at both ends of the fuel element to meet criticality control analysis requirements.

16. Load the shielded transfer cask and basket module with the selected fuel contents.
17. Place the shielded transfer cask containing a loaded fuel basket module onto the dry transfer system components positioned on the top of the cask.
18. Lower the loaded basket module from the transfer cask into the shipping cask.
19. Repeat the loading and transfer of loaded basket modules until the approved cask loading plan is completed.
20. Install the closure lid onto the cask using the dry transfer system. Visually verify that the lid is properly seated.
21. Remove the dry transfer system components from the top of the cask.
22. Install and tighten the 12 closure bolts to 260 ± 20 ft-lb in three passes, using the sequence stamped on the lid.
23. Connect a gas supply line to the vent valve and the drain line to the drain valve.
24. Open the air, nitrogen or helium gas supply valve and pressurize the cask cavity (< 30 psig) to force any residual water out the drain line. Continue to supply pressurized gas to the cask for a minimum of five minutes after the last residual free water discharges from the drain. Remove the drain and gas supply lines and attach a vacuum drying system (VDS) to the vent.
25. Evacuate the cask cavity to less than or equal to 10 torr (13 mbar) and continue vacuum pumping for a minimum of 15 minutes.
26. At the end of the vacuum pumping period, isolate the cask cavity from the vacuum pump and stop the vacuum pump. Monitor the cask cavity pressure for a minimum of 10 minutes. If the pressure rise is less than 5 torr (6.7 mbar), the cavity is verified as dry of

position prior to fuel element loading to ensure that the maximum number of three TRIGA fuel elements is not exceeded:

- TRIGA Stainless Steel (SS) LEU fuel elements having > 169 g ^{235}U < 275 g ^{235}U; or
- TRIGA SS HEU fuel elements having > 138 g ^{235}U < 175 g ^{235}U

The licensee's approved fuel loading plan shall ensure compliance with all fuel loading restrictions.

Each poisoned basket module may contain up to 28 TRIGA fuel elements for a total of 140 elements, or up to 112 TRIGA fuel cluster rods for a total of 560 rods per basket assembly.

Damaged TRIGA fuel elements and cluster rods and fuel debris are required to be loaded into sealed DFCs. The sealed DFCs are provided in two lengths. The short sealed DFC may be used in the base or top basket module. The long sealed DFC may be used in only the top module. The sealed DFCs are vacuum dried prior to loading into a TRIGA fuel basket (see sealed DFC loading procedure in Section 7.1.7).

There are two separate GA IFM FHU designs. One FHU is designed to hold research reactor fuel and the other is designed to hold High-Temperature Gas-Cooled Reactor fuel pellets. Each FHU consists of a sealed inner canister within a sealed outer canister. Each FHU contains irradiated fuel materials as described in Chapter 1. When loading the GA IFM FHUs, each individual sealed FHU will be loaded separately into a single GA IFM basket. This single basket containing two GA IFM FHUs and a spacer will comprise the entire cask load. Loading of the GA IFM basket into the NAC-LWT cask will utilize the TRIGA dry configuration loading procedure that is described in the following paragraphs.

TRIGA fuel elements that can be loaded into the cask are limited to a maximum decay heat of 7.5 watts per element, as discussed in Section 1.2.3. The decay heat load of the element must be calculated, and verified to be equal to or less than 7.5 watts per element prior to loading. TRIGA fuel cluster rods that can be loaded into the cask are limited to a maximum decay heat of 1.875 watts per element, as discussed in Section 1.2.3 (by reference to Table 5.1.1). The decay heat load of the fuel cluster rod must be calculated, and verified to be equal to or less than 1.875 watts per element prior to loading.

The procedure for loading the package with TRIGA fuel in a dry configuration is as follows:

1. Perform a receipt inspection of the empty cask and trailer/ISO container, inspecting for transport damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation

trunnions, release brakes and remove the chocks when required to complete uprighting operations. If an ISO container is used, it may be removed from the trailer and secured in the unloading area.

3. Remove the personnel barrier or the roof and roof cross-members from the ISO container.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.

4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

   Note: A receiving survey of the cask and transporter must be performed as soon as practicable after arrival at the site to assure compliance with 10 CFR 20, 10 CFR 71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.

5. Remove the top and bottom impact limiters.

6. Remove the cask tie-down strap.

7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to keep the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.

8. Place the cask onto the dry loading station. Disengage the lifting yoke and move clear.

9. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.

10. Remove the vent and drain valve port covers. Prior to reinstallation of the port covers, carefully inspect the O-rings and, if the O-rings show any damage, replace them with approved spares. Ensure that the replacement O-rings are properly installed and seated. Visually inspect the valve quick-disconnect nipples and replace them, if necessary.

    Note: For Alternate B port covers, replace the metallic O-ring with an approved spare prior to reinstallation.

11. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid and, if it shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-rings are properly installed and seated. Inspect the lid bolts and replace any that are damaged.

12. Visually inspect the inner cavity for foreign material or damage. Install, or verify the presence of the proper drain tube and drain alignment ring.

d. The acceptance criterion for the test is no measurable drop in pressure during the minimum test time. An acceptable test assures that the minimum assembly verification leakage test sensitivity is achieved.

Note: Alternate B port covers, if used, shall have a helium maintenance leakage rate test performed to confirm a leaktight containment closure. Install the Alternate B port cover and perform the maintenance leakage rate test per the requirements of Section 8.1.3.3.2.

34. Decontaminate the cask surfaces. Survey the cask for surface contamination and radiation dose rates.

    Note: Ensure compliance with 10 CFR 71.87(i) and 10 CFR 71.47.

35. Engage the cask lifting yoke to the lifting trunnions.

36. Lift the cask and position the cask rotation sockets in the rear rotation trunnions of the rear support structure. Carefully lower the cask to the horizontal transport orientation resting on the front saddle by moving the crane and/or the trailer as required to maintain cask engagement to the rear supports.

37. Disengage the lifting yoke from the lifting trunnions and remove it from the area. Install the cask tie-down strap. Install the top and bottom impact limiters. Install a TID to an attachment point on the top impact limiter.

38. Install ISO container bracing and lid, or personnel barrier.

39. Complete radiation and contamination surveys of the external surfaces of the package and record the data. Ensure removable contamination and radiation dose rate survey results comply with the limits specified in 10 CFR 71.87(i) and (j).

40. Measure the dose rate in millirems per hour at one meter from the package surface to determine the Transport Index (TI). Indicate the TI on the Radioactive Material labels applied to the package in accordance with 49 CFR 172, Subpart E.

41. Determine the appropriate Criticality Safety Index (CSI) assigned to the package contents in accordance with the CoC, and indicate the correct CSI on the Fissile Material label applied to the package per 49 CFR 172, Subpart E.

42. Apply appropriate placards to the transport vehicle in accordance with 49 CFR 172, Subpart F.

43. Complete the shipping documents and provide the carrier with instructions regarding the requirements for maintaining an exclusive use shipment.

7.1.7 Procedure for Loading TRIGA Damaged Fuel or Fuel Debris into a TRIGA Sealed Damaged Fuel Can (DFC)

1. Examine the sealed damaged fuel can (DFC) body and inspect for damage. Verify that the lid sealing surface is clean and free of defects. Visually verify that the drain plug seal is installed and the drain plug is partially threaded into the drain plug adapter to allow for draining.

2. Lower the DFC into the pool and position it for fuel loading.
3. Load the damaged TRIGA fuel cluster rods or fuel debris into the DFC. Verify that no more than the equivalent of 2 design base fuel elements, or 6 fuel cluster rods, as damaged fuel or fuel debris are loaded into the sealed DFC as specified in the CoC. Visually verify that there is no debris in the lid sealing surface and thread areas.
4. Examine the DFC lid and inspect for damage. Visually verify that the sealing surface is clean and free of defects. Lubricate the lid bolts, install the lid seal and verify that the lid valve is in the open position and the valve lock set screw is retracted.
5. Attach the testing hose to the lid test connection and ensure that the fitting is properly seated.
6. Install the lid and torque the lid bolts to 150 ± 10 inch-pound.

   Note: Torque any two diametrically opposed bolts first, then torque the remaining two bolts. Complete the torque sequence by verifying the torque of all four bolts in a clockwise direction.

7. Pressurize the sealed DFC with air or helium to 5-15 psig to remove the water. Continue the purge for at least 5 minutes after bubbles appear from the base of the DFC.
8. Access and torque the DFC drain plug to 50 ± 2 ft-lbs.
9. Evacuate the DFC to a pressure below 10 torr (13 mbar) and continue vacuum pumping for 10 minutes.
10. Stop and isolate the vacuum pump and monitor the DFC vacuum pressure for a minimum of 10 minutes. If the pressure rise is <5 torr (6.7 mbar) in 10 minutes, the DFC is verified as dry of free water. If the pressure rise is >5 torr (6.7 mbar) in 10 minutes or less, the DFC is not considered dry of free water. Repeat vacuum drying and pressure rise testing until the dryness verification results are satisfactory.
11. Backfill the DFC with helium to a pressure of 1 atmosphere (0 psig), +1, -0 psi.
12. Shut and lock the lid diaphragm valve. The DFC is now sealed, dried and backfilled.
13. Disconnect the testing hose from the lid test connection.
14. The sealed DFC is now ready for loading into a TRIGA basket module.

7.1.8 Procedure for Wet Loading of PWR/BWR Fuel Rods or TPBARs into the PWR/BWR Transport Canister

For the shipment of PWR and BWR fuel rods and nonfuel-bearing components (e.g., PWR guide tubes or BWR water rods), the PWR/BWR transport canister has three configurations: sealed canister, screened canister, and free-flow canister. All three canister configurations may be used to contain either intact or damaged fuel rods, or a combination of both damaged and intact fuel

NAC-LWT casks to be used to transport the TPBAR consolidation canisters shall be configured as shown on Drawing No. 315-40-128, including Alternate B port covers. NAC-LWT casks to be used to transport a PWR/BWR Rod Transport Canister shall be configured as shown on Drawing No. 315-40-104, Assembly 95, including Alternate B port covers.

1. Perform a receiving survey of the empty cask and inspect for damage. Verify, by cask serial number, that the cask is approved for TPBAR shipment.
2. Position a trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release brakes and remove the chocks when required to complete uprighting operations. If an ISO is used, it may be removed from the trailer and secured in the unloading area.
3. Remove the roof from the ISO container and open the front and rear ISO doors. Remove roof cross-members, if installed.

   Note: Verify that the package nameplate displays the package identification number, USA/9225/B(M)-96, as required by the CoC for TPBAR contents.
4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

   Note: A receiving survey of the cask and transporter must be performed as soon as practical after arrival at the site to assure compliance with 10 CFR 71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.
5. Remove the top and bottom impact limiters.
6. Remove the cask tie-down strap.
7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to keep the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.
8. Place the cask in the decontamination pit or other designated area. Disengage the lifting yoke. Clean cask surfaces of road dirt as required for entry into the spent fuel pool.
9. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.
10. Remove the Alternate B vent and drain valve port covers. Prior to reinstallation of the port covers, replace the metallic O-ring seal with an approved spare and inspect the Viton® O-ring seal for each port cover. If the Viton® O-ring shows any damage, replace it. Ensure that the replacement O-rings are properly installed and seated. Store the port covers to protect the seal surfaces. Visually inspect the valved quick-disconnect nipples and replace them, if necessary.

11. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid. If the O-ring shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-rings are properly installed and seated. Inspect the lid bolts and replace any that are damaged. Ensure that the TPBAR spacer is installed on the bottom of the cask lid for consolidation canister transports and not damaged when the lid is set down.
12. Visually inspect the inner cavity for foreign material or damage. Install or verify the presence of the standard drain tube and the TPBAR basket assembly (Drawing No. 315-40-10, Assembly 96 or Assembly 95) for loading of the consolidation canister; or the standard drain tube, TPBAR basket assembly (Drawing No. 315-40-10, Assembly 95), and the PWR Insert (Drawing No. 315-40-105, Assembly 99) for the loading of the PWR/BWR Rod Transport Canister containing TPBARs.

    Note: The PWR inset may be installed during the placement of the loaded PWR/BWR Rod Transport Canister into the NAC-LWT cask.
13 Fill the cask cavity with clean water. Install lift yoke arm guides and remote actuation components on the cask lifting yoke.
14. Engage the cask lifting yoke with the cask lifting trunnions and pick up the cask. Carefully lower the cask to the bottom of the cask loading area while spraying the cask down with clean water.
15. Disengage the lifting yoke from the cask and remove the yoke from the pool.
16. Identify the TPBAR consolidation canister or the PWR/BWR Rod Transport Canister containing TPBARs to be loaded.
17. Pick up the consolidation canister or the PWR/BWR Rod Transport Canister using the required grapple system.
18. Position the container over the cask and then carefully lower it into the cask to avoid damage to the cask sealing surfaces. Orient the consolidation canister bail so that it is aligned with the drain tube location. Confirm that the container is fully seated, then release and raise the grapple to the full up position.
19. Position the cask lifting yoke over the cask closure lid. Attach the slings to the closure lid and cask lifting yoke. Lower the yoke over the cask.
20. Position the closure lid over the cask and slowly lower it into place. For the consolidation canister, ensure the bail is properly aligned to the TPBAR spacer on the bottom of the lid. Use the cask and lid match marks as guides to properly align the lid. Visually confirm that the closure lid is seated.
21. Lower the cask handling yoke to slack the closure lid cables. Engage the lift yoke to the lifting trunnions and begin lifting.

    Note: Visually verify the yoke engagement before lifting the cask.

The NAC-LWT cask will be loaded dry, utilizing a transfer cask for loading each of the four basket modules. The basket modules will be preloaded with the PULSTAR fuel contents. The damaged fuel cans will be preloaded, closed, drained and dried, if applicable, prior to loading in either the top or base basket module. The PULSTAR cans shall be loaded and prepared for transport in accordance with the applicable steps of Section 7.1.7.

The NAC-LWT dry PULSTAR fuel loading and preparation for transport procedures are as follows.

1. Perform a receipt inspection of the empty cask and trailer/ISO container, inspecting for transport damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release brakes and remove the chocks when required to complete uprighting operations. If an ISO container is used, it may be removed from the trailer and secured in the unloading area.
3. Remove the lid/top of the ISO container and remove any bracing.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.
4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

   Note: A receiving survey of the cask and transporter must be performed as soon as practical after arrival at the site to assure compliance with 10 CFR 71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.
5. Remove the top and bottom impact limiters.
6. Remove the cask tie-down strap.
7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to keep the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.
8. Place the cask into the dry loading station.
9. Disengage the lift yoke.
10. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.
11. Remove the vent and drain port covers. Prior to reinstallation of the port covers, carefully inspect the port cover O-ring seals and, if the O-rings show any damage, replace them with approved spares. Ensure that the replacement O-rings are properly

installed and seated. Visually inspect the vent and drain quick-disconnect nipples and replace them, if necessary.

Note: For Alternate B port covers, replace the metallic O-ring with an approved spare prior to reinstallation.

12. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid. If the O-ring shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-rings are properly installed and seated. Inspect the lid bolts and replace any that are damaged.
13. Visually inspect the cask cavity for foreign material or damage. Clean as necessary. Install or verify the presence of a correct drain tube assembly including alignment ring.
14. Install the required dry transfer system components to the top of the cask.
15. Position the shielded transfer cask components for basket module loading, as appropriate.
16. Identify the PULSTAR fuel assemblies, fuel rod holders, and fuel cans to be loaded, and verify that the PULSTAR fuel contents comply with the authorized content, heat load and quantity conditions of the CoC. Four basket modules (e.g., one base module, two intermediate modules, and a top module) constitute the 28 MTR basket assembly. Spacers will be used as provided to position the PULSTAR fuel contents, as required.
17. Each module is capable of containing up to seven intact fuel assemblies, fuel rod inserts or a PULSTAR fuel can. Fuel cans are restricted to being loaded into the top and base modules, where the cans may be loaded with intact fuel assemblies or fuel rod holders without loading preference. There are no limitations on loading location for intact fuel assemblies or fuel rod holders in any of the four basket modules.

    The base module is loaded into the cask first, followed by the two intermediate modules and the top module is loaded last.
18. Load the shielded transfer cask with the loaded base basket module.
19. Place the shielded transfer cask containing the base module unit onto the dry transfer system components positioned on the top of the cask.
20. Lower the fuel basket from the transfer cask into the NAC-LWT cask cavity.
21. Repeat the loading and transfer of loaded basket modules until the approved cask loading plan is completed.
22. Install the closure lid onto the cask using the dry transfer system. Visually verify that the lid is properly seated.
23. Remove the dry transfer cask system components from the top of the cask.

greater than 10, the package must be shipped as "exclusive use" and meet the requirements of 10 CFR 71.47 (b), (c) and (d). If the dose rate and shipping requirements of 10 CFR 71.47 (b), (1), (2), (3) and (4) cannot be met, the package cannot be shipped.

Note: 10 CFR 71.47 (c) and (d) require the shipper to provide the carrier with written instructions for maintenance of the exclusive use shipment. The instructions must be included with the shipping paper information. The instructions must be sufficient so that, when followed, they cause the carrier to avoid actions that unnecessarily delay delivery or unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public.

- If the dose rate is > 10 mSv/h (1000 mrem/hr) at any point on the external surface of the cask, the cask exceeds the limits of 10 CFR 71.47 and cannot be shipped.

29. Complete the shipping document, carrier instructions (if required), and apply appropriate placards and labels.

7.1.12 Procedure for Wet Loading PWR MOX Fuel Rods in a Transport Canister Into the NAC-LWT Cask

PWR MOX fuel rods (or combinations of PWR MOX and UO_2 PWR fuel rods) are required to be loaded into a screened or free flow PWR/BWR Rod Transport Canister prior to loading into the NAC-LWT cask for transport. Although a maximum quantity of 16 MOX fuel rods may be shipped, it is required that the 5 × 5 rod insert be used to position the rods in the transport canister (i.e., the 4 × 4 insert is not authorized for use for the transport of MOX fuel rods).

In order to satisfy the increased potential for release of significant quantities of radioactive materials, and as recommended by NUREG-1617, Supplement 1, the NAC-LWT cask assembly specified for the transport of PWR MOX fuel rods contained in a transport canister provides a leaktight containment boundary.

The screened or free flow transport canister with a 5 × 5 rod insert will be loaded with up to 16 PWR MOX fuel rods (or a combination of up to 16 PWR MOX and UO_2 PWR fuel rods). In addition to the 16 PWR MOX fuel rods, up to 9 zirconium alloy-based burnable poison rods (BPRs) may be loaded into the unused insert openings.

NAC-LWT casks to be used for the transport of MOX fuel rods shall be configured as shown on Drawing No. 315-40-104, Assembly 97.

1. Perform a receiving survey of the empty cask and inspect for damage.
2. Position a trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist

that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release brakes and remove the chocks when required to complete uprighting operations. If an ISO is used, it may be removed from the trailer and secured in the unloading area.

3. Remove the roof from the ISO container and open the front and rear ISO doors. Remove roof cross-members, if installed.

   Note: Verify that the package nameplate displays the package identification number, USA/9225/B(U)F-96, as required by the CoC for PWR MOX fuel rods.

4. Perform a Health Physics survey of the cask and adjacent surfaces of the trailer.

   Note: A receiving survey of the cask and transporter must be performed as soon as practical after arrival at the site to assure compliance with 10 CFR 71.87(i) and 10 CFR 71.47, and to assure timely reporting of any reportable noncompliance.

5. Remove the top and bottom impact limiters.

6. Remove the cask tie-down strap.

7. Using the lifting yoke with the guides removed, engage the lifting trunnions. Raise the cask to vertical by rotating the cask rotation sockets on the rear cask supports, moving the crane and/or trailer as required to maintain the lift yoke engaged to the trunnions and the cask engaged in the rear supports. When the cask is fully vertical, lift the cask from the supports and remove it from the trailer/container.

8. Place the cask in the decontamination pit or other designated area. Disengage the lifting yoke. Clean cask surfaces of road dirt, as required, for entry into the spent fuel pool.

9. Visually inspect the neutron shield tank fill, drain and level inspection plugs for signs of neutron shield fluid leakage.

10. Remove the vent and drain valve port covers. Prior to reinstallation of the port covers, carefully inspect the valve port cover O-ring seals and, if the O-rings show any damage, replace them with approved spares. Ensure that the replacement O-rings are properly installed and seated. Visually inspect the valved quick-disconnect nipples and replace them, if necessary.

    Note: For Alternate B port covers, replace the metallic O-ring with an approved spare prior to reinstallation.

11. Remove closure lid bolts. Attach the lid lift slings to the closure lid. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to reinstallation of the lid, carefully inspect the Teflon O-ring seal in the underside of the closure lid. If the O-ring shows any damage, replace it. Remove the metallic O-ring and replace it with an approved spare. Ensure that the replacement O-ring(s) is properly installed and

Note: Alternate B port covers, if used, require the satisfactory completion of a helium maintenance leakage rate test to confirm a leaktight seal condition for each loaded transport. Install the Alternate B port cover and perform the maintenance leakage rate test per the requirements of Section 8.1.3.3.2.

36. Decontaminate the cask. Survey the cask for surface contamination and radiation dose rates.

    Note: Ensure compliance with 10 CFR 71.87(i) and 10 CFR 71.47.

37. Remove lift yoke arm guides. Engage the cask lifting yoke to the lifting trunnions.
38. Lift the cask and position the cask rotation sockets in the rear rotation trunnions of the rear support structure. Carefully lower the cask to the horizontal transport orientation resting on the front saddle by moving the crane and/or trailer, as required, to maintain cask engagement to the rear supports.
39. Disengage the cask lifting yoke from the cask lifting trunnions and remove it from the area.
40. Install the cask tie-down strap. Install the top and bottom impact limiters.
41. Install a TID to an attachment point on the top impact limiter.
42. Install roof cross-members, close ISO container doors, and replace ISO container roof.
43. Complete radiation and contamination surveys of the external surfaces of the package and record the data. Ensure removable contamination and radiation dose rate survey results comply with the limits specified in 10 CFR 71.87(i) and (j).
44 Measure the dose rate in millirems per hour at one meter from the package surface to determine the Transport Index (TI). Indicate the TI on the Radioactive Material labels applied to the package in accordance with 49 CFR 172, Subpart E.
45. Determine the appropriate Criticality Safety Index (CSI) assigned to the package contents in accordance with the CoC, and indicate the correct CSI on the Fissile Material label applied to the package per 49 CFR 172, Subpart E.
46. Apply appropriate placards to the transport vehicle in accordance with 49 CFR 172, Subpart F.
47. Complete the shipping documents and provide the carrier with instructions regarding the requirements for maintaining an exclusive use shipment.

7.1.13 Procedures for Dry Loading of MTR-28 Basket Modules Containing SLOWPOKE Fuel Canisters into the NAC-LWT Cask

This section presents the steps for dry loading, using a transfer cask, of the MTR-28 basket modules containing SLOWPOKE Fuel Canisters into the NAC-LWT cask. For transport, two MTR-28 basket modules, consisting of a top module and upper intermediate module, and two empty MTR-28 modules (lower intermediate and bottom modules) must be loaded into the NAC-LWT cask. Only the top and upper intermediate MTR-28 basket modules can each contain up to a maximum of four (4) SLOWPOKE Fuel Canisters. The three central fuel cells of these two modules are blocked with cell block spacers. Therefore, the maximum payload for a single

NAC-LWT cask is a maximum of eight SLOWPOKE Fuel Canisters. The two empty lower MTR-28 basket modules are used to ensure proper axial positioning of the complete basket assembly in the NAC-LWT cavity.

For the transport of SLOWPOKE Fuel Canisters, the NAC-LWT package shall be assembled for transport and identified as specified on NAC License Drawing 315-40-158.

The maximum decay heat load of a single SLOWPOKE Fuel Canister is 0.625 Watts and the maximum package decay heat load is 5 Watts.

The procedure for loading the NAC-LWT package with AECL SLOWPOKE Fuel Rods in a dry configuration is as follows:

1. Perform a receiving survey of the ISO and trailer, and inspect for damage. The cask user shall verify by reference to the NAC provided Certificate of Conformance(s) that the identified NAC-LWT cask and associated lift yoke are within the allowable annual maintenance period specified on the certificate(s) prior to loading and release for transport.
2. Position the trailer in the designated cask unloading area. Level the trailer. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release the brakes, and remove the chocks when required to complete the uprighting operations. Prior to cask removal, the ISO container may be removed from the trailer and secured in the unloading area, if required.
3. Licensees shall receive and survey the NAC-LWT cask for radiation and removable contamination (for both gross beta-gamma and alpha) per 10 CFR 20 and 49 CFR 173. Open the ISO container front and/or read doors and record the survey results. If radiation or contamination levels exceed the limits of 49 CFR 173.441 or 173.443, respectively, the user/licensee shall notify the shipper, NAC, and ensure the appropriate notifications are completed.
4. Undo tiedowns and remove the roof from the ISO container. Remove the ISO roof cross members, if installed.
5. Remove the top and bottom impact limiters; collect any TIDs that may be present.
6. Remove the cask tie-down strap. Complete the radiation and contamination surveys of the cask as additional surfaces become accessible. Clean the cask surfaces, as required.
7. Remove the Alt. vent valve port cover. Store the Alt port cover to protect the seal surfaces. Visually inspect the vent valve quick-disconnect nipple and replace if necessary. Prior to installation, inspect the Viton® O-ring seal on the Alt. port cover, and if the O-ring shows any damage, replace it.

8. Install the cask lifting yoke to a crane of sufficient capacity in accordance with the user facilities' heavy lifting program and engage the two lifting trunnions at the front end of the cask. Raise and rotate the cask to a vertical position on the rear cask supports, moving the crane and/or trailer, as required, to maintain the cask engaged in the rear cask supports. When the cask is vertical, lift the cask from the ISO container.
9. Move and place the cask on a base plate, if required. Connect the base plate to the cask's attachment points using chains and take up slack with the tensioners. Disengage the lifting yoke.
10. Loosen and remove all closure lid bolts. Prior to installation, inspect the lid bolts and replace any that are damaged.
11. Remove the closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of the closure lid. Prior to installation, carefully inspect the Teflon O-ring seal in the underside of the closure lid. If the O-ring shows any damage, replace it. Remove the metallic O-ring from the groove and discard. Clean and visually inspect the groove and lid recess seating surfaces for cleanliness, damage, or degradation. If the groove and lid recess seating surfaces are acceptable, install a new metallic O-ring with an approved spare. Ensure the replacement O-rings are properly installed and seated.
12. Visually inspect the inner cavity for foreign material, free water, or damage. Note deficiencies and correct as required. Remove any shipping dunnage as necessary. Clean all accessible surfaces to include lid sealing surface. Install, or verify the presence of the drain tube and drain alignment ring.
13. Install the LWT internal shield ring.
14. Lift and install the Dry Transfer System (DTS) transfer cask adapter onto the cask. Attach the four retention clamps around the LWT lift trunnions.
15. Verify the proper installation, or install, the empty bottom and lower intermediate MTR-28 basket modules.
16. Identify the top and upper intermediate MTR-28 basket modules to be loaded with SLOWPOKE Fuel Canisters. The mandatory basket module loading sequence is as follows: load or verify installed empty bottom and lower intermediate MTR-28 basket modules; load upper intermediate basket module containing up to four (4) SLOWPOKE Fuel Canisters; and, finally, load the top basket module containing up to four (4) SLOWPOKE Fuel Canisters. The top and upper intermediate MTR-28 basket modules shall each have three (3) cell block spacers installed in the three central fuel cells to prevent inadvertent fuel canister loading. All transports shall consist of all four MTR-28 basket modules assembled in accordance with the mandatory loading sequence.

17. For the initial SLOWPOKE Fuel Canister basket module loading, place the upper intermediate basket module in the Intermediate Transfer System (ITS) inner shield.
18. Move the ITS inner shield into position in the hot cell for the transfer of the loaded SLOWPOKE Fuel Canisters (loaded in accordance with the procedures of Section 7.1.14).
19. Lift the SLOWPOKE Fuel Canister using the handle and lower the Fuel Canister into one of the open (unblocked) fuel cells of the MTR-28 basket module in the ITS inner shield. Disengage the Fuel Canister handling tool. Repeat as required to load up to four (4) SLOWPOKE Fuel Canisters into the basket module.
20. Install the inner shield lid.
21. Move the ITS inner shield assembly containing the loaded MTR-28 basket module to the pre-staged transfer system location.
22. Lift the inner shield assembly containing the loaded MTR-28 basket module and place it through the ITS shield assembly adapter and into the outer shield of the ITS.
23. Disengage the inner shield lid. Lift and remove the inner shield lid through the shield assembly adapter and close the shield assembly adapter gate.
24. Place the DTS transfer cask onto the ITS shield assembly adapter.
25. Open the DTS transfer cask gate.
26. Open the ITS shield assembly adapter gate.
27. Lower the transfer cask grapple into the ITS and engage the MTR-28 basket module.
28. Retract grapple and loaded MTR-28 basket module into the transfer cask.
29. Close the DTS transfer cask shield gate.
30. Lift the DTS transfer cask and place it on the cask adapter assembly positioned on top of the NAC-LWT cask.
31. Open the cask adapter shield gate.
32. Open the DTS transfer shield cask gate and lower the loaded MTR-28 basket module into the NAC-LWT cask cavity.
33. Disengage grapple and retract back into the transfer cask.

    Note: Grapple release can be verified by checking cable for tension.
34. Verify grapple is fully retracted.

    Note: Indication will be physical indicator attached to cable.
35. Close cask adapter shield gate.

36. Repeat steps 17-35 for the top MTR-28 basket module.
37. Perform an independent verification that the loaded AECL fuel rod contents loaded are in full compliance with the NAC-LWT CoC content conditions.
38. Install shield plug and remove shield ring/plug assembly through the cask adapter.
39. Carefully lower the closure lid into position through the cask adapter and visually verify that it is properly seated.
40. Inspect and install lid bolts hand tight.
41. Remove four retention clamps from the cask trunnions and carefully remove transfer cask adapter and position for subsequent decontamination.
42. Tighten all 12 closure lid bolts to 260 ± 20 ft-lbs in three passes using the torque sequence indicated on the closure lid.

    Note: If water was introduced to cask cavity during dry loading operations (due to weather conditions, i.e. snow rain, etc), the NAC LWT cask may be "blown-down" using compressed air or gas in the vertical orientation.

    Note: At the option of the user, the NAC-LWT cask can be placed in a horizontal position in the ISO at this point in the procedure in accordance with Step 49.
43. Connect a vacuum pump to the cask vent valve and evacuate the cask cavity to less than or equal to 10 torr (13 mbar) and continue vacuum pumping for a minimum of 15 minutes.
44. At the end of the vacuum pumping period, isolate the cask cavity from the vacuum pump and stop the vacuum pump. Monitor the cask cavity pressure for a minimum of 10 minutes. If the pressure rise is less than 5 torr (6.7 mbar), the cavity is verified as dry of free water. If the pressure rise is greater than 5 torr (6.7 mbar), repeat the vacuum drying until the dryness verification results are satisfactory.
45. Backfill the cask cavity with helium to 0 psig (1 atmosphere, absolute), +1, -0 psi and disconnect the VDS from the vent valve.
46. Perform a helium leakage rate test of the closure lid containment O-ring using a Helium Mass Spectrometer Leak Detector in accordance with the procedural requirements of Section 8.1.3.1, Steps 3 through 10.
47. Install Alt. port cover in the vent port and torque each port cover bolt to 100, +10,-0 in-lbs.
48. Survey the cask surface for gross beta-gamma and tritium removable contamination levels, and radiation dose rates. Decontaminate the cask, if required.

    Note: Removable contamination levels and radiation levels shall comply with 49 CFR 173.443 and 173.441, respectively.

49. Using the cask lifting yoke, lift, and position the cask in the rear cask supports on the ISO/trailer. Engage the trunnion pockets in the bottom end of the cask with the rotation trunnions. Lower the cask to rest on the front tie-down saddle, moving the crane, and/or trailer, as required. Disengage the cask lifting yoke from the cask lifting trunnions and set it aside.

50. Install and attach the cask tie-down strap. Install the cask top and bottom impact limiters.

51. Install a tamper-indicating seal to one of the top impact limiter ball lock pins.

52. Install roof cross-members, if used; replace ISO container roof and close ISO container doors.

53. Complete a Health Physics survey on the external surface of the package and record the results. Complete dose rate measurements at the cask surface, at 1 meter from the cask surface, and at 2 meters from the vertical plane of the side of the transport vehicle. The maximum dose rate at 1 meter from the cask is the transport index (TI). Ensure compliance with 10 CFR 71.87(i) and observe the following criteria.

    - If the dose rate is less than 2 mSv/h (200 mrem/hr) at all accessible points on the external surface of the cask, and the TI is less than 10, the package meets the requirements of 10 CFR 71.47 (a).

    - If the dose rate is greater than 2 mSv/h (200 mrem/hr), but is less than 10 mSv/h (1000 mrem/hr) at any point on the external surface of the package, or the TI is greater than 10, the package must be shipped as "exclusive use" and meet the requirements of 10 CFR 71.47 (b), (c) and (d). If the dose rate and shipping requirements of 10 CFR 71.47 (b), (1), (2), (3) and (4) cannot be met, the package cannot be shipped.

      Note: 10 CFR 71.47 (c) and (d) require the shipper to provide the carrier with written instructions for maintenance of the exclusive use shipment. The instructions must be included with the shipping paper information. The instructions must be sufficient so that, when followed, they cause the carrier to avoid actions that unnecessarily delay delivery or unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public.

    - If the dose rate is > 10 mSv/h (1000 mrem/hr) at any point on the external surface of the cask, the cask exceeds the limits of 10 CFR 71.47 and cannot be shipped.

54. Complete the shipping document, carrier instructions (if required), and apply appropriate placards and labels.

7.1.14 Procedure for Loading AECL SLOWPOKE Fuel Rod Contents Into the SLOWPOKE Fuel Canister

The following general procedures provide guidance for the loading of AECL SLOWPOKE fuel rod contents into individual fuel can inserts, which are then subsequently placed into a SLOWPOKE Fuel Canister. The Fuel Canister is subsequently loaded into a MTR-28 upper intermediate or top basket module for dry transferred into the NAC-LWT cask using the Dry Transfer System (DTS).

The SLOWPOKE Fuel Canister includes a welded fuel canister body into which four (4) 5 x 5 inserts (assembled of 0.40 inch nominal internal diameter insert tubes for intact SLOWPOKE Fuel Rods) and/or four (4) 4 x 4 inserts (assembled of 0.53 inch nominal internal diameter insert tubes for damaged SLOWPOKE Fuel Rods) are stacked to allow for the placement of up to 100 fuel rods in each SLOWPOKE Fuel Canister. The Fuel Canister is closed by a lockable, spring-loaded lid assembly, which incorporates a lid handle for loaded Fuel Canister handling. The lid assembly incorporates two lid latch bolts with lock washers and torque to 30 ± 5 in-lbs, which prevent inadvertent lid removal during shipment and handling. The SLOWPOKE Fuel Canister is provided with an aluminum bottom screened opening and two upper side aluminum screened openings to allow for the self-draining of the Fuel Canister if stored in water at the receiving facility prior to final processing. Each of the insert tubes is notched at the base of the tube to facilitate draining of each insert tube through the bottom screened opening. The screened openings and tight fitting lid retains fuel debris and minimizes the potential for release of fuel debris from the SLOWPOKE Fuel Canister to the NAC-LWT internal cavity.

The SLOWPOKE Fuel Canisters are visually inspected, load tested, and the welds examined following fabrication prior to acceptance for use. The AECL SLOWPOKE fuel rod contents shall be verified as meeting the quantity, decay heat and fissile content limits of the NRC Certificate of Compliance (CoC) prior to loading. The radioactive materials to be loaded in each SLOWPOKE Fuel Canister shall be identified and recorded as part of the packaging manifest for the cask shipment. Independent confirmation of the identification and location of the radioactive materials shall be made during the loading operations.

The procedure for loading AECL SLOWPOKE fuel rod contents into the Fuel Canister is as follows:

1. Verify the specific AECL SLOWPOKE Fuel Rod contents to be loaded into the 5 x 5 or 4 x 4 canister insert meet the content condition limits of the CoC for quantity, maximum mass, maximum decay heat, maximum fissile content and waste form. Damaged fuel rods shall be placed in 4 x 4 rod insert assemblies, as required.

2. Verify the SLOWPOKE Fuel Canister and insert assemblies comply with the requirements of NAC Drawing 315-40-156.
3. Visually inspect the Fuel Canister, Lid and rod insert assemblies and verify the components condition do not show signs of damage—e.g., bulging or buckling, breaching, and does not have rips, tears, holes or pointed dents that could affect packaging or transport operations. Record the SLOWPOKE Fuel Canister serial number and the results of the visual inspection on the Cask Loading Report.
4. Position the appropriate 5 x 5 or 4 x 4 insert assembly in the hot cell.
5. Individually load the AECL SLOWPOKE fuel rods into the designated insert assembly.
6. After completion of loading the designated fuel rods, lift and place the loaded fuel rod insert into the SLOWPOKE Fuel Canister.
7. Repeat Steps 5 through 7 until a total of four (4) fuel rod insert assemblies are loaded and positioned in the SLOWPOKE Fuel Canister.
8. With the spring plunger in the unlocked position, insert the self-locking lid assembly into the top of the Fuel Canister. Torque the two lid latch bolts and lock washers to 30 ± 5 inch-pounds.
9. Lift the filled SLOWPOKE Fuel Canister and place it into the MTR-28 upper intermediate or top basket module per the procedures in 7.1.13.

18. Carefully lower the cask to rest on the bottom of the cask unloading area while spraying the exterior surfaces of the cask with clean water to minimize contamination.
19. Disengage the lifting yoke from the cask and slowly raise the yoke until the closure lid is raised clear of the cask. Remove the yoke from the vicinity of the cask to provide clearance for unloading the cask.

    Note: Closure lid may be brought out of the pool and later assembled to the empty cask.
20. Unload the contents of the cask cavity using the required grapple system.
21. Position the cask lifting yoke with the cask closure lid over the cask cavity and slowly lower it into place using the cask and closure lid match marks as guides. Visually confirm that the closure lid is seated.
22. Engage the cask lifting yoke with the cask trunnions and raise the cask.
23. Raise the cask until the lid is slightly above the surface of the pool. At the option of the licensee/user, several of the closure lid bolts (i.e., 4-12) may be installed hand tight.
24. Raise the cask clear of the pool, rinsing the yoke and cask with clean water.
25. Transfer the cask to the decontamination pit or other work area. Remove the yoke and lid lift slings.
26. Install and tighten the 12 closure lid bolts to 260 ± 20 ft-lb in three passes, using the torque sequence stamped on the closure lid.
27. At the option of the licensee/user, a 25 to 50 gallon clean water flush of the cask cavity may be performed by connecting a valved, clean water line to the drain valve and a valved drain line to the vent valve. After the cavity flushing is completed, if performed, disconnect the water supply and drain lines.
28. Connect a gas (air, nitrogen or helium) supply line to the vent valve and the drain line to the drain valve.
29. Open the gas supply valve and pressurize the cask cavity (<30 psig) to force out the water. Continue to supply gas to the cask cavity for a minimum of five minutes after the last residual free water discharges from the drain line.
30. Remove the gas supply and drain lines.
31. Install the alternate port covers over the vent and drain valves and tighten the port cover bolts to 100 ± 10 in-lb. For Alternate B port covers, install and torque the high-strength bolts to 285 ± 15 inch-pound.

    Note: It is not necessary to inspect or replace the port cover seals. Seal inspection and replacement, if required, will be performed prior to the next loaded transport.

7.2.3 Procedure for Wet Unloading of MTR, TRIGA, DIDO, ANSTO, PULSTAR, or SLOWPOKE Fuel Basket Contents

The procedure for the unloading of MTR, TRIGA, DIDO, ANSTO, ANSTO-DIDO, PULSTAR, or SLOWPOKE fuel basket contents from the package in a spent fuel pool is as follows:

1. Perform a receiving survey of the cask and inspect for transport damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release the brakes and remove the chocks when required to complete the uprighting operations. If an ISO container is used, it may be removed from the trailer and secured in the loading area.
3. Remove the roof from the ISO container, and open the front and rear ISO doors. Remove roof cross-members, if installed.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.
4. Licensees shall monitor the package for radioactive contamination and radiation levels in accordance with 10 CFR 20.1906. If contamination levels exceed 10 CFR 71.87(i) or radiation levels exceed the limits of 10 CFR 71.47, the licensee shall notify the NRC Operations Center.
5. Verify the TID identification number on the top impact limiter to confirm tampering with the package did not occur.
6. Remove the top and bottom impact limiters.
7. Remove the cask tie-down strap.
8. Using the cask lifting yoke with left yoke arm guides removed, engage the lifting trunnions of the front end of the cask. Raise the cask to a vertical position on the rear cask support, moving the crane as necessary to keep the cask engaged in the rear rotation supports and the crane cable vertical. When the cask is vertical, lift the cask from the container supports.
9. Place the cask in the decontamination pit or other site designated area. Disengage the lifting yoke. Clean cask surfaces of road dirt as required for entry into the spent fuel pool.
10. Remove the vent valve and drain valve port covers. Connect a pressure gauge and isolation valve assembly to the cask vent valve. Open the isolation valve and record the internal pressure reading (if any). Using a suitable air line and the gauge/valve assembly, vent the cask cavity to an off-gas handling unit.
11. Connect vent and clean water fill lines to the vent and drain valves.
12. Open the water supply valve to allow water to slowly enter the cask cavity.

Note: Gases or steam exiting the vent may be radioactive. The vent line should be routed to an off-gas process system or a HEPA filter. The system for cooling down the package shall contain a pressure relief device set to ensure that the cask internal pressure is maintained below 100 psig. Coolant flow rates are to be controlled to avoid thermal shock to the fuel contents.

13. Continue the filling procedure until the cask cavity is filled with water. Remove fill and vent lines.

14. Loosen and remove the 12 closure lid bolts. At the option of licensee/user, some bolts (i.e., 4-12) may be left installed hand tight for the cask movement to the spent fuel pool.

15. Engage the cask lifting yoke (with slings, yoke arm guides and remote actuation system components attached) with the cask lifting trunnions and connect the closure lid to the lifting yoke slings.

16. Position the cask over the spent fuel storage pool and lower the cask until the top of the cask is at an elevation which allows access for the removal of the closure lid bolts.

17. Remove any remaining closure lid bolts, inspect and store.

18. Carefully lower the cask to rest on the bottom of the cask unloading area while spraying the exterior surfaces of the cask with clean water to minimize contamination. Disengage the lifting yoke from the lifting trunnions and slowly raise the yoke until the closure lid is raised clear of the cask. Remove the yoke from the vicinity of the cask to provide for clearance for unloading the cask.

    Note: The closure lid may be brought out of the pool and later assembled to the empty cask.

19. Unload the MTR, TRIGA, DIDO, spiral, MOATA plate, PULSTAR, or SLOWPOKE fuel assemblies, plate canisters, or fuel canisters from the top basket module using the appropriate grapple or handling system. As required, remove empty basket modules from the cask cavity to allow access to the next basket module. Continue fuel unloading operations until all fuel assemblies, plate canisters, fuel canisters and empty basket modules are removed from the cavity. Alternatively, each basket module containing fuel assemblies, plate canisters or fuel canisters may be unloaded from the cask cavity and stored in the spent fuel pool. Continue unloading until all basket modules have been removed.

20. Position the cask lifting yoke with guide arms and remote actuation components installed over the cask closure lid. Attach the slings to the cask closure lid and cask lifting yoke.

21. Position the cask lifting yoke and closure lid over the cask cavity and slowly lower it into place using the cask and closure lid match marks as guides. Visually confirm that the closure lid is seated.

    Note: The closure lid may be installed separately after the empty cask is removed from the spent fuel pool.

22. Engage the cask lifting yoke with the cask trunnions and raise the cask.
23. Raise the cask until the lid is slightly above the surface of the pool. At the option of the licensee/user, several of the closure lid bolts (i.e., 4-12) may be installed hand tight.
24. Raise the cask clear of the pool, rinsing the yoke and cask with clean water.
25. Transfer the cask to the decontamination pit or other work area. Remove the yoke and lid lift slings.
26. Install and tighten four closure lid bolts to 100 ± 10 ft-lb using the torque sequence stamped on the closure lid.
27. At the option of the licensee/user, a 25 to 50 gallon clean water flush of the cask cavity may be performed by connecting a valved, clean water line to the drain valve and a valved drain line to the vent valve. After the cavity flushing is completed, if performed, disconnect the water supply and drain lines.
28. Connect a gas (air, nitrogen or helium) supply line to the vent valve and the drain line to the drain valve.
29. Open the gas supply valve and pressurize the cask cavity (<30 psig) to force out the water. Continue to supply gas to the cask cavity for a minimum of five minutes after the last residual free water discharges from the drain line.
30. Remove the gas supply and water drain lines.
31. Remove the four closure lid bolts and lift the lid clear of the cask.

    Note: It is not necessary to inspect or replace the closure lid metallic seal. A new metallic seal will be installed and tested prior to the next loaded transport.

32. Remove the drain tube assembly and drain tube alignment ring from the cask cavity.
33. Reinstall the closure lid and install the 12 closure lid bolts. Torque the bolts to 260 ± 20 ft-lbs in three passes using the torque sequence indicated in the closure lid.
34. Install the alternate port covers over the vent and drain valves and tighten the port cover bolts to 100 ± 10 in-lb. For Alternate B port covers, install and torque the high-strength bolts to 285 ± 15 inch-pound.

Note: It is not necessary to inspect or replace the port cover seals. Seal inspection and replacement, if required, will be performed prior to the next loaded transport.

7.2.4 Procedure for Dry Unloading of MTR, TRIGA, DIDO, ANSTO, PULSTAR, or SLOWPOKE Fuel Contents

This section describes the procedure for unloading of MTR, TRIGA, DIDO, ANSTO, ANSTO-DIDO, PULSTAR, or SLOWPOKE fuel basket contents from the NAC-LWT in a cell or a dry unloading fixture.

1. Perform a receiving survey of the cask and inspect for transport damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release the brakes and remove the chocks when required to complete the uprighting operations. If an ISO container is used, the ISO container may be removed from the trailer and secured in the unloading area.
3. Remove the roof from the ISO container and open the front and rear ISO doors. Remove roof cross members, if installed.

   Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.
4. Licensees shall monitor the package for radioactive contamination and radiation levels in accordance with 10 CFR 20.1906. If contamination levels exceed 10 CFR 71.87(i) or radiation levels exceed the limits of 10 CFR 71.47, the licensee shall notify the NRC Operations Center.
5. Verify the TID identification number on the top impact limiter to confirm tampering with the package did not occur.
6. Remove the top and bottom impact limiters.
7. Remove the cask tie-down strap. Clean the cask surfaces as required for entry into the hot cell.
8. Using the cask lifting yoke with lift yoke arm guides removed, engage the lifting trunnions of the front end of the cask. Raise the cask to a vertical position on the rear cask support, moving the crane and/or trailer, as required, to keep the cask engaged in the rear rotation supports and the crane cable vertical. When the cask is vertical, block the trailer wheels and lift the cask from the container.
9. Place the cask in the cell transfer cart or unloading fixture. Disengage the lifting yoke.
10. Remove the vent valve port cover.
11. Connect vent line to the vent valve.

Note: The hot gases exiting from the vent may be highly radioactive and the exhaust gas should be routed to an off-gas process system or to a HEPA filter.

12. Allow the cask to vent. Remove vent line.
13. Loosen and remove the 12 closure lid bolts. Visually inspect and store the bolts.
14. Attach the lid removal fixture.
15. Using the hot cell transfer cart or unloading fixture, move the cask into the unloading position.
16. Remove the cask lid.

    Note: It is not necessary to inspect or replace the closure lid metallic seal. A new metallic seal will be installed and tested prior to the next loaded shipment.

17. Install the seal surface protector in the lid cavity, if required.
18. Unload the MTR, TRIGA, DIDO, ANSTO, PULSTAR, or SLOWPOKE fuel basket modules from the cask cavity using the required grapple or handling system.
19. Remove the cask seal surface protector, if installed, and replace the cask lid.
20. Using the cell transfer cart or unloading fixture, remove the cask.
21. Remove the lid from the cask and remove the drain tube and drain tube alignment ring.
22. Replace the cask lid and remove the lid removal fixture.
23. Install and tighten all 12 closure lid bolts to 260 ± 20 ft-lbs in three passes using the torque sequence indicated on the closure lid.
24. Install the port covers over the vent and drain valves and tighten the port cover bolts to 100 ±10 inch-pounds. For Alternate B port covers, install and torque the high-strength bolts to 285 ±10 inch-pounds.

    Note: It is not necessary to inspect or replace the port cover seals. Seal inspection replacement and leak testing will be performed prior to the next loaded transport.

7.2.5 Procedure for Dry Unloading of TPBAR Contents

This section describes the procedure for the unloading of a consolidation canister, a PWR/BWR Rod Transport Canister or waste container that contains TPBARs from the NAC-LWT in a dry unloading facility.

1. Perform a receiving survey of the ISO container and trailer, and inspect for damage.
2. Position the trailer in the designated cask unloading area. Set the trailer brakes and chock the wheels to prevent unintended movement. If site-specific conditions exist that require the trailer to move to allow the cask to be uprighted on its rotation trunnions, release the brakes and remove the chocks when required to complete the