February 2015

NAC-LWT

Legal Weight Truck Cask System

LWT-15A SLOWPOKE Fuel Core Initial Submittal

NON-PROPRIETARY VERSION

Docket No. 71-9225

NAC INTERNATIONAL

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Enclosure 1

No. 71-9225 for NAC-LWT Cask

List of SAR Changes

NAC-LWT SAR, Revision LWT-15A

SLOWPOKE Fuel Core Amendment

List of SAR Changes, NAC-LWT SAR, Revision LWT-15A

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, added the last bullet and modified the punctuation and text ("and") at the end of the previous two bullets in the list.
- Page 1-2, added "SLOWPOKE Fuel Core" and the associated CSI to the last row to the bottom of table in the "Approved Contents" section.
- Pages 1-3 thru 1-4, text flow changes.
- Page 1-5, added the sixth bullet on the page in Table 1.1-1 in the "Contents (payload)" section.
- Pages 1-6 thru 1-7, text flow changes.
- Page 1-8, modified the "Damaged SLOWPOKE" term and added the term and definition for ""Undamaged SLOWPOKE Fuel Rods in the SLOWPOKE Fuel Core" to Table 1.1-1.
- Page 1.1-1, deleted the end of a sentence near the end of the first paragraph in Section 1.1
- Page 1.1-2, added second to last (9th) bullet, with an editorial change to the end of the previous bullet deleting, "or," from the end.
- Page 1.1-3, added SLOWPOKE fuel core paragraph (fourth on page).
- Page 1.1-4, text flow change.
- Page 1.2-5, added SLOWPOKE fuel core text to end of item 4 in Section 1.2.3.
- Page 1.2-7, added item 24 near the top of the page.
- Page 1.2-8 thru 1.2-13, text flow change.
- Page 1.2-18, modified heading 1.2.3.12 to, "SLOWPOKE Fuel Rods in a SLOWPOKE Canister."
- Page 1.2-20, added Section 1.2.3.15, "SLOWPOKE Fuel Core."
- Page 1.2-60, added Table 1.2-17.

List of SAR Changes, NAC-LWT SAR, Revision LWT-15A

Chapter 2

- Pages 2.2.1-2 thru 2.2.1-3, added two new rows for SLOWPOKE Fuel Core data, and a "continued" page to Table 2.2.1-1. Note: adding the new rows caused a text flow shift of the last two rows that had previously been on page 2.2.1-2.
- Page 2.2.1-4, text flow changes.
- Page 2.2.1-5, added two new rows for SLOWPOKE Fuel Core data near the end of Table 2.2.1-2, including a new footnote.
- Page 2.6.12-137, text flow change due to insertion of Section 2.6.12.16 that follows.
- Pages 2.6.12-138 thru 2.6.12-140, added Section 2.6.12.16, "SLOWPOKE Fuel Core Basket."
- Page 2.6.12-140, renumbered previous Section 2.3.12.16 to 2.6.12.17, "Conclusion;" modified text near the end of the paragraph.
- Pages 2.7.7-103 thru 2.7.7-104, added Section 2.7.7.18, "SLOWPOKE Fuel Core Basket for the Accident Conditions of Transport."
- Pages 2.9-21 thru 2.9-24, added new Section 2.9.5, "SLOWPOKE Fuel Core Assembly."

Chapter 3

- Page 3.1-2, added the last two sentences to end of the partial paragraph at the top of the page, and "in the fuel canisters" near the end of the previous sentence for SLOWPOKE information.
- Page 3.1-3, text flow change.
- Pages 3.4-42 thru 3.4-43, added Section 3.4.1.21, "Thermal Evaluation of SLOWPOKE Fuel Core."
- Pages 3.4-44 thru 3.4-85, text flow changes.
- Page 3.4-86 added Figure 3.4-23, "Finite Element Model for SLOWPOKE Fuel Core Basket."
- Pages 3.4-87 thru 3.4-108, text flow changes.
- Page 3.4-109, added Table 3.4-28, "Maximum Component Temperatures – SLOWPOKE Fuel Core," and its associated footnote.
- Page 3.5-16, added Section 3.5.3.19, "Evaluation of SLOWPOKE Fuel Core."
- Pages 3.5-17 thru 3.5-22, text flow changes.

Chapter 4

- Page 4.2-4, modified the last row of Table 4.2-1 and Note 8.

List of SAR Changes, NAC-LWT SAR, Revision LWT-15A

Chapter 5

- Pages 5-3 thru 5-4, added the second to last paragraph on page 5-3 and the last paragraph on page 5-4 in Section 5, to include SLOWPOKE fuel elements and core information.
- Pages 5.1.1-1 thru 5.1.1-2, modified the last two bullets on page 5.1.1-1 by adding a semi-colon to end of second to last bullet, and deleting "or" from the last bullet, and added two additional bullets at the top of page 5.1.1-2.
- Pages 5.1.1-3 thru 5.1.1-5, text flow changes.
- Pages 5.1.1-6 thru 5.1.1-7, inserted the fourth and fifth paragraphs on the page, to include SLOWPOKE information; text flow also occurs on page 5.1.1-7.
- Pages 5.1.1-12 thru 5.1.1-13, added new information for SLOWPOKE Fuel Rods and Fuel Core to the end of Table 5.1.1-1.
- Pages 5.1.1-14 thru 5.1.1-16, text flow changes.
- Page 5.1.1-17, added continuing information to Table 5.1.1-2 for SLOWPOKE Fuel Rods and Fuel Core.
- Page 5.1.1-18, added last two lines to the bottom of Table 5.1.1-3 for SLOWPOKE Rods and Fuel Core.
- Pages 5.1.1-19 thru 5.1.1-21, text flow.
- Pages 5.3.23-1 thru 5.3.23-49, added new Section 5.3.23, "SLOWPOKE Core Configuration."

Chapter 6

- Page 6-1, added "or one SLOWPOKE fuel core" and editorial changes in the second to last sentence of the first paragraph.
- Page 6-2, added the last row to the "Criticality Safety Index" table.
- Page 6.1-6, added the last paragraph summarizing the SLOWPOKE fuel core analyses.
- Page 6.2-1, added ", or one SLOWPOKE fuel core" to the end of the first sentence in Section 6.2.
- Page 6.7.5-1 thru 6.7.5-16, added new Section 6.7.5, "SLOWPOKE Fuel Core."

Chapter 6 Appendices

- Pages 6.6.19-1 thru 6.6.19-3, added Section 6.6.19, "SLOWPOKE Fuel Core MCNP Input."

List of SAR Changes, NAC-LWT SAR, Revision LWT-15A

Chapter 7

- Pages 7.1-77 thru 7.1-82, added Section 7.1.18, Procedure for the Dry Loading of AECL SLOWPOKE Fuel Core into the NAC-LWT Cask."

Chapter 8

- No changes.

Chapter 9

- No changes.

Enclosure 2

No. 71-9225 for NAC-LWT Cask

List of Drawing Changes

NAC-LWT SAR, Revision LWT-15A

SLOWPOKE Fuel Core Amendment

List of Drawing Changes, NAC-LWT SAR, Revision LWT-15A

Drawing 315-40-185, Revision 0

1. Initial issuance.

Drawing 315-40-186, Revision 1

1. Initial issuance.

Drawing 315-40-187, Revision 0

1. Initial issuance.

Enclosure 3

No. 71-9225 for NAC-LWT Cask

List of Calculations

NAC-LWT SAR, Revision LWT-15A

SLOWPOKE Fuel Core Amendment

List of Calculations, NAC-LWT SAR, Revision LWT-15A

Enclosure 4 Contents:

1. Calculation 50026-2001, Revision 0, "Structural Evaluation of the SLOWPOKE Fuel Core Basket"
2. Shielding Calculation Data Input/Output Files, Disk 1 of 1
3. Criticality Calculation Data Input/Output Files, Disk 1 of 1

NAC Calculation 50026-2001, Rev. 0
Withheld In Its Entirety Per 10 CFR 2.390

Enclosure 4

No. 71-9225 for NAC-LWT Cask

Proposed Changes for Revision 61 of Certificate of Compliance

NAC-LWT SAR, Revision LWT-15A

SLOWPOKE Fuel Core Amendment

Drawings (new)

CoC Page 4 of 32:

LWT 315-40-185, Rev. 0	LWT Transport Cask Assembly, SLOWPOKE Contents
LWT 315-40-186, Rev. 1	Fuel Core Basket Assembly, SLOWPOKE
LWT 315-40-187, Rev. 0	Basket LID Assembly, SLOWPOKE

CoC Sections (new)

CoC Page 19 of 31:

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

(xxi) SLOWPOKE Fuel Core as specified below:

Parameter	Liquid HEU
Maximum Cask Heat Load (W)	45
Payload Limit (lb)	15
Maximum Number of Rods per Core	298
Maximum Initial ^{235}U per rod (g)	2.83
Maximum Initial Enrichment (wt% ^{235}U)	95.3
Maximum Initial ^{235}U per core (g)	837
Minimum Initial Enrichment (wt% ^{235}U)	90
Minimum Cool Time	2 weeks
Maximum Core Average Depletion (% ^{235}U)	2.1%

CoC Page 28 of 31:

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

(xxii) For the SLOWPOKE Fuel Core described in Item 5.(b)(1)(xxi):

Up to 298 undamaged SLOWPOKE fuel rods may be loaded into a SLOWPOKE fuel core basket. A single loaded SLOWPOKE fuel core basket, accompanied by empty intermediate MTR-42 baskets and a bottom MTR-42 basket, such that the SLOWPOKE fuel core basket is adjacent to the NAC-LWT cask lid, as shown in NAC Drawing No. 315-40-185, may be transported in the NAC-LWT.

CoC Sections (revised)

CoC Page 30 of 31:

5(c) Criticality Safety Index (CSI)

For the SLOWPOKE Fuel Core described in 5.(b)(1)(xxi) and limited in 5.(b)(2)(xxii) 100

CoC Page 31 and 32 of 32

22. Revision 61 of this certificate may be used until February 28, 2015.

23. Expiration Date: February 28, 2015.

REFERENCES

NAC International, Inc., application dated June 18, 2010.

NAC International, Inc., supplements dated February 3, March 2, and May 24, October 26, and December 5, 2012; January 14, February 14, July 19 (two supplements), and October 18, 2013; December 28, 2012, March 14, 2013, March 5, 2014, July 16, 2014, October 17, 2014, and TBD 2015.

Enclosure 5

No. 71-9225 for NAC-LWT Cask

LOEP and SAR Page Changes

NAC-LWT SAR, Revision LWT-15A

SLOWPOKE Fuel Core Amendment

February 2015

Revision LWT-15A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Volume 1 of 3

NON-PROPRIETARY VERSION

Docket No. 71-9225

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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;
- up to eight (8) SLOWPOKE Fuel Canisters;
- up to eighteen (18) NRU or NRX Fuel Assemblies (or equivalent number of fuel rods);
- HEUNL; and
- One SLOWPOKE Fuel Core.

The authorized contents previously listed, except for HEUNL, 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:

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[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 300 Tritium Producing Burnable Absorber Rods (TPBARs), of which two can be prefailed; and
- 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
NRU / NRX Fuel Assemblies	100
HEUNL containers	0.0
SLOWPOKE Fuel Core	100

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 1.1-1 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 eighteen (18) NRU or NRX fuel assemblies. Fuel assemblies may be cropped. NRU fuel assemblies have the flow tube removed. NRX fuel assemblies/rods must be placed into the fuel rod caddy assembly as criticality analysis applied the fuel rod caddy as geometry constraints. Each basket tube is limited to the equivalent content of one assembly. One single fuel type may be loaded into one NRU/NRX basket assembly. NRU or NRX undamaged fuel assemblies/rods will be loaded into an 18 tube basket. • 4 HEUNL containers. Containers shall be empty or filled with HEUNL material, such that a minimum under filled cavity void of one gallon exists. • One SLOWPOKE fuel core. The SLOWPOKE fuel core contains up to 298 undamaged SLOWPOKE fuel rods, the upper and lower plates, and center tube. • 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.

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

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; iii. is missing fuel pins that have not been replaced by filler rods that displace a volume equal to, or greater than, that of the original fuel rod; iv. cannot be handled using normal handling methods. 3. The fuel is no longer in the form of an intact fuel assembly and consists of, or contains, debris such as loose pellets, rod segments, etc.
Damaged Fuel (TRIGA)	TRIGA fuel (elements and cluster rods) with known or suspected clad breach (i.e., cladding defects that permit the release of gas from the interior of the rod and/or allow water intrusion into the clad to fuel gap while submerged).
Fuel Debris (TRIGA)	TRIGA damaged fuel that does not maintain its structural integrity, including fuel particles, fuel debris, and broken fuel rods.
Degraded ANSTO Fuel	ANSTO fuel elements (Mark II MOATA, Mark III Spiral, and Mark IV DIDO fuel) that have corrosion, destructive examination and/or mechanical damage to the fuel plates, but are structurally acceptable for transport (i.e., will not result in appreciable fuel debris formation under transport conditions). Fuel elements may be disassembled and plates may contain significant corrosion or mechanical damage in nonfueled plate areas; fuel elements may have sections of the fueled plate removed for examination; or fuel elements may have fuel core exposure due to through-clad corrosion or mechanical damage. The fuel (core material) area exposed may not exceed 5% of the total fueled cross-sectional area of the element.

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

Undamaged Aluminum-Based Fuel	Aluminum-based reactor fuel plates/elements that are structurally sound, but may have fuel core exposure due to corrosion or mechanical damage of the clad. Through-clad corrosion and/or mechanical damage is limited to 5% of the fueled surface area of the element.
TPBAR	Tritium Producing Burnable Absorber Rod
Irradiated Fuel Material (IFM)	High-Temperature Gas-Cooled Reactor (HTGR/IFM) and Reduced-Enrichment Research and Test Reactor (RERTR/IFM) type TRIGA fuel entities produced by General Atomics.
PULSTAR Fuel Element	PULSTAR fuel rod. May be contained in either assembly, rod holder or can form for shipment. PULSTAR fuel elements may be intact or damaged.
Damaged PULSTAR Fuel Element	PULSTAR fuel rods having cladding failures greater than hairline cracks or pinhole leaks. The damaged fuel definition for PULSTAR fuel elements includes fuel debris. Damaged PULSTAR fuel elements may also be referred to as failed and must be transported in either of two types of PULSTAR cans.
Irradiated Hardware	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). Authorized quantity of irradiated hardware and components is limited to 4,000 lbs (including spacers, dunnage and containers) and a gamma source term as defined in Table 1.2-13.
MOX Fuel Rods	The term as used in this SAR is defined as irradiated or unirradiated mixed uranium-plutonium oxide (MOX) fuel rods. The MOX fuel rods can be made with plutonium having various compositions of plutonium isotopes. The evaluated mixes of the various grades of plutonium are defined in the shielding (Chapter 5) and criticality (Chapter 6) evaluations.

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

Damaged SLOWPOKE Fuel Rods in a SLOWPOKE Fuel Canister	SLOWPOKE fuel rod with visible damage to the clad and fuel debris. Damage to the fuel rods may be the result of through-clad corrosion and/or mechanical damage. Damaged fuel may include any size fuel section, including fuel debris. All SLOWPOKE fuel rods, damaged or undamaged, and fuel debris must be placed into the SLOWPOKE fuel canister. The SLOWPOKE fuel canister maintains a boundary for gross release of fuel material.
Undamaged NRU or NRX Fuel Assembly	NRU HEU and NRX HEU contain uranium aluminum alloy fuel meat. NRU LEU fuel meat is U-Al-Si. Fuel assemblies may be cropped. NRU fuel assemblies have their flow channel removed. Loose rods, including rods with through clad damage, are considered to be undamaged provided they retain their structural integrity.
HEUNL Container	A stainless steel container for highly enriched uranyl nitrate liquid (HEUNL). The container is comprised of a welded lid, shell, bottom plate structure with lid penetrations for fill and drain operations. HEUNL is a solution containing uranyl nitrates, various other nitrates, and water.
Undamaged SLOWPOKE Fuel Rods in the SLOWPOKE Fuel Core	SLOWPOKE fuel rods that are structurally sound, but may have fuel core exposure due to corrosion or mechanical damage of the clad. SLOWPOKE fuel rods are "Undamaged Aluminum-Based Fuel" therefore limiting through-clad corrosions and/or mechanical damage of the clad to 5% of the fueled surface area of the element.

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. 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);
- up to 18 NRU or NRX undamaged fuel assemblies (one per flow tube) or the equivalent number of loose rods as an assembly per basket tube (12 rods for NRU or 7 rods for NRX);
- 4 HEUNL containers (empty or filled such that a minimum under filled cavity void of one gallon exists);
- One SLOWPOKE fuel core containing up to 298 undamaged SLOWPOKE fuel rods; 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.

NRU or NRX undamaged fuel assemblies/rods will be loaded into an 18 tube basket. Assemblies or loose fuel rods may be placed into an aluminum caddy which in turn is placed into the basket tube. NRX fuel assemblies/rods must be placed into the fuel rod caddy assembly as criticality analysis applied the fuel rod caddy as geometry constraints. The NRU/NRX basket is spaced to the top of the cask cavity.

Four HEUNL containers may be loaded directly into the NAC-LWT cask cavity. Each container may be filled to the point when material reaches the vent port during fill operations. Partially filled containers are permitted for transport. [redacted]

One SLOWPOKE fuel core will be loaded into a SLOWPOKE fuel core basket. A spacer is bolted to the basket lid and will locate the fuel core at the bottom of the fuel core basket. The fuel core basket will be placed on top of empty intermediate and bottom MTR-42 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.

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; 640 W for NRU/NRX fuel assemblies; 4.65 W for HEUNL; 45 W for the SLOWPOKE fuel core; 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.
22. Maximum 18 NRU or NRX fuel assemblies or the equivalent number of loose rods. NRX assemblies or rods must be placed into a fuel rod caddy assembly for handling and geometry constraint. NRU fuel rods may be placed in a caddy. Only a single fuel type (NRU or NRX) shall be loaded in a single NRU/NRX fuel basket assembly.
23. Four HEUNL containers. Containers shall be empty or filled with HEUNL material such that a minimum under filled cavity void of one gallon exists.
24. One SLOWPOKE fuel core containing up to 298 undamaged SLOWPOKE fuel rods. The SLOWPOKE fuel core is packaged in the SLOWPOKE fuel core basket.

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.
3. Damaged TRIGA cluster rods and cluster rod fuel debris (up to the equivalent of six cluster rods) shall be loaded into a sealed DFC and then loaded into a top or base basket module.

1.2.3.1.2 TRIGA Fuel Baskets and Damaged Fuel Cans

The TRIGA fuel basket assembly configurations consist of five modules – a base module, three intermediate modules, and a top module. The three intermediate modules are interchangeable, but the base and top modules are required to be in their proper positions. Two basket configurations are available, "nonpoisoned" and "poisoned," where the poisoned basket configuration utilizes borated steel plates for additional criticality control. Each module has up to seven cells (fuel positions) for loading TRIGA fuel elements or cluster rods. The center cell of each module of the nonpoisoned basket configuration is blocked by a welded stainless steel baffle that prevents loading of that cell. The nonpoisoned configuration is also referred to as the 24-element basket or the 120-element loading, based on the maximum of 120 intact TRIGA fuel elements that may be loaded into the baskets in this configuration. The nonpoisoned configuration may also be loaded with a mixed loading of TRIGA fuel elements and TRIGA fuel cluster rods in separate cells of the basket module. The poisoned configuration is also referred to as the 28-element basket or the 140-element loading, based on the maximum of 140 intact TRIGA fuel elements that may be loaded into the baskets in this configuration. Additionally, the nonpoisoned configuration can accommodate up to 480 intact TRIGA fuel cluster rods, while the poisoned basket can hold up to 560 intact TRIGA fuel cluster rods.

Each basket module is a Type 304 stainless steel weldment consisting of longitudinal divider plates with circular support plates near each end; the top module also has a support plate at its midpoint due to its longer length. The poisoned basket modules contain four borated stainless steel plates that are seal welded to surfaces of the divider plates in the central region of the basket cross-section. The nonpoisoned basket modules are shown in Drawings 315-40-070, -071, and -072 and the poisoned basket modules are shown in Drawings 315-40-080, -081, and -082.

The nonpoisoned TRIGA fuel basket assembly in the NAC-LWT cask is shown in Drawing 315-40-079. The poisoned basket assembly in the NAC-LWT cask is shown in Drawing 315-40-084. In the poisoned basket configuration, an alternate assembly is presented that utilizes one base

module and four intermediate modules, along with a spacer (Drawing 315-40-083). The spacer is utilized to fill the space differential in the cask cavity resulting from the use of an additional intermediate module, rather than a top module. This additional assembly configuration is provided for flexibility in situations where the extra length provided by the top module is not needed. The fuel basket modules are described in further detail in Section 2.6.12.8. Damaged TRIGA fuel and fuel debris shall be loaded into sealed DFCs.

The sealed DFC is a 3.25-inch outside diameter tube with a 0.065-inch thick wall. The bottom of the sealed fuel can includes a check valve and drain plug to facilitate draining of the can. The top of the sealed DFC is closed by a bolted lid that is sealed with a metallic O-ring and includes a diaphragm valve to facilitate draining, drying, and helium backfilling of the can. The sealed DFC is constructed of austenitic stainless steels as shown on Drawings 315-40-086, -087, and-088.

1.2.3.2 MTR and DIDO Fuel and Basket Description

The MTR fuel elements to be shipped are 33 to 57 inches long, including the upper and lower nonfuel-bearing hardware, which may be removed from the element prior to transport. The MTR element fuel plates consist of a U-Al, U_3O_8-Al, or U_3Si_2-Al fuel meat clad with aluminum. The fuel plates are held in a parallel arrangement with two thick aluminum slotted pieces to form a fuel element. The active fuel region is typically 22.75 inches in height, and the fuel meat is typically 0.023-inch thick. MTR elements/plates may contain cadmium wires. A maximum 100-gram cadmium source is addressed in the shielding evaluations documented in Chapter 5. Axial fuel spacers and plates may be used in the cells of the basket modules to position MTR elements to facilitate fuel unloading and handling. The axial fuel spacers do not perform a safety function and are considered dunnage. The axial fuel spacers and plates are shown on Drawing 315-40-085.

A maximum of 42 MTR fuel elements has been analyzed for transport in the NAC-LWT cask. This configuration consists of up to seven fuel elements placed radially in each of the six axial fuel basket modules. Two alternate configurations of MTR fuel element loading provide for loads of 35 elements in five basket modules or 28 elements in four basket modules. HEU MTR fuel elements having > 380 g ^{235}U, but less than 460 g ^{235}U, shall have a minimum of 2.0 cm (0.8 inch) of nonfuel hardware and/or spacers/plates at both ends of the fuel element. The minimum 2.0 cm nonfuel hardware and/or spacer/plate dimension assures criticality control. The axial fuel spacer and plate design is shown on Drawing 315-40-085. For the shipment of MTR fuel elements (or an equivalent number of plates in a plate canister) having ^{235}U greater than 490 g per element, or greater than 23.5 g per plate (up to a maximum of 640 g per element or 32 g per plate), the maximum quantity of elements per basket module is limited to four, which are to be loaded in basket positions 4, 5, 6 and 7. Cell block spacers shall be installed in basket openings

1, 2 and 3 to block these cells from being inadvertently loaded with fuel elements. The cell block spacer design is shown on Drawing 315-40-085. Therefore, for the transport of elements of greater than 490 g ^{235}U, if only one element exceeds the 490 g (23.5 g per plate) limit, a maximum of four elements shall be loaded into the seven-element basket module and cell block spacers shall be placed in basket opening positions 1, 2 and 3.

Loose MTR fuel plates may be shipped in an MTR plate canister to facilitate handling. The contents of the canister are limited to the number of plates in the original intact fuel assembly, and the fuel plate dimensions and fuel masses must be bounded by the MTR fuel element limits in Table 1.2-4. The total weight per basket module cell for the fuel element, spacer(s) and fuel plate canister, as applicable, shall be limited to a maximum of 80 pounds.

A maximum of 42 DIDO fuel assemblies has been analyzed for transport in the NAC-LWT cask. Again, up to seven fuel assemblies may be placed radially in each of six axial fuel basket modules.

DIDO fuel assemblies are similar to MTR fuel elements in that the fuel bearing hardware consists of plates of fuel meat sandwiched by cladding. However, in DIDO fuel, the plates have been formed into tubular elements that are arranged in a concentric configuration. Typical DIDO assemblies contain four of the concentric tubes.

MTR and DIDO fuel characteristics are presented in Table 1.2-4.

1.2.3.3 General Atomics Irradiated Fuel Material (GA IFM) and Basket Description

The GA IFM is made up of two separate types of fuel material—the High-Temperature Gas-Cooled Reactor (HTGR) type fuel and the Reduced-Enrichment Research and Test Reactor (RERTR) type fuel. Each type of IFM is packaged in its own unique Fuel Handling Unit (FHU). Figures 1.2-7 and 1.2-8 illustrate the HTGR and RERTR FHUs. Detailed drawings for the GA and IFM FHUs are in Section 1.4.

The HTGR IFM is comprised of fuel in four forms: fuel particles (kernels), fuel particles (coatings), fuel compacts (rods), and fuel pebbles. Fuel kernels are solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of: UC_2, UCO, UO_2, $(Th,U)C_2$, or $(Th,U)O_2$. The as-manufactured enrichment of the HTGR fuel varies from ~10.0 to 93.15 wt % ^{235}U. Fuel coatings are solid, spheridized, isotropic, discrete multi-layered fuel particle coatings with chemical composition including pyrolitic-carbon (PyC) and silicon carbide (SiC). Fuel compacts are multi-coated ceramic fuel particles, bound in solid, cylindrical, injection-molded, high-temperature heat-treated compacts. The fuel compact matrix is composed of carbonized graphite shim, coke, and graphite powder. Fuel pebbles are multi-

coated fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles. The fully-cured binding matrix is composed of carbonized graphite shim, coke, and graphite powder.

The RERTR IFM is comprised of 20 irradiated TRIGA fuel elements; 13 of the elements are intact and the remaining seven have been previously sectioned for examination purposes. Parameters characterizing the RERTR/TRIGA fuel elements are shown in Table 6.2.9-1. Three distinct mass loadings of uranium were used in the 20 TRIGA elements: 20, 30, and 45 wt % U; the average mass of the fueled portion of these elements is 551g with an enrichment of 19.7 wt % ^{235}U. The RERTR IFM consists of U-ZrH metal alloy fuel material and as a solid meets the requirement of 10 CFR 71.63.

Two GA IFM Fuel Handling Units (FHU) are intended for a single shipment in the NAC-LWT. The first IFM FHU contains HTGR type fuel and the second contains RERTR type fuel. Each IFM FHU consists of stainless steel weld-encapsulated primary and secondary enclosures. The FHUs are filled and sealed with air at atmospheric pressure. The two IFM FHUs are placed in the top of the NAC-LWT cavity with a bottom spacer to facilitate unloading of the IFM packages.

The GA IFM fuel characteristics are presented in Table 1.2-7.

1.2.3.4 PWR Fuel Assembly

The NAC-LWT cask is analyzed for the PWR fuel assemblies listed in Table 1.2-5. This table provides the dimensional and enrichment constraints for the PWR fuel. The burnup and decay heat limits are specified in Table 1.2-4.

The PWR fuel rod cladding is a zirconium alloy type (Zircaloy-2, Zircaloy-4, Zirlo, M-5, etc.). Minor variations of alloy composition have no impact on performance of cladding material.

1.2.3.5 BWR Fuel Assembly

The NAC-LWT cask is analyzed for the BWR fuel assemblies listed in Table 1.2-6. This table provides the dimensional constraints for the BWR fuel. The enrichment, burnup and decay heat limits are specified in Table 1.2-4.

The BWR fuel rod cladding is a zirconium alloy type (Zircaloy-2, Zircaloy-4, Zirlo, M-5, etc.). Minor variations of alloy composition have no impact on performance of cladding material.

1.2.3.6 TPBARs

The NAC-LWT cask is analyzed for the transport of three separate Tritium Producing Burnable Absorber Rod (TPBAR) content configurations. For the transport of production TPBARs from

the reactor facility to the DOE processing facility, an open (i.e., unsealed) stainless steel consolidation canister is utilized to contain up to 300 TPBARs, two of which can be prefailed. The characteristics of the production TPBARs are listed in Table 1.2-8. The consolidation canister assembly is shown in Figure 1.2.3-10. Up to 25 TPBARs may also be transported within the 5×5 rod holder located within the PWR/BWR Rod Transport Canister. For a TPBAR shipment, the transport canister is located within the TPBAR basket. Up to two of the 25 TPBARs located within the rod holder may be classified as prefailed.

The third transport configuration is for the shipment of segmented TPBARs, following post-irradiation examination (PIE), contained in a welded stainless steel waste container containing segments and debris from up to 55 TPBARs. The characteristics of the TPBAR PIE segments are provided in Table 1.2-12. The waste container and extension weldment assembly is shown in Figure 1.2.3-16.

TPBARs are similar in size and nuclear characteristics to standard, commercial PWR, stainless steel-clad burnable absorber rods. The exterior of a typical TPBAR is a stainless steel clad tube. The internal components of the TPBAR are designed and selected to produce and retain tritium. Internal configurations differ for various TPBAR designs (see DOE reports provided in the Chapter 1 Appendices). The internal components of a typical TPBAR include a plenum spacer tube (getter tube), a spring clip or a plenum (compression) spring, pellet stack assemblies (pencils), and a bottom spacer tube. A pencil consists of a zirconium alloy liner around which lithium aluminate absorber pellets are stacked and then confined in a getter tube as shown in Figure 1.2.3-9. The unclassified design details of the various TPBAR designs are provided in the unclassified DOE documents and drawings provided in the Chapter 1 Appendices.

The transport assembly arrangements for the consolidation canister and waste container TPBAR content configurations are identical and include a closure lid spacer assembly, a TPBAR basket and Alternate B port covers with bolting installed. The detailed requirements for the NAC-LWT assembly are provided in license drawing 315-40-128 in Section 1.4. The overall payload arrangement for the NAC-LWT with the consolidation canister and waste container are shown in Figure 1.2.3-12 and Figure 1.2.3-17, respectively. For the transport of fewer than 300 TPBARs in the consolidation canister, stainless steel dunnage may be used to align and protect the contents. The weight and volume of the dunnage and the reduced TPBAR contents of the consolidation canister must be less than, or equal to, the weight and volume of 300 TPBARs. Up to 25 TPBPAR rods may also be transported in a PWR/BWR Rod Transport Canister in a NAC-LWT assembly as shown in License Drawing No. 315-40-104 (assembly 95).

The TPBAR content conditions are analyzed and evaluated for compliance with structural, thermal, containment and shielding conditions of the NAC-LWT in the appropriate SAR chapters. TPBARs do not contain fissile material and, therefore, criticality evaluations have not

been performed. The operating procedures for the wet and dry loading and dry unloading of the TPBAR contents are provided in Chapter 7. The special leakage and pressure testing requirements for NAC-LWT casks intended for the transport of TPBAR contents are provided in Chapter 8.

1.2.3.7 PWR/BWR Fuel Rods

PWR and BWR fuel rods are transported within the fuel lattice (skeleton) or 4×4 or 5×5 inserts (rod holder). The rod holder is located within a free flow, screened or sealed PWR/BWR transport canister. (The rod holder may also contain a nonfuel-bearing irradiated hardware component [e.g., BWR water rod, PWR instrument/guide tube].)

The PWR and BWR fuel rod cladding is of Zirconium alloy type (Zircaloy-2, Zircaloy-4, Zirlo, M-5, etc.). Minor variations of alloy composition have no impact on performance of cladding material.

1.2.3.8 PULSTAR Fuel Element and Transport Configuration Description

PULSTAR fuel elements are transported in the NAC-LWT in the 28 MTR fuel basket assembly, which contains four modules with seven cells per module. The basket assembly is composed of a top module, a base module, and two intermediate modules (Dwgs 315-40-051, -049, and -050, respectively).

PULSTAR fuel elements may be loaded into the module cells in one of four configurations: a) intact PULSTAR fuel assemblies b) intact PULSTAR fuel elements loaded into the 4×4 TRIGA fuel rod insert (Dwg. 315-40-096); c) intact or damaged PULSTAR fuel elements, fuel debris and nonfuel-bearing components of PULSTAR fuel assemblies in the PULSTAR screened can (Dwg. 315-40-135); or d) intact or damaged PULSTAR fuel elements, fuel debris and nonfuel-bearing components of PULSTAR fuel assemblies in the PULSTAR sealed can (Dwg. 315-40-130). The contents of either can type are restricted to a quantity of fissile material and a total volume of material equivalent to 25 PULSTAR fuel elements. The sealed cask contents are restricted to the displaced volume of 25 intact PULSTAR fuel elements. The total cask payload shall not exceed 700 PULSTAR fuel elements. Loading of modules with mixed PULSTAR payload configurations is allowed, but PULSTAR cans, either screened or sealed, are restricted to loading in the base and top modules.

PULSTAR fuel elements are low enriched (< 7 wt %) uranium oxide rods, with zirconium alloy cladding. During reactor operation, 25 PULSTAR fuel elements are arranged in a rectangular 5×5 lattice, surrounded by a zirconium alloy box, and capped by top- and bottom-end fittings to form a PULSTAR fuel assembly. The nonfuel components of a PULSTAR fuel assembly are primarily aluminum and zirconium alloy and do not contain a significant activation source. A

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 1.2-13. 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 in a SLOWPOKE Canister

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 are not loaded and contain 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 1.2-14.

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 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.

1.2.3.13 NRU/NRX Fuel Assemblies or Fuel Rods

NRU/NRX fuel assemblies and fuel rods are transported in the NAC-LWT in an 18 tube basket. The basket assembly is composed of 18 fuel tubes arranged in two concentric rings. The basket is spaced towards the top of the cask cavity by a bottom basket spacer.

NRX fuel assemblies or loose fuel rods must be loaded into a fuel rod caddy assembly. Loose NRU fuel rods may be loaded into a caddy. Mixed loading of NRU and NRX assemblies in a basket is not permitted. NRX assemblies are composed of (7) fuel rods and the NRU assemblies are composed of (12) fuel rods.

NRU/NRX HEU fuel rods are composed of highly enriched (> 90 wt%) uranium-aluminum alloy fuel meat within aluminum cladding. NRU LEU fuel meat is composed of <20% wt% ^{235}U enriched material composed of uranium-aluminum-silicon. NRU and NRX rods have a fin structure attached to the clad. The NRX rods have spiral fins to retain rod spacing. NRU assemblies in addition to the fins have a set of spacer disks assuring that rod pitch is maintained. A sketch of both NRU and NRX fuel assemblies is provided in Figure 1.2.3-20. Key physical, radiation protection and thermal characteristics of the NRU and NRX fuel assemblies are listed in Table 1.2-15.

The NRU/NRX caddy is constructed of aluminum. The aluminum caddy provides geometry constraint to fuel rod movement. Due to the increased reactivity of NRX fuel relative to high enriched NRU fuel, only NRX criticality evaluations credited this constraint.

1.2.3.14 HEUNL Containers

HEUNL material packaged in HEUNL containers may be directly loaded into the NAC-LWT cavity. Four containers must be packaged in the NAC-LWT for transport. The containers may be partially filled. [redacted] A sketch of the HEUNL container is provided in Figure 1.2.3-21. The container design is presented in NAC drawing 315-40-181. All hardware indicated on drawing 315-40-181 has been determined to be "Important to Safety" and has been evaluated, characterized and will be controlled in accordance with NAC's QA Program as described in Section 1.3.

[redacted]

[redacted]

HEUNL material consists of a solution of uranyl nitrate, various other nitrates (primarily aluminum nitrate), and water. The solution may contain uranyl nitrates with up to 7.40 g/L ^{235}U. Key physical, radiation protection, and thermal characteristics of the HEUNL material are provided in Table 1.2.3-16.

1.2.3.15 SLOWPOKE Fuel Core

One SLOWPOKE fuel core containing up to 298 undamaged SLOWPOKE fuel rods may be transported in the NAC-LWT. The SLOWPOKE fuel core is packaged in the SLOWPOKE fuel core basket. A spacer is attached to the SLOWPOKE fuel core basket lid locating the fuel core at the bottom of the basket. The basket is transported with empty intermediate and bottom MTR-42 basket modules to provide axial spacing. The SLOWPOKE fuel core basket is therefore located next to the NAC-LWT cask lid.

The SLOWPOKE fuel core primary components are up to 298 undamaged SLOWPOKE fuel rods, a center tube, and upper and lower plates. SLOWPOKE fuel rods are composed of highly enriched uranium-aluminum alloy fuel meat within aluminum cladding. As discussed in Section 1.2.3.12, criticality in a SLOWPOKE core during reactor operations is achieved by the use of a thick beryllium neutron reflector surrounding the core. The beryllium reflector is not part of the packaged contents. 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 core are listed in Table 1.2-17.

Table 1.2-16 HEUNL Characteristics

Parameter	Value
Maximum HEUNL payload per Container	15.35 gal
Maximum Cask Heat Load	4.65 W
Maximum Per Container Heat Load	1.16 W
Maximum HEUNL Heat Load	0.02 W/L
Maximum Curie Content (gamma emitters) [1]	9.0 Ci/L
Maximum ^{235}U content[2]	7.4 g ^{235}U/L
Maximum ^{235}U enrichment [2]	93.4 wt%

[1] Maximum Curie content defined by source term and shielding evaluations.
[2] Maximum ^{235}U content and enrichment defined by criticality evaluation.

Table 1.2-17 SLOWPOKE Fuel Core

Parameter	Value
Maximum Cask Heat Load (W)	45
Payload Limit (lb)	15
Maximum Number of Rods per Core	298
Maximum Initial ^{235}U per rod (g)	2.83
Maximum Initial Enrichment (wt % ^{235}U)	95.3
Maximum Initial ^{235}U per core (g)	837
Minimum Initial Enrichment (wt% ^{235}U)	90
Minimum Cool Time	2 weeks
Maximum Core Average Depletion (%^{235}U)	2.1%

Notes:

1 Heat load limit established by thermal analysis.

2 Maximum number of rods per core, fissile material (^{235}U) initial mass per rod limit, and maximum initial enrichment established by criticality analysis.

3 Fissile material (^{235}U) initial mass per fuel core, minimum initial enrichment, depletion percentage, and cool time established by shielding analysis.

4 Payload weight limit established by structural analysis.

B-227530

REV	CHANGE
0	INITIAL ISSUE

Security-Related Information
Figure Withheld Under 10 CFR 2.390

QUANTITY

UNLESS OTHERWISE STATED

GROUP	NAME	DATE
PREPARER	[illegible]	1-28-15
CHECKER	[illegible]	1-28-15
PROJECT MANAGER	[illegible]	1-28-15
ENGINEERING	[illegible]	1-28-15
LICENSING	[illegible]	1/28/15
QUALITY	Roy W. B[illegible]	2-2-15

NEXT ASSEMBLY: N/A
DRAWING TYPE: LICENSE

NAC INTERNATIONAL

LWT TRANSPORT CASK ASSEMBLY, SLOWPOKE CONTENTS

PROJECT 315-40 | DRAWING 185 | REV 0
SCALE N.T.S. | WEIGHT N/A | SH 1 OF 1

R-227724

REV	CHANGE
0	INITIAL ISSUE
1	INC OCR QA

Security-Related Information
Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL CORE BASKET
ASSEMBLY,
SLOWPOKE

GROUP	NAME	DATE
PREPARER	[illegible]	2-11-15
CHECKER	[illegible]	2-11-15
PROJECT MANAGER	[illegible]	2-11-15
ENGINEERING	[illegible]	2-11-15
LICENSING	[illegible]	2/11/15
QUALITY	Roy W. Bar[illegible]	2/12/15

UNLESS OTHERWISE STATED

NEXT ASSEMBLY 315-40-185

DRAWING TYPE LICENSE

PROJECT 315-40

DRAWING 186

REV 1

SCALE N T S

WEIGHT N/A

SH 1 OF 2

Security-Related Information
Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

FUEL CORE BASKET
ASSEMBLY,
SLOWPOKE

PROJECT 315-40		DRAWING 186	REV 1
SCALE N.T.S.	WEIGHT AS NOTED	SH 2 OF 2	

R-2275322

REV	CHANGE
0	INITIAL ISSUE

Security-Related Information
Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

UNLESS OTHERWISE STATED	GROUP	NAME	DATE
	PREPARER	[illegible]	1-30-15
	CHECKER	[illegible]	1-30-15
	PROJECT MANAGER	[illegible]	1-30-15
	ENGINEERING	[illegible]	1-30-15
NEXT ASSEMBLY 315-40-188	LICENSING	[illegible]	1/30/15
DRAWING TYPE LICENSE	QUALITY	[illegible]	2/2/15

BASKET LID ASSEMBLY, SLOWPOKE

PROJECT 315-40 | DRAWING 187 | REV 0

SCALE N.T.S. | WEIGHT N/A | SH 1 OF 2

Security-Related Information
Figure Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

BASKET LID
ASSEMBLY,
SLOWPOKE

PROJECT 315-40		DRAWING 187	REV 0
SCALE N. T. S.	WEIGHT AS NOTED	SH 2 OF 2	1/9/2015

February 2015

Revision LWT-15A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Volume 2 of 3

NON-PROPRIETARY VERSION

Docket No. 71-9225

NAC INTERNATIONAL

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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
SLOWPOKE Four Unit Basket	982	96.20
SLOWPOKE Fuel Payload	840[5]	96.20
NRU/NRX Basket & Spacer	845	113.03
NRU/NRX Basket + Spacer + Fuel	1205[6]	116.34

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 25 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.
5 A fuel weight of 30 lbs/assembly is used to compute the weight for this table as compared to the maximum weight for the SLOWPOKE canister of 25 pounds.
6 Each fuel tube in the NRU/NRX basket is limited to 20 pounds of fuel and aluminum caddy.

Table 2.2.1-1 Weights of the NAC-LWT Cask Major Components (cont.)

Component	Weight (pounds)	Axial Center of Gravity Location (inches)
HEUNL Container & Spacer[7]	1,422	104
HEUNL Payload	760	98
SLOWPOKE Fuel Core Basket & Five MTR Baskets	1,160	108
SLOWPOKE Fuel Core Payload	15	164

[7] Includes 4 HEUNL Containers, Container Guide and Container Spacer.

Table 2.2.1-2 Weights and Center of Gravity Locations for the NAC-LWT Cask Shipping Configurations

Component	Weight (pounds)	Axial Center of Gravity Location (inches)
Package -Loaded for Shipment (PWR) Maximum Payload	51,208	98.96
Package – Loaded for Shipment PWR High Burnup Rods	49,702	99.0
Package - Empty for Shipment (PWR)	48,082	99.12
Package - Loaded for Shipment (BWR)	49,832	99.00
Package - Empty for Shipment (BWR)	48,332	99.07
Package - Loaded for Shipment* (Metallic Fuel)	45,910	98.88
Package - Empty for Shipment* (Metallic Fuel)	43,830	99.09
Package - Loaded for Shipment (PULSTAR Fuel, MTR Four Unit Basket)	50,430	99.1
Package - Loaded for Shipment (MTR Fuel, Four Unit Basket)	49,030	99.1
Package - Empty for Shipment (MTR Fuel, Four Unit Basket)	48,190	98.9
Package - Loaded for Shipment (MTR Fuel, Five Unit Basket)	49,273	99.1
Package - Empty for Shipment (MTR Fuel, Five Unit Basket)	48,223	98.9
Package - Loaded for Shipment (MTR Fuel, Six Unit Basket)	49,470	99.1
Package - Empty for Shipment (MTR Fuel, Six Unit Basket)	48,210	98.9
Package - Loaded for Shipment (GA IFM Fuel and Basket)	48,147	99.3
Package - Empty for Shipment (GA IFM Basket)	48,026	99.3
Package – Loaded for Shipment (TPBARs and Basket)	48,861	99.2
Package – Empty for Shipment (TPBAR Basket)	47, 883	99.2
Package - Loaded for Shipment (ANSTO Fuel and Basket)	48,875	99.2
Package - Empty for Shipment (ANSTO Basket)	48,119	99.1

* Neutron Shield Tank is empty.

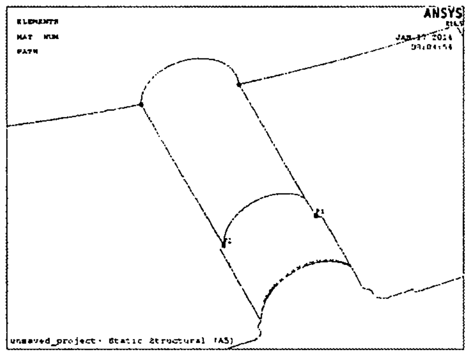
Table 2.2.1-2 Weights and Center of Gravity Locations for the NAC-LWT Cask Shipping Configurations (cont'd)

Component	Weight (pounds)	Axial Center of Gravity Location (inches)
Package - Loaded for Shipment TPBARs in the PWR/BWR Rod Transport Canister	49,109	99.2
Package - Empty for Shipment (TPBAR Basket for PWR/BWR Rod Transport Canister)	47,783	99.1
Package - Loaded for Shipment (SLOWPOKE Fuel, Four Unit Basket)[1]	49,030	99.1
Package - Empty for Shipment (SLOWOKE Fuel, Four Unit Basket)	48,190	98.9
Package – Loaded for Shipment (NRU/NRX Basket with Fuel Assemblies)	41,111	97.0
Package – Empty for Shipment (NRU/NRX Basket)	40,751	96.8
Package – Empty for Shipment (HEUNL)	48,656	99.2
Package – Loaded for Shipment (HEUNL)	49,416	99.2
Package – Loaded for Shipment (SLOWPOKE Fuel Core, Basket, Five MTR Baskets)[2]	48,400	99.3
Package – Empty for Shipment (SLOWPOKE Fuel Core Basket, Five MTR Baskets)[2]	48,400	99.3
Package - Design for Shipment	52,000	98.93

[1] A fuel weight of 30 lbs/assembly is used to compute the weight for this table as compared to the maximum weight for the SLOWPOKE canister of 25 pounds.

[2] Weight rounded up to nearest 100 pounds.

Figure 2.6.12-21 Stress Linearization Paths in Fill/Drain Ports



The maximum linearized stresses at these two locations are 16.78 ksi for membrane stress and 20.16 ksi for membrane plus bending stress. Since this is a displacement controlled load, the allowable stress for membrane plus bending is $3S_m$. For SA 240, Type 304 at -40 °F, S_m is 20 ksi. Therefore the margin of safety for the membrane plus bending stress is 1.98.

For additional details refer to item 1 in Section 2.6.12.15.5.

2.6.12.15.5 HEUNL Structural Calculations

1. 65008500-2010 "Canister Structural Evaluations for HEUNL in the NAC-LWT"

2.6.12.16 SLOWPOKE Fuel Core Basket

The SLOWPOKE fuel core basket assembly consists of the basket weldment, lid assembly weldment and an internal spacer assembly weldment. The basket weldment is 26.05 inches in length and consists of a circular cylinder which has an outer diameter of 10.75 inches with a 0.5-inch wall thickness. A base plate is welded to the bottom end of the cylinder and an annular ring with a 13.27-inch outer diameter and 0.75-inch thickness is welded to the top of the cylinder. A second annular ring with a 13.27-inch outer diameter and 0.5-inch thickness is welded near the midpoint of the cylinder. The lid assembly weldment is bolted to the top annular ring with six cone head bolts. The internal spacer assembly weldment fits inside of the basket and is bolted to the lid assembly with six socket head screws. The spacer assembly is 15.75 inches long and consists of a cylinder with two end plates attached. The outer diameter of the spacer cylinder is 8.0 inches and has a wall thickness of 0.38 inches. A shield plate, which has an outer diameter of 9.5 inches and is 1.5 inches thick, is welded to the lower end of the spacer cylinder and a base plate, with an outer diameter of 9.5 inches and 2.5 inches thick, is welded to the top end of the spacer cylinder.

The axial stack up of the basket, lid assembly and spacer leaves an axial gap of 9.8 inches between the end of the spacer assembly and the top of the basket base plate for the SLOWPOKE fuel core assembly. This space is sufficient to contain the SLOWPOKE fuel core assembly.

Due to the relatively small length of the SLOWPOKE basket as compared to the internal length of the LWT cask, five empty MTR basket assemblies are used as spacers between the lower end of the SLOWPOKE basket and the bottom forging of the cask. There are four intermediate MTR basket assemblies and one MTR base basket assembly between the SLOWPOKE basket and the bottom of the LWT cask.

The basket assembly, lid assembly and spacer assembly are fabricated from either SA-240/SA-269, Type 304 stainless steel or SA-269, Type 304 stainless steel. The cone head bolts are fabricated of SA-240, Type XM-19 stainless steel and the socket head screws are fabricated from SA-193, Grade B8S stainless steel.

2.6.12.16.1 SLOWPOKE Drop Analysis for the Normal Conditions of Transport

This section includes the evaluation of the SLOWPOKE basket, lid and spacer assemblies for longitudinal (end) and lateral (side) drop cases. The accelerations for the 1-foot drop cases are: an acceleration of 20 g for the end drop cases and 25 g for the side drop case.

The structural evaluation of the SLOWPOKE basket utilizes classic hand calculations.

Top End Drop

The acceleration used for the 1-foot top end drop is 20 g. The components which resist the load in a top end drop are; the lid assembly which is subjected to the weight of the loaded SLOWPOKE basket plus the 5 MTR spacers. For conservatism a bounding weight of 350 pounds is used for the loaded SLOWPOKE basket weight. The MTR basket assemblies have a bounding weight of 200 pounds each. The actual weights of these components are less.

A summary of the calculated stresses and margins of safety for the top end drop case are given in Table 2.6.12-4. Details of the calculations are given in Item 1 of Section 2.6.12.16-2.

Table 2.6.12-4 Stress Summary for Top End Drop Case

Assembly	Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Basket	Base Plate	Bearing	1.08	19.0
	Shell	Compression	1.49	11.9
Lid	Collar	Compression	9.09	1.12
	Collar cover plate	Bearing	0.87	23.8
Spacer	Shell	Compression	0.22	86.7
	Spacer/lid(1)	Bearing	0.034	633
	Spacer/lid(1)	Shear	0.16	70.9

Note:
(1) The spacer assembly bears against the base plate of the lid assembly for the top end drop case.

Bottom End Drop

The top end drop is the limiting case for the SLOWPOKE basket assembly since the basket components are only subjected to the weight of the loaded basket and will not be subjected to the weight of the empty MTR basket assemblies used as spacers below the SLOWPOKE basket. There is an increase in the bottom end drop bolts loads for the bolts attaching the basket lid spacer to the lid. However the bolt loads for the basket lid spacer are bounded by the side drop case.

Side Drop

The acceleration used for the 1 foot side drop is 25 g. The components which resist the load in a side drop are; the basket base plate and annular rings. In addition the spacer assembly is cantilevered from the lid assembly in the side drop case which produces bending and shear stresses in the spacer shell. This load case also results in bounding bolt loads for both the socket head screws and the cone head bolts.

A summary of the calculated stresses and margins of safety for the side drop case are given in Table 2.6.12-5. Details of the calculations are given in in Item 1 of Section 2.6.12.16-2.

Table 2.6.12-5 Stress Summary for Side Drop Case

Assembly	Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Basket	Shell	Bending	1.42	19.4
		Shear	0.212	20.8
	Base plate/rings	Bearing	5.25	3.14
Spacer	Shell	Bending	0.9	31.2
		Shear	0.2	56.6
Lid	Socket head screws	Tensile	6.11	2.42
		Thread shear	0.93	5.74
Lid	Cone head bolts	Tensile(2)	14.21	1.17
		Thread shear(1)	8.48	2.05

Notes:
(1) Bounding thread shear stress for external/internal shear
(2) For the reduced section of the cone head bolt

2.6.12.16.2 SLOWPOKE Structural Calculation

1. NAC Calculation 50026-2001, "Structural Evaluation of the SLOWPOKE Fuel Core Basket."

2.6.12.17 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, +7.78 as reported in Section 2.6.12.13 for the SLOWPOKE fuel canister assembly; and +1.12 as reported in Section 2.6.12.14 for the NRU/NRX basket and 1.12 as reported in Section 2.6.12.16 for the SLOWPOKE fuel core. The HEUNL container has a minimum margin of safety of +0.19 as reported in Section 2.6.12.15.2. Therefore, it can be concluded that all basket designs have sufficient structural integrity for adequate service during normal conditions of transport.

2.7.7.18 SLOWPOKE Fuel Core Basket for the Accident Conditions of Transport

This section includes the evaluation of the SLOWPOKE basket, lid and spacer assemblies for longitudinal (end) and lateral (side) drop cases. The accelerations for the 30-foot drop cases are: an acceleration of 60 g for the end drop case, and 55 g for the side drop case.

The structural evaluation of the SLOWPOKE basket utilizes classic hand calculations.

Top End Drop

The acceleration used for the 30-foot top end drop is 60 g. The components which resist the load in a top end drop are: the lid assembly, which is subjected to the weight of the loaded SLOWPOKE basket, plus the 5 MTR spacers. For conservatism, a bounding weight of 350 pounds is used for the loaded SLOWPOKE basket. The MTR basket assemblies have a bounding weight of 200 pounds each. The actual weights of these components are less.

A summary of the calculated stresses and margins of safety for the top end drop case are given in Table 2.7.7-7. Details of the calculations are given in Item 1 of Section 2.7.7.18.1.

Table 2.7.7-7 Stress Summary for Top End Drop Case

Assembly	Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Basket	Base Plate	Bearing	N/A	N/A
	Shell	Compression	4.47	9.2
Lid	Collar	Compression	27.3	0.67
	Collar cover plate	Bearing	N/A	N/A
Spacer	Shell	Compression	0.66	68.0
	Spacer/lid(1)	Bearing	N/A	N/A
	Spacer/lid(1)	Shear	0.48	66.4

Note:
(1) The spacer assembly bears against the base plate of the lid assembly for the top end drop case.

Bottom End Drop

The top end drop is the limiting case for the SLOWPOKE basket assembly, since the basket components are only subjected to the weight of the loaded basket and will not be subjected to the weight of the empty MTR basket assemblies used as spacers below the SLOWPOKE basket. There is an increase in the bottom end drop bolt loads for the bolts attaching the basket lid spacer to the lid. However, the bolt loads for the basket lid spacer are bounded by the side drop case.

Side Drop

The acceleration used for the 30-foot side drop is 55 g. The components which resist the load in a side drop are the basket base plate and annular rings. In addition, the spacer assembly is cantilevered from the lid assembly in the side drop case, which produces bending and shear stresses in the spacer shell. This load case also produces increased bolt loads for both the socket head screws and the cone head bolts.

A summary of the calculated stresses and margins of safety for the side drop case are given in Table 2.7.7-8. Details of the calculations are given in Item 1 of Section 2.7.7.18.1.

Table 2.7.7-8 Stress Summary for Side Drop Case

Assembly	Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Basket	Shell	Bending	3.12	19.9
		Shear	0.467	26.8
	Base plate/rings	Bearing	N/A	N/A
Spacer	Shell	Bending	1.98	31.9
		Shear	0.44	78.2
Lid	Socket head screws	Tensile	20.6	0.51
		Thread shear(1)	3.13	8.5
Lid	Cone head bolts	Tensile(2)	23.39	0.80
		Thread shear(1)	7.44	5.22

Notes:
(1) Bounding thread shear stress for external/internal shear
(2) For the reduced section of the cone head bolt

2.7.7.18.1 SLOWPOKE Structural Calculation

1. NAC Calculation 50026-2001, "Structural Evaluation of the SLOWPOKE Fuel Core Basket."

2.9.5 SLOWPOKE Fuel Core Assembly

The structural integrity of the SLOWPOKE fuel core is evaluated in this section for normal and accident conditions of transport. The SLOWPOKE fuel Core assembly consists of 298 fuel pins supported by a circular cage. The fuel cage consists of two perforated plates (upper and lower) that the fuel pins pass through. The two plates are held in position by three round posts between the two that are plug welded on both ends. The lower plate has three bottom pins that project beyond the bottom surface which act as spacers. The upper plate has two rivet pins that project beyond the top surface. The fuel pins have an end cap with a reduced section that penetrates through the lower plate. These ends of the fuel pins are peened into the lower plate. This restrains the fuel pins with respect to the cage.

The maximum loading for each condition is due to the drop cases. For the end drop cases the load is due to a 20 g acceleration for the Normal Conditions of Transport and a 60 g acceleration for the Accident Conditions of Transport. For the side drop cases the load is due to a 25 g acceleration for the Normal Conditions of Transport and a 55 g acceleration for the Accident Conditions of Transport.

2.9.5.1 Normal Conditions of Transport

End Drop Cases

Top End Drop

For the top end drop, the members evaluated are the fuel pins, the rivet pins located on the upper plate and the posts between the upper and lower plates. The fuel pins are restrained at the bottom plate and the fuel core assembly weight is reacted by the two rivet pins on the upper plate.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Tensile	0.06	55
Rivet Pins	Bearing	1.96	3.17
Posts	Compressive	1.3	5.28

The posts separating the upper and lower plates were also checked for potential buckling. The Euler buckling load was conservatively calculated to be 609 lbs. The load on each post was calculated to be 75 lbs. Then the margin of safety for buckling is 7.12.

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

Bottom End Drop

For the bottom end drop, the members evaluated are the fuel pins and the bottom pins located on the lower plate. The fuel core assembly weight is reacted by the three bottom pins attached to the lower plate.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Compressive	0.06	55
Bottom Pins	Compressive	1.3	5.25
Posts	Compressive	1.3	5.28

Note that the load on the posts for the bottom end drop is the same as for the top end drop.

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

Side Drop Case

For the side drop case, the members evaluated are the fuel pins, the upper and lower plates and the posts.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Bending	3.83	0.32
Fuel Pins	Shear	0.038	52.2
Posts	Bending	1.21	9.10
Posts	Shear	0.01	489
Support plates	Shear Tear Out	0.125	38.2
Support plates	Bearing	0.81	23.4

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

2.9.5.2 Accident Conditions of Transport

End Drop Cases

Top End Drop

For the top end drop, the members evaluated are the fuel pins, the rivet pins located on the upper plate and the posts between the upper and lower plates. The fuel pins are restrained at the bottom plate and the fuel core assembly weight is reacted by the two rivet pins on the upper plate.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Tensile	0.18	38.3
Rivet Pins	Compressive	5.88	3.17
Posts	Compressive	3.9	5.26

The posts separating the upper and lower plates were also checked for potential buckling. The Euler buckling load was conservatively calculated to be 609 lbs. The load on each post was calculated to be 75 lbs. Therefore, the margin of safety for buckling is 7.12.

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

Based on the comparison to the conservative analysis performed for the SLOWPOKE fuel core assembly, the SLOWPOKE fuel core would not be adversely affected by the normal or accidental drop conditions.

Bottom End Drop

For the bottom end drop, the members evaluated are the fuel pins and the bottom pins located on the lower plate. The fuel core assembly weight is reacted by the three bottom pins attached to the lower plate.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Compressive	0.12	57.9
Bottom Pins	Compressive	3.9	5.26
Posts	Compressive	3.9	5.26

Note that the load on the posts for the bottom end drop is the same as for the top end drop.

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

Side Drop Case

For the side drop case, the members evaluated are the fuel pins, the upper and lower plates and the posts.

Component	Stress Category	Calculated Stress (ksi)	Margin of Safety
Fuel Pins	Bending	8.42	0.2
Fuel Pins	Shear	0.083	59.8
Posts	Bending	2.66	8.21
Posts	Shear	0.022	556
Support plates	Shear Tear Out	0.275	43.5
Support plates	Bearing	N/A	N/A

Detailed calculations are contained in Item 1 of Section 2.9.5.3.

2.9.5.3 SLOWPOKE Structural Calculation

1. NAC Calculation 50026-2001, "Structural Evaluation of the SLOWPOKE Fuel Core Basket."

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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. Up to eighteen (18) NRU or NRX fuel rod assemblies, or equivalent fuel rods, will be loaded into each basket in the NAC-LWT cask enclosed in an ISO container. Four (4) HEUNL containers, empty or filled with HEUNL material such that a minimum under filled void of one gallon exists, will be loaded into the NAC-LWT cask enclosed in an ISO container.

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 intermediate basket modules and bottom basket module are installed as axial spacers. The total package decay heat for SLOWPOKE fuel contents in the fuel canisters is 5 Watts. In addition, a SLOWPOKE fuel core can be loaded in a SLOWPOKE fuel basket, which is placed on top of empty intermediate basket modules and a bottom MTR-42 fuel basket module. The maximum decay heat of the fuel core is 45 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). The maximum heat load for the maximum number of (18) NRU or NRX fuel rod assemblies is 0.64 kW per cask. The maximum heat load for four (4) HEUNL containers filled to capacity is 4.65 Watts per cask. 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.

whole cask is used and is distributed over the cavities of the four (4) containers. The corresponding heat generation rate applied to the liquid in the two-dimensional model is 0.05 W/liter (0.0028 Btu/hr-in^3 which bounds the value reported in Section 4.5.6.1).

A constant temperature of 134°F is applied to the outer surface of the HEUNL container model (Figure 3.4-22), which corresponds to the maximum inner shell temperature. This maximum cask inner shell temperature is obtained using the two-dimensional ANSYS model for normal condition (Condition 1) from Section 3.4.1.3 after deleting all elements inside the cask inner shell. A heat flux computed based on a heat load of 12.88 W/cask, as follows, is applied to the inner surface of the cask inner shell of this two-dimensional model to obtain the maximum cask inner shell temperature of 134°F.

$$H\_flux = \frac{3.22 \times 3.413}{\pi \times 13.375 \times 30} = 0.0088\, Btu/(hr - in.^2)$$

where: 3.22 Watts is the heat load for a container;
13.375 inches is the ID of the LWT cask inner shell;
30 inches is conservatively used for the axial length for one container assembly.

The maximum temperature in the HEUNL model is computed to be 139°F. This confirms that for normal condition of transport, the HEUNL remains in the liquid state and that the maximum HEUNL container temperature is significantly lower than the allowable temperature of 800°F for stainless steel (Table 3.4-10). Maximum temperature of slide bars [redacted] of the HEUNL container is bounded by the liquid temperature of 139°F, which is lower than the allowable temperature of 180°F for [redacted]. This allowable temperature is established [redacted] Specifications. The heat load of the HEUNL (12.88 W) is bounded by the heat load of the MTR fuel (1.26 kW, Section 3.1). Therefore, the maximum temperatures of the cask components for the MTR contents (Condition 1, Table 3.4-6) bounds the maximum temperatures for the cask components for HEUNL.

3.4.1.21 Thermal Evaluation of SLOWPOKE Fuel Core

Heat transfer analysis of the NAC-LWT containing a SLOWPOKE fuel core is performed using a quarter-symmetry, two-dimensional, planar finite element model and the ANSYS computer code. The NAC-LWT is supported in an ISO container with solar insolance on the surface of the ISO container, and the NAC-LWT is considered to be insulated from the environment (only for the 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.

The SLOWPOKE fuel core consists of 298 undamaged SLOWPOKE fuel rods, center tube, and upper and lower plates. SLOWPOKE fuel rods are composed of highly enriched uranium-aluminum alloy fuel meat within aluminum cladding. The SLOWPOKE fuel core is packaged with a spacer in the SLOWPOKE basket, which is placed at the top of empty MTR-42 basket modules. A bounding decay heat of 45 Watts is considered in the thermal evaluation for fuel core. The 45 Watts total heat load is enveloped by the 1.26 kW total heat load for the NAC-LWT MTR fuel contents contained in Section 3.4.1.3. Since the NAC-LWT cask ambient conditions is the same for the SLOWPOKE fuel core as for the MTR fuel, the maximum temperature of all cask body components for the SLOWPOKE contents are enveloped by the maximum cask component temperatures for the MTR fuel contents. Therefore, the cask inner shell temperature for the MTR fuel contents bounds the maximum cask inner shell temperature for the SLOWPOKE fuel core configuration. The maximum cask inner shell temperature is used as the boundary condition for the finite element model for the SLOWPOKE thermal evaluation.

The evaluation of the maximum component temperatures for SLOWPOKE fuel core is performed using a two-dimensional finite element model as shown in Figure 3.4-23. This model corresponds to the stainless steel basket shell, the SLOWPOKE fuel rods, and the helium inside the cask cavity and helium inside the basket shell.

Any axial conductance of the contents is conservatively neglected in this two-dimensional, quarter-symmetry, planar model. The ANSYS PLANE55 and MATRIX50 elements are used. While convection is conservatively ignored, radiation is considered using radiation matrix elements in the following regions.

- Between the outer surfaces of SLOWPOKE fuel rods.
- Between the outer surfaces of the SLOWPOKE fuel rods and the inner surface of the basket shell.
- Between the outer surface of the basket shell and the inner surface of the cask inner shell.

A constant temperature of 214°F (Table 3.4-6, Condition 1) is applied to the outer surface of the model, which corresponds to the inner surface of the cask inner shell.

The decay heat generated by the fuel rods is applied via a heat generation rate to the fuel meat region of each fuel rod.

The thermal analysis demonstrates that the temperature of the SLOWPOKE fuel core is maintained within a conservative limit of 400°F for normal conditions of transport. At 400°F, the aluminum retains its capability as a mechanical component.

Maximum component temperatures for the NAC-LWT containing SLOWPOKE fuel core are summarized in Table 3.4-28. Maximum cask component temperatures for normal conditions are conservatively obtained from the analysis results corresponding to the MTR fuel contents, as shown in Table 3.4-6 (Condition 1).

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. The maximum temperature inside the cask for NRU/NRX fuels is 245°F. The maximum temperatures of the cask components for the MTR contents (Condition 1, Table 3.4-6) bound the maximum temperatures for the cask components for the NRU/NRX material contents. The maximum temperature of HEUNL is 139°F. The maximum temperatures of the cask components for the MTR contents (Condition 1, Table 3.4-6) bound the maximum temperatures for the cask components for the HEUNL contents.

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:

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

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

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:

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

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

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

V_{total} = 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:

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

$P_{initial}$ = 29.3 psia

The minimum free gas volume is calculated as:

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

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

$V_{canister}$ = 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} = ~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:

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

where:

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

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

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

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

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

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

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

then:

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

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:

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

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:

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

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

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/UO2 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, MTR, and NRU/NRX 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.

NRU/NRX payloads are not evaluated for system pressure as inputs into the analysis outlined below; all indicate a conservative system pressure being obtained from the MTR payload:

- Total heat load and temperature are below that of the MTR payload. Furthermore, total fuel and fissile material mass (U-Al, or UAl-Si) in the 18 NRU/NRX assemblies is less than MTR fuel mass (2 elements maximum).

- MTR elements were evaluated at higher burnup levels than NRU/NRX and therefore, the NRX/NRU fuel will contain less fission gas.

The void space in the NRU/NRX cask cavity is higher than that of the fully loaded MTR system as the NRU/NRX bottom basket spacer occupies very little volume versus a loaded MTR basket and the NRU/NRX fuel assembly and basket cross section contains significant void areas.

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.

3.4.4.9 Maximum Internal Pressure for HEUNL Contents

3.4.4.9.1 Cask Containment

[redacted]

3.4.4.9.2 HEUNL Container

[redacted]

3.4.5 Maximum Thermal Stresses

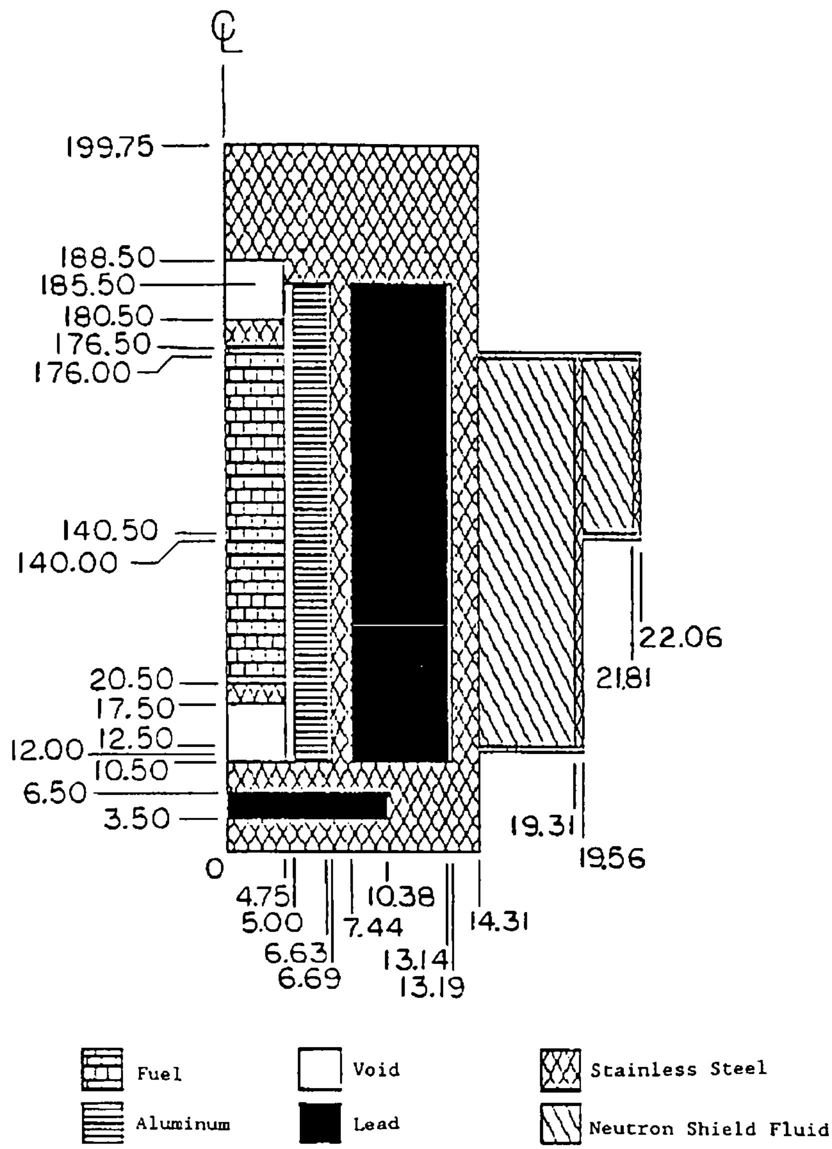
The conditions within the range of normal transport conditions and fabrication that result in the limiting combination of thermal gradient and isothermal stresses have been evaluated. The analyses are performed in Sections 2.5 through 2.7. The resulting isothermal temperature plots are presented in Section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



(Dimensions in inches)

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

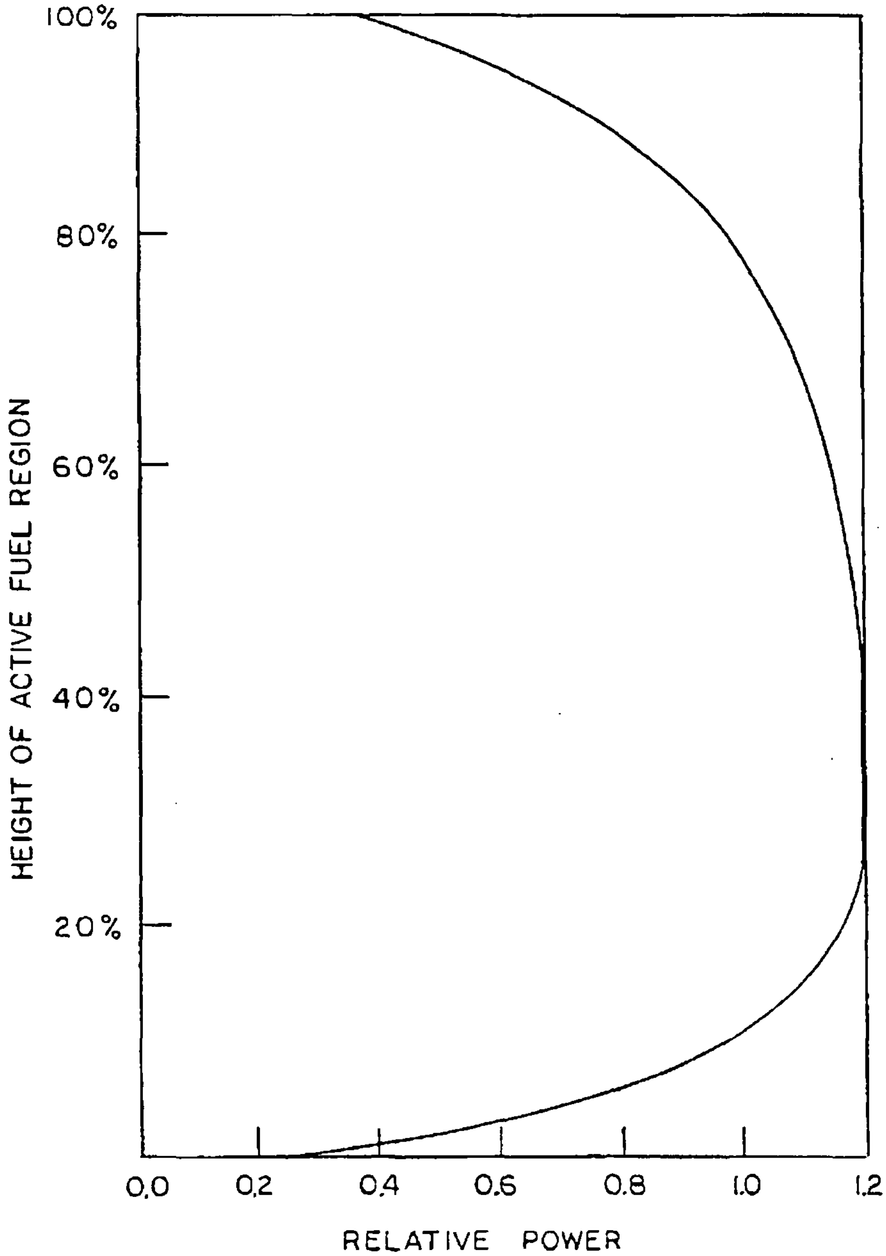


Figure 3.4-3 ANSYS MTR Fuel Design Basis Heat Load Thermal Model (Uniform 30-Watt/Element Configuration Heat Load)

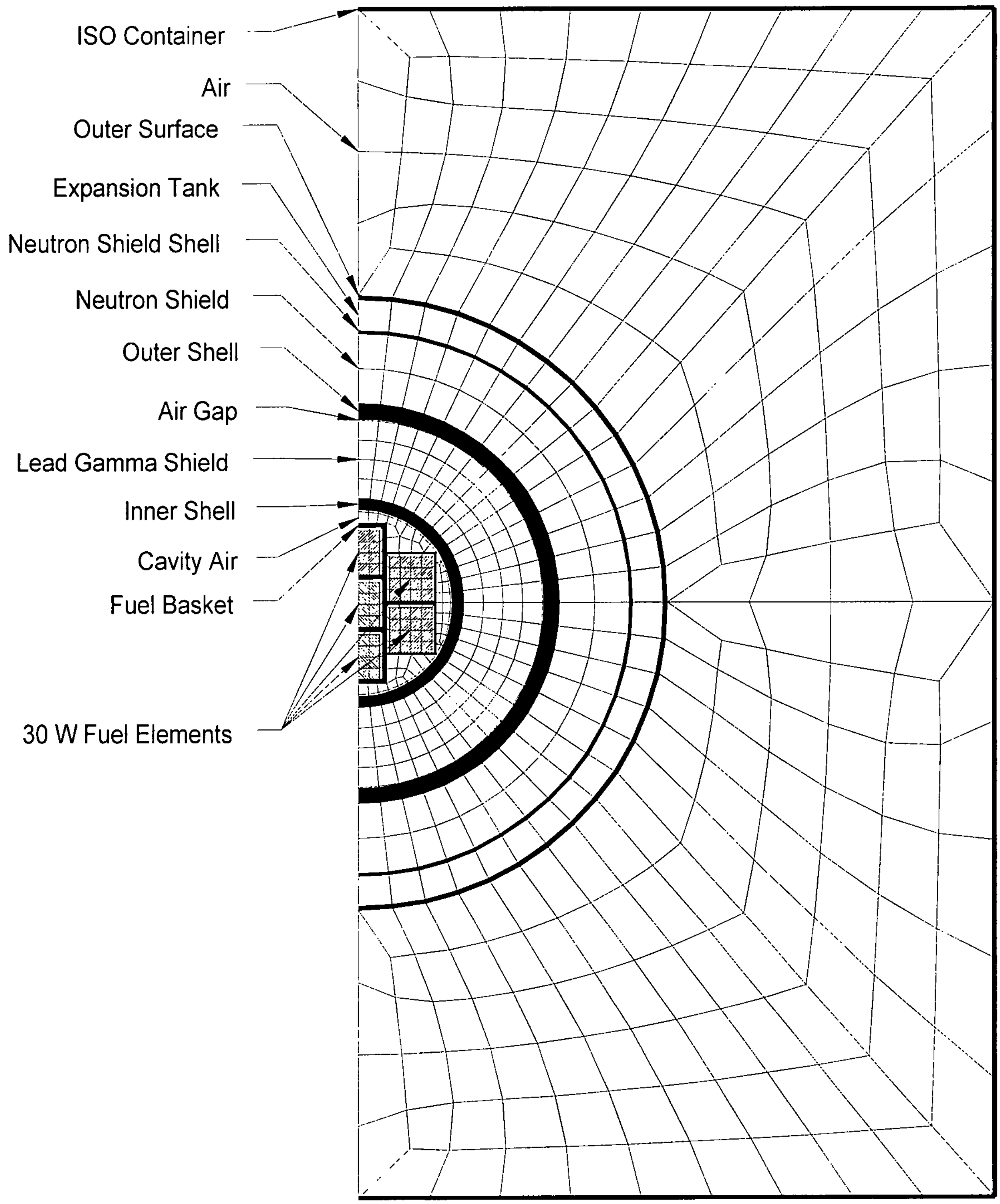


Figure 3.4-4 MTR Fuel Variable Decay Heat ANSYS Thermal Model
(120-Watt / 70-Watt / 20-Watt Configuration Heat Load)

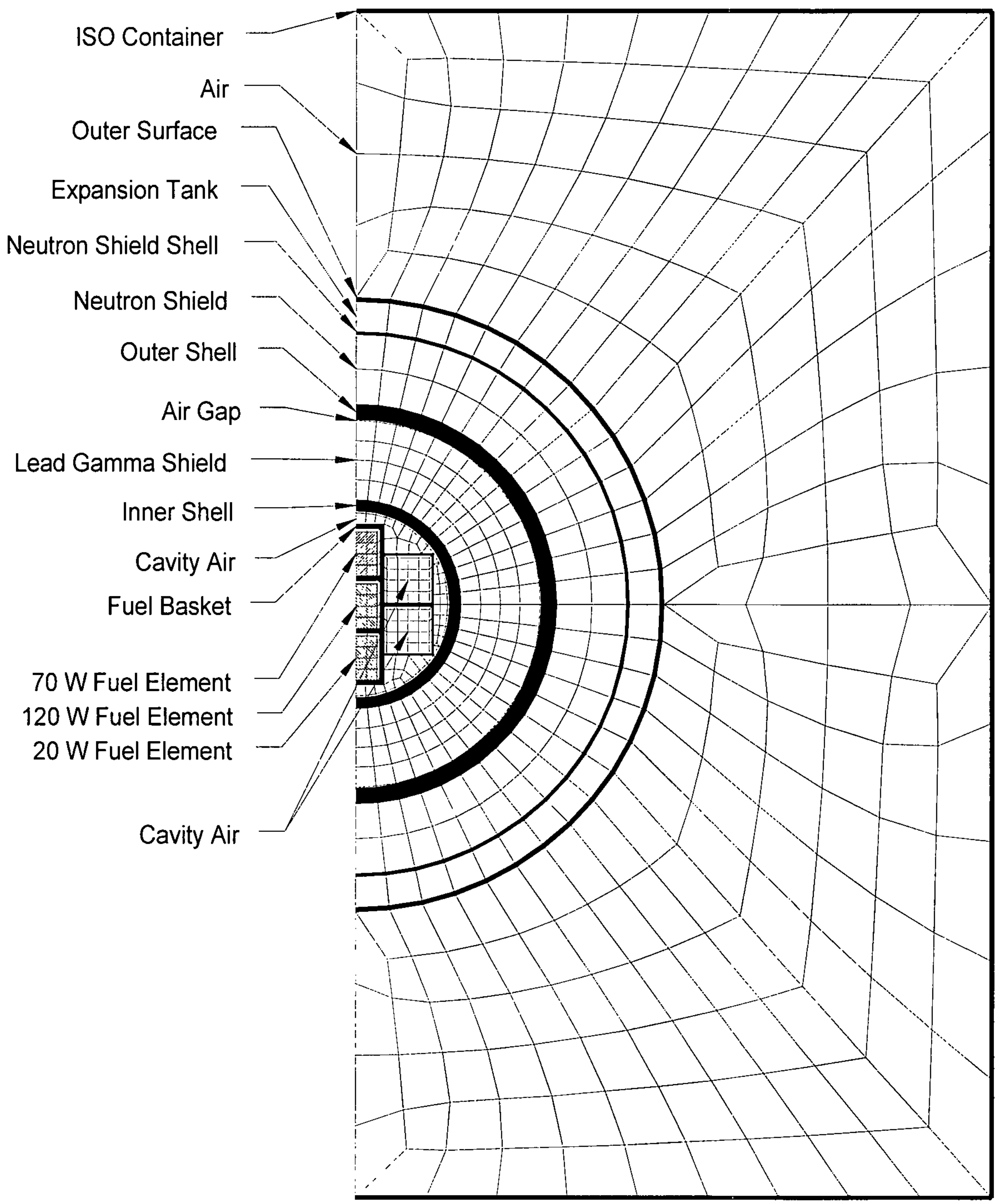
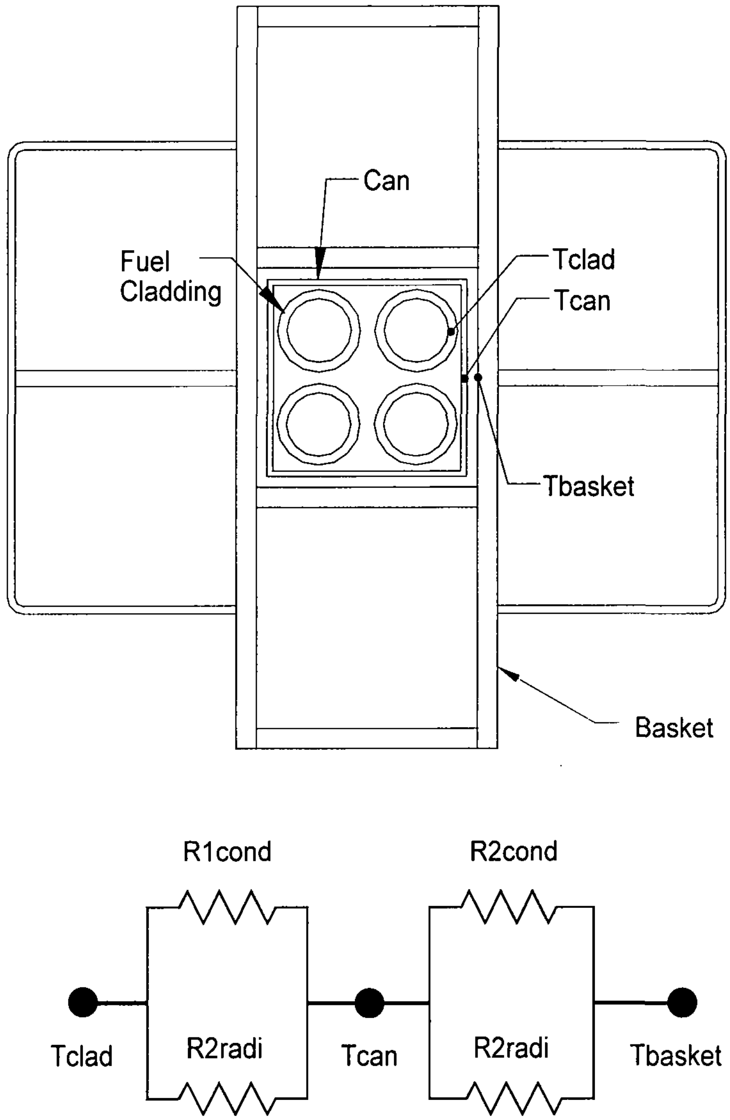


Figure 3.4-5 Thermal Resistance Model for TRIGA Fuel Elements



Where:

Rcond =	Thermal resistance for conduction	Tclad =	Maximum fuel cladding temperature
Rradi =	Thermal resistance for radiation	Tcan =	Maximum can temperature
Tbasket =	Maximum basket temperature		

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

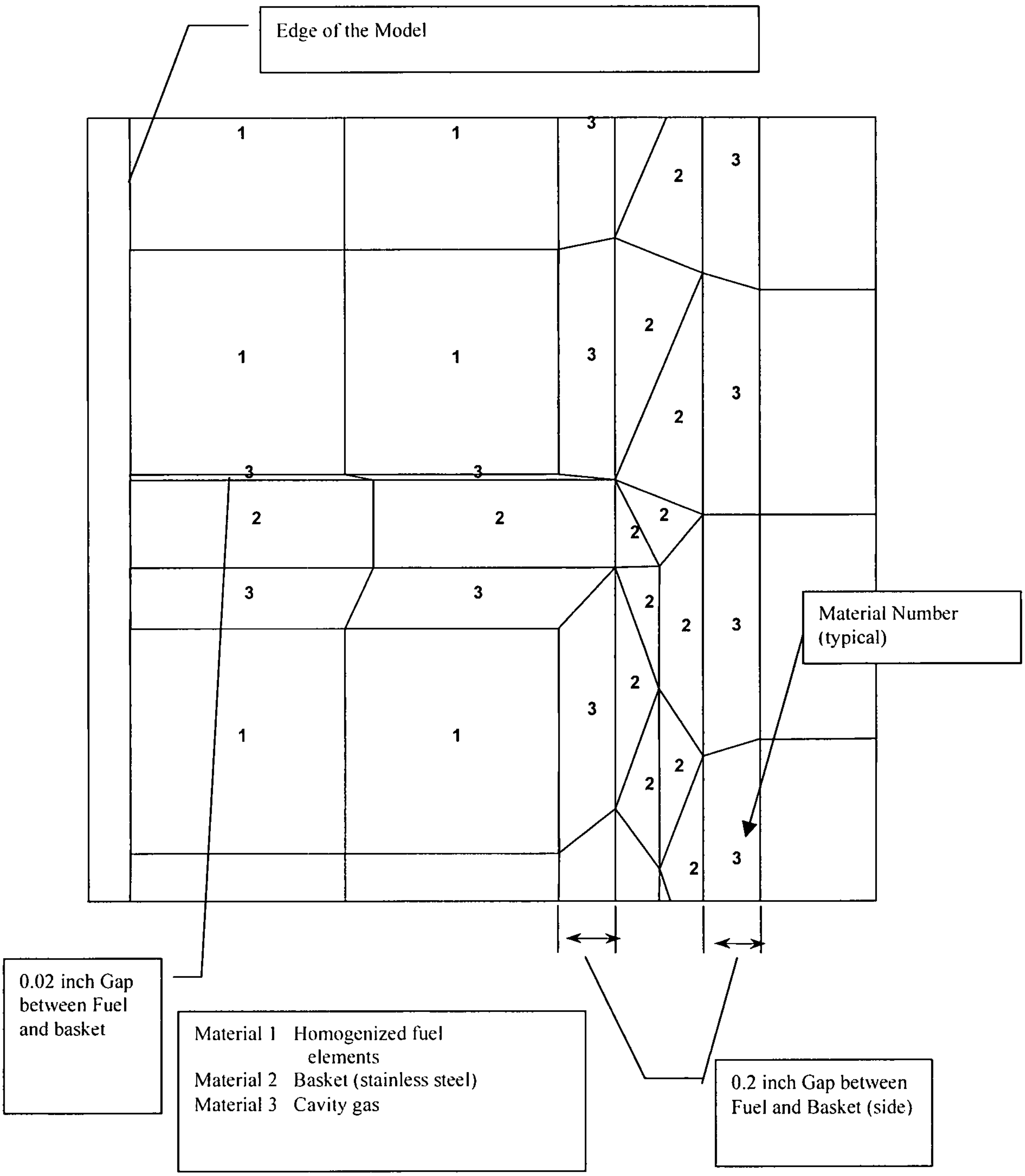


Figure 3.4-7 Finite Element Thermal Model for TRIGA Fuel Cluster Rods

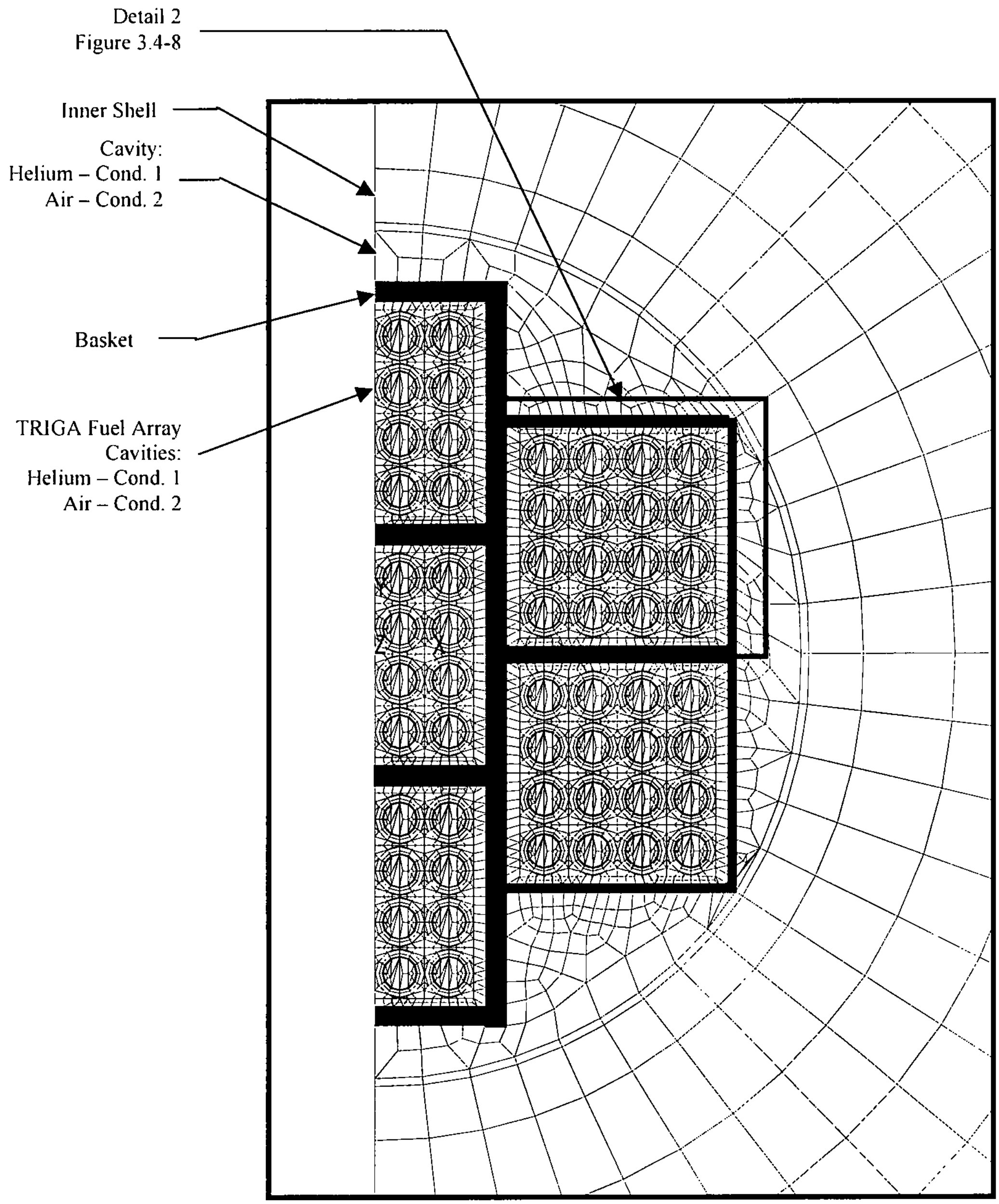


Figure 3.4-8 Details of the TRIGA Fuel Cluster Rods in the Finite Element Model

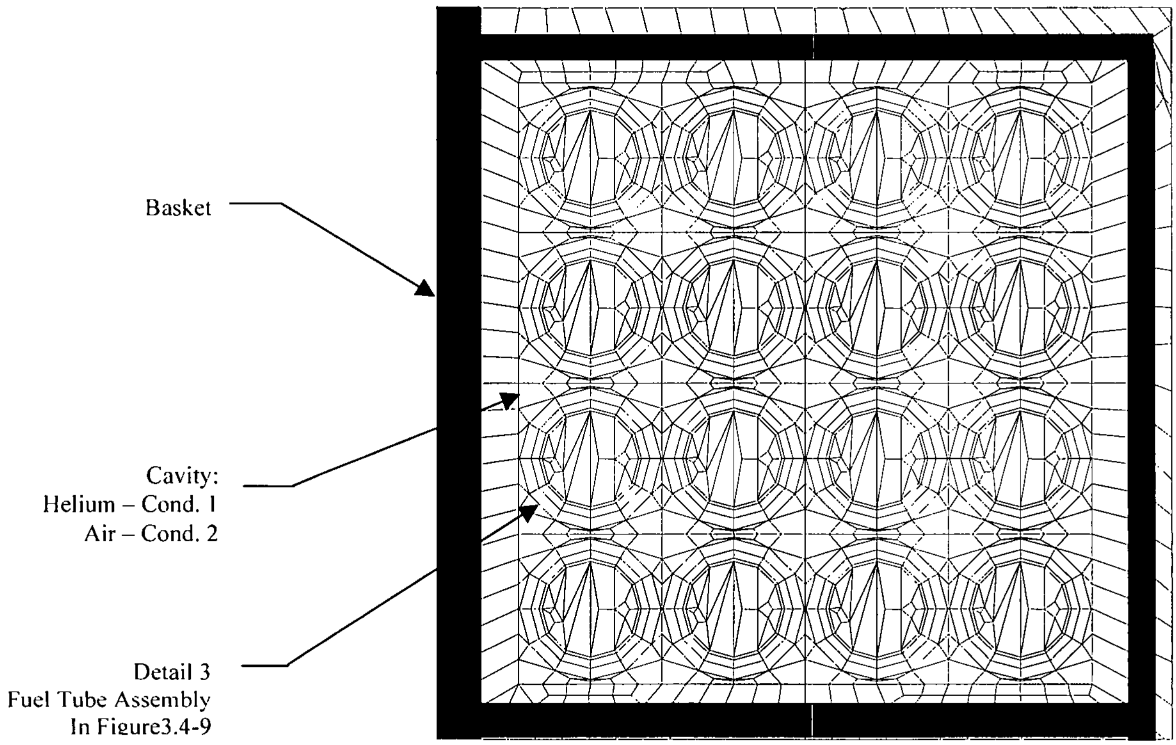


Figure 3.4-9 Individual TRIGA Fuel Cluster Rod Finite Element Model Details

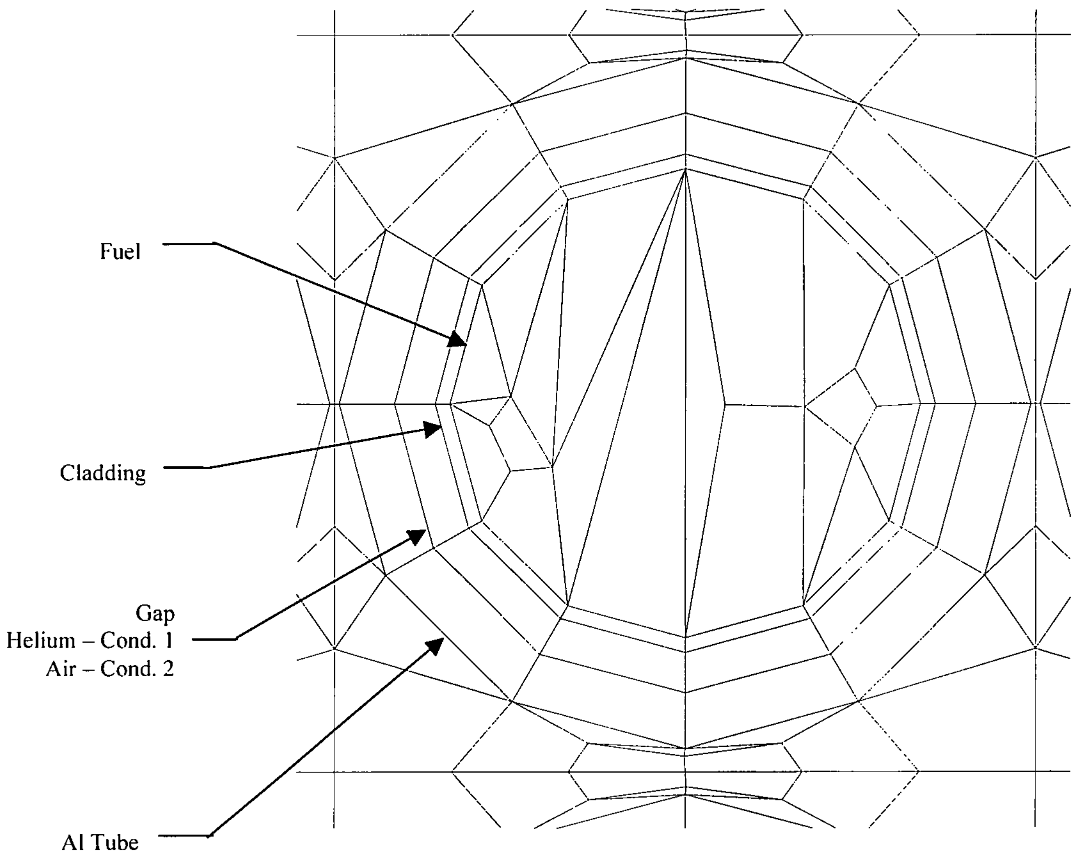


Figure 3.4-10 PWR and BWR High Burnup Fuel Rods Normal Condition ANSYS Thermal Model (Condition 1)

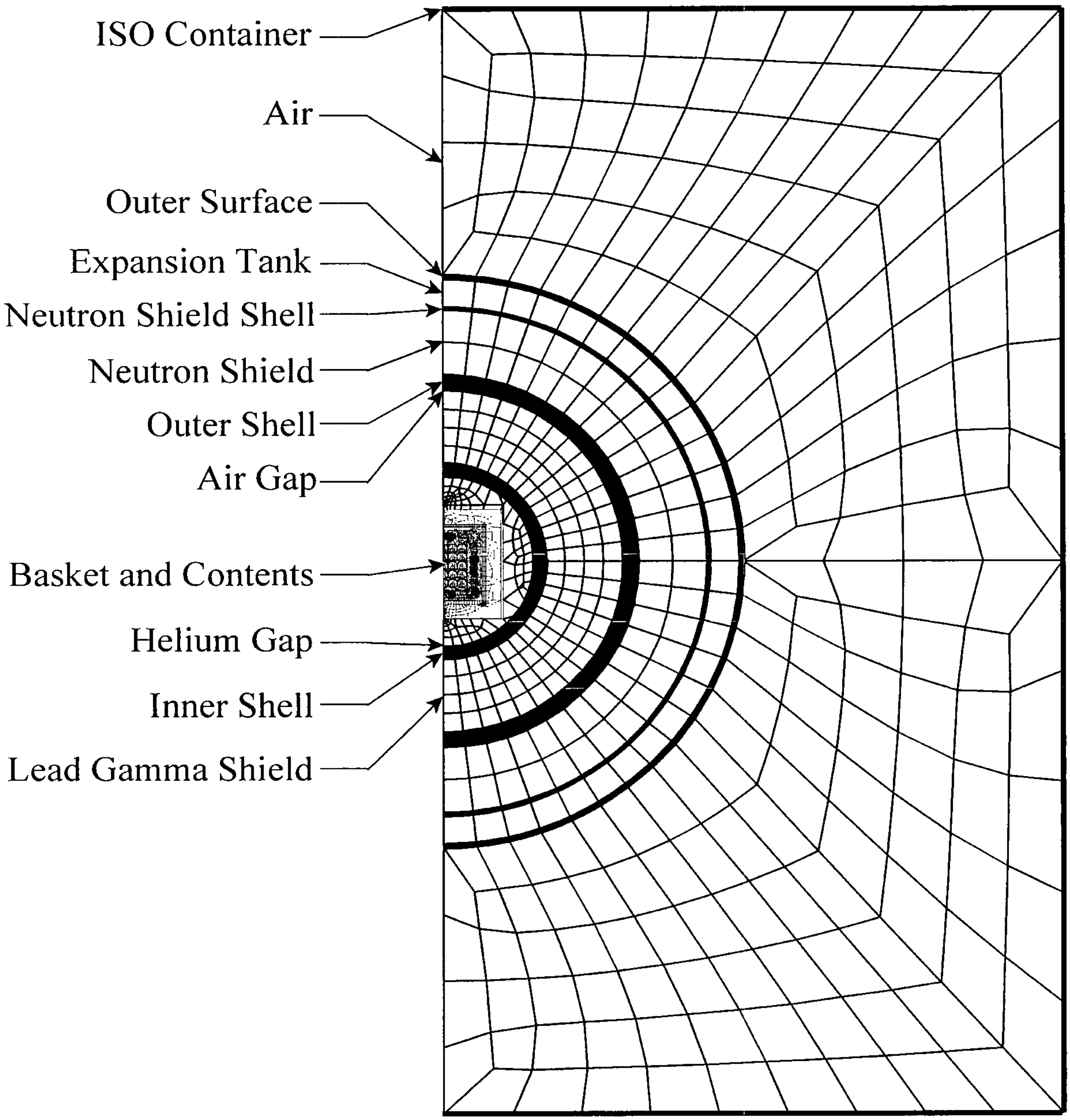
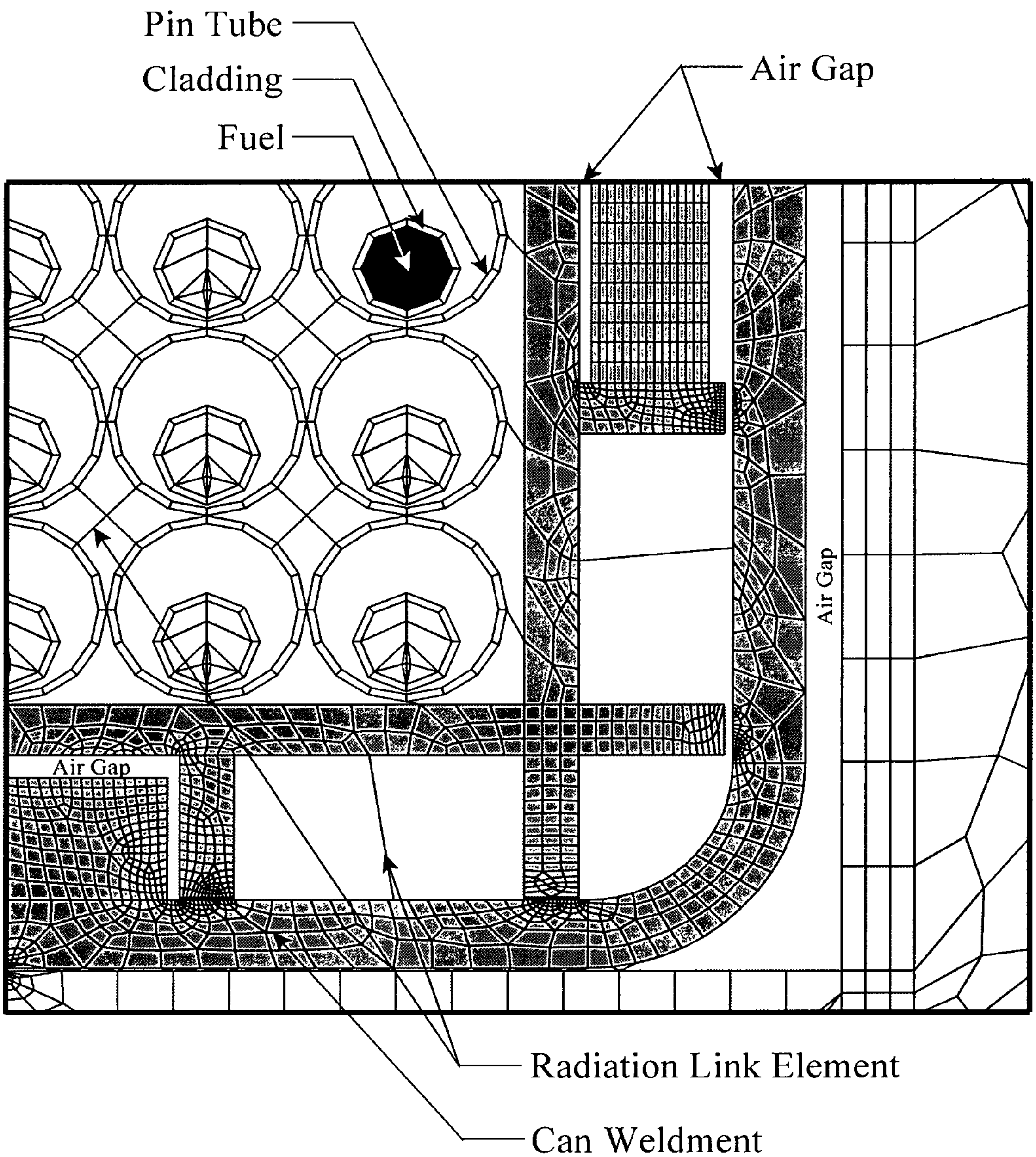


Figure 3.4-11 Close-up of PWR and BWR High Burnup Fuel Rods Normal Condition ANSYS Thermal Model



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)

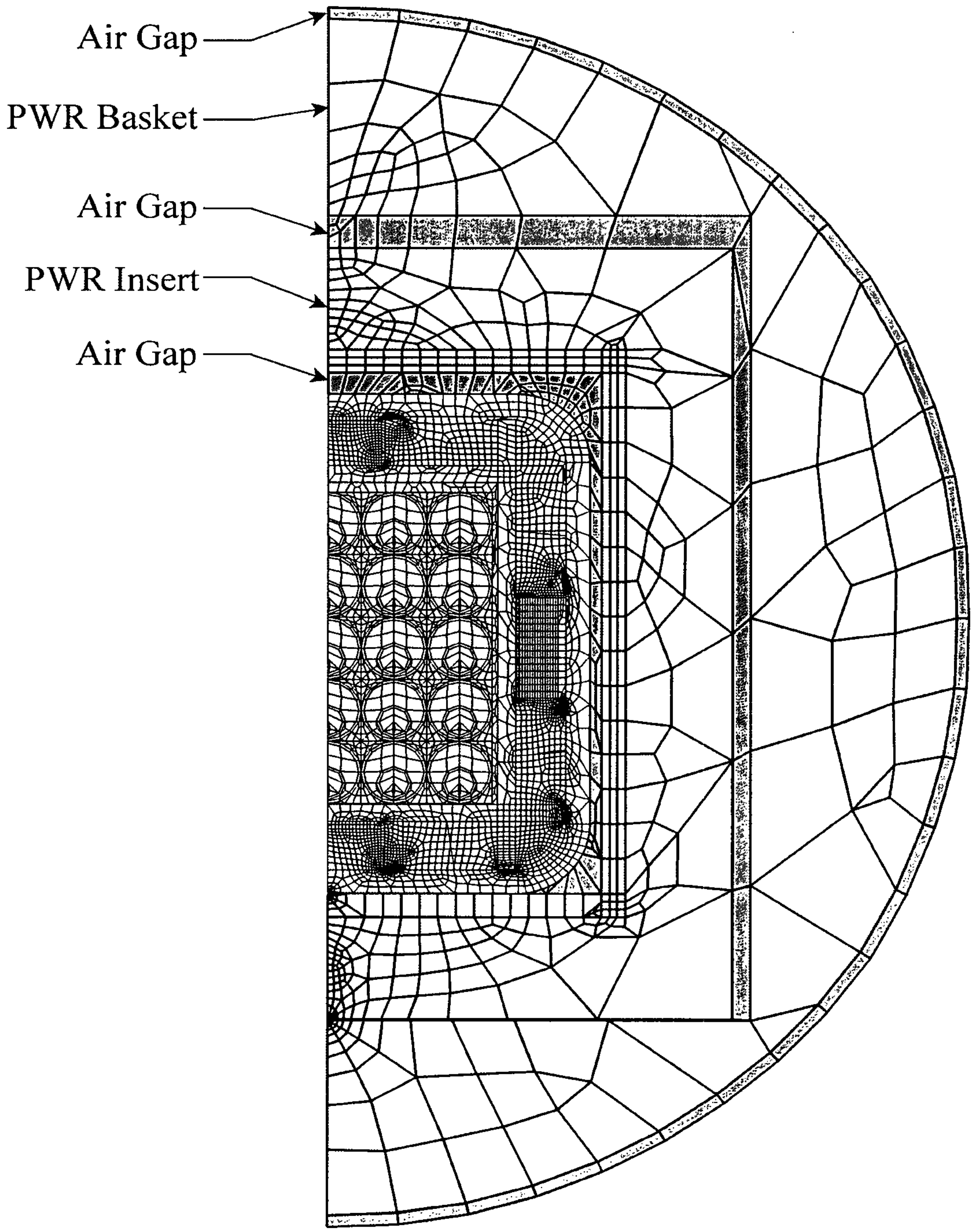
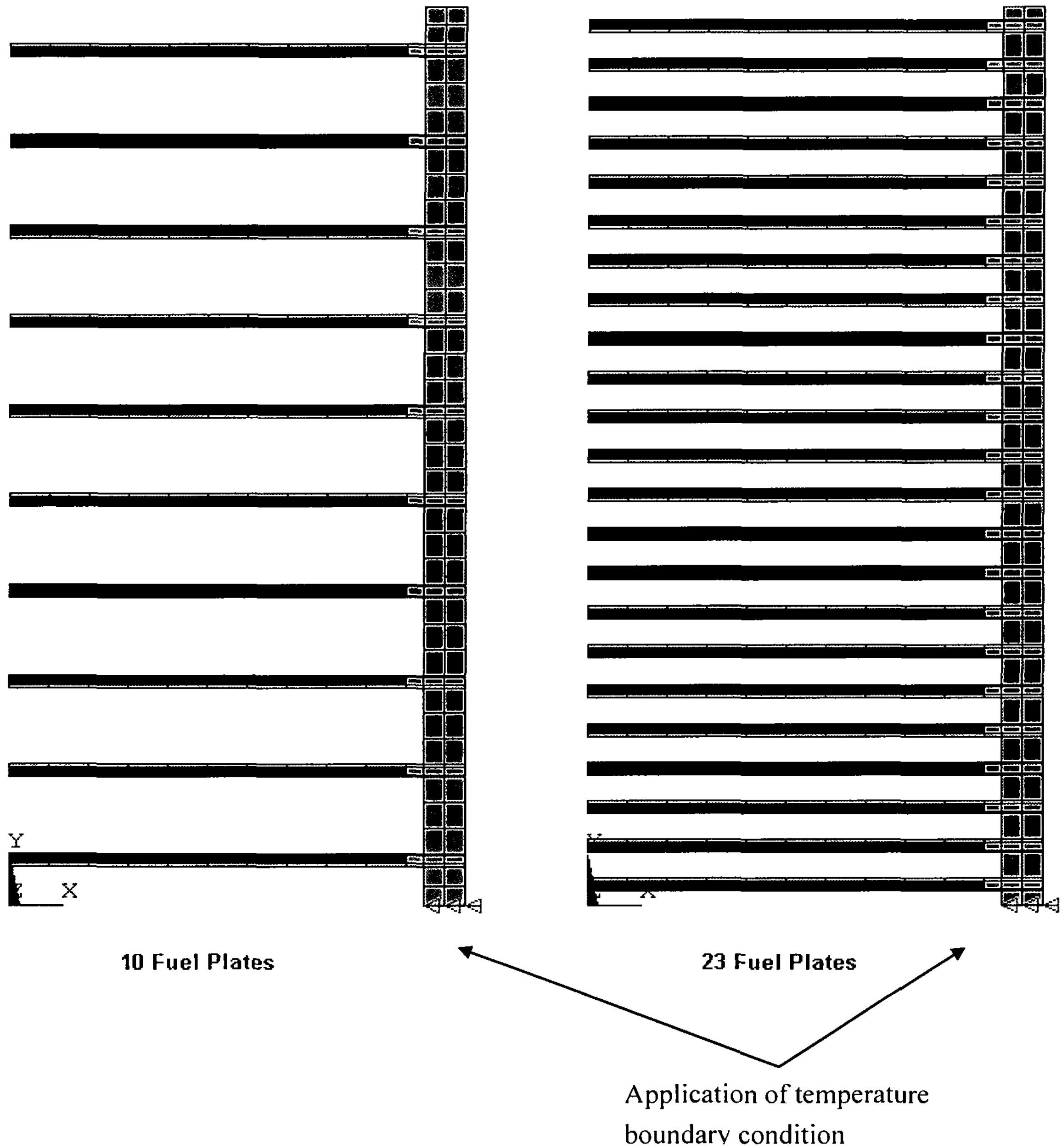


Figure 3.4-13 Finite Element Thermal Model for MTR Fuel Element



(Air elements omitted for clarity)

Figure 3.4-14 Detailed DIDO Basket Module Finite Element Model

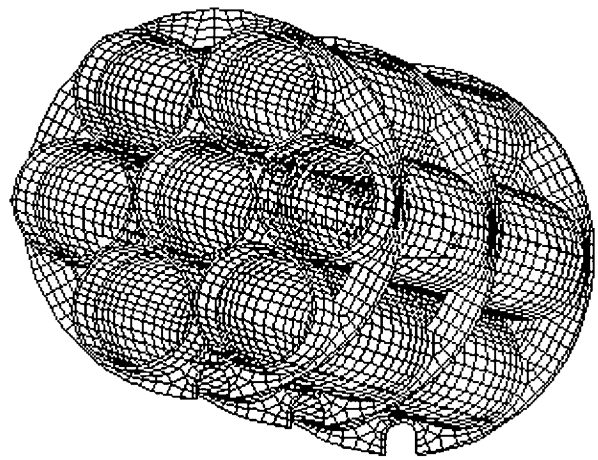


Figure 3.4-15 Detailed DIDO Fuel Assembly Model

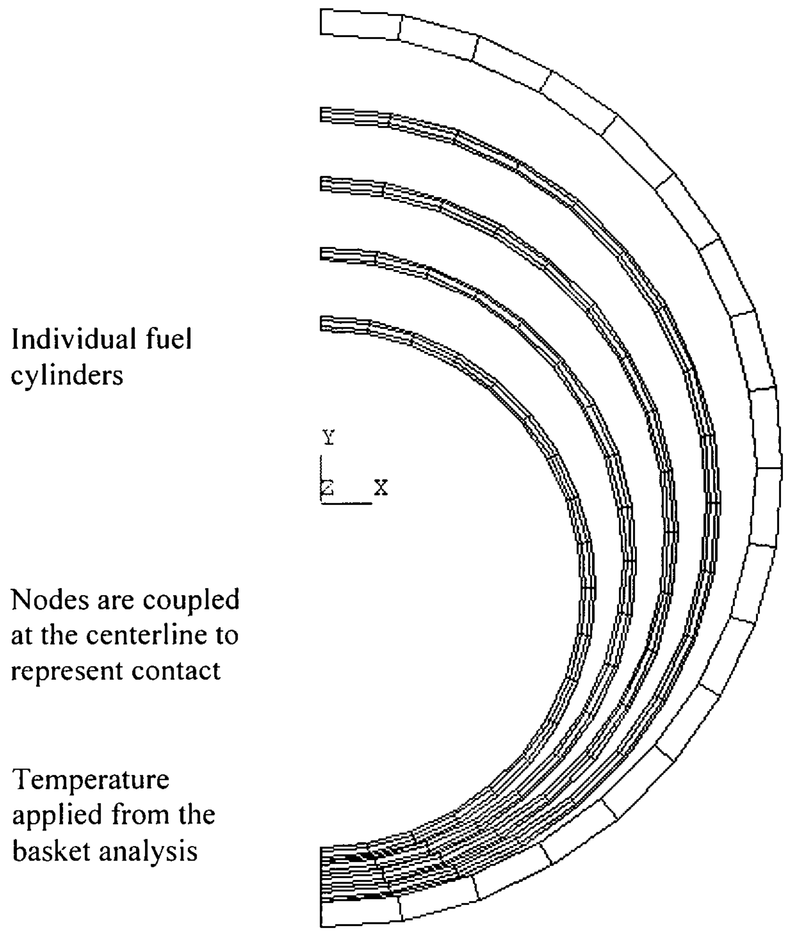


Figure 3.4-16 ANSYS Model for BWR 7 × 7 Fuel Lattice with 25 High Burnup Fuel Rods

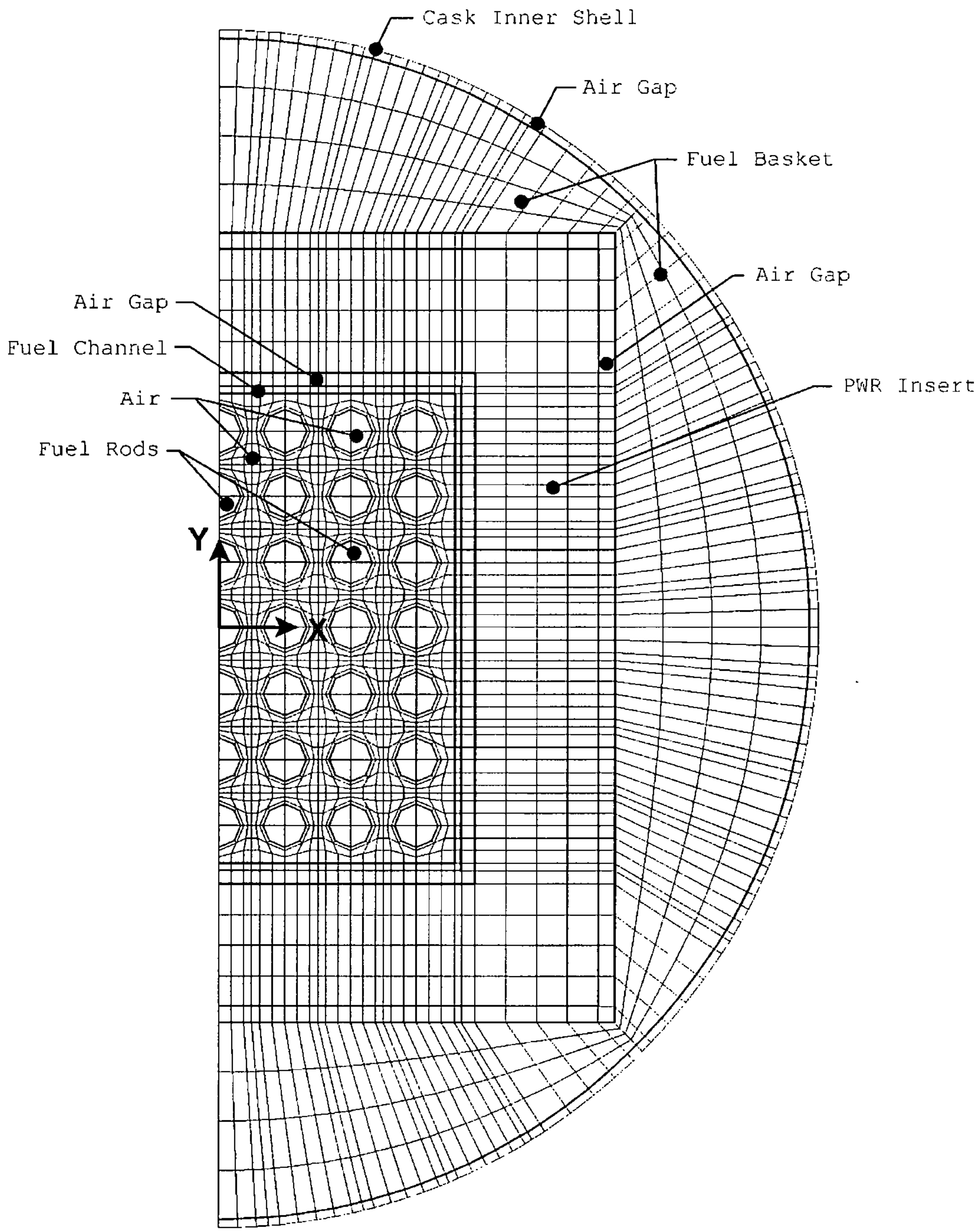


Figure 3.4-17 Fuel Rod Locations in the Thermal Model for Damaged Fuel

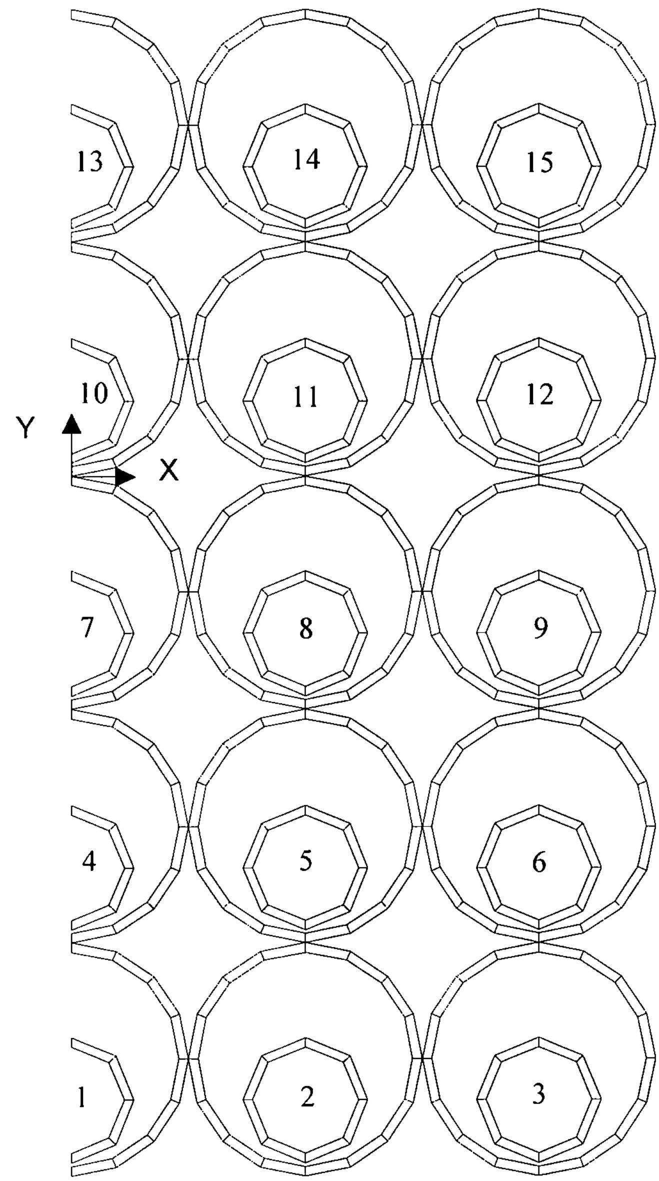
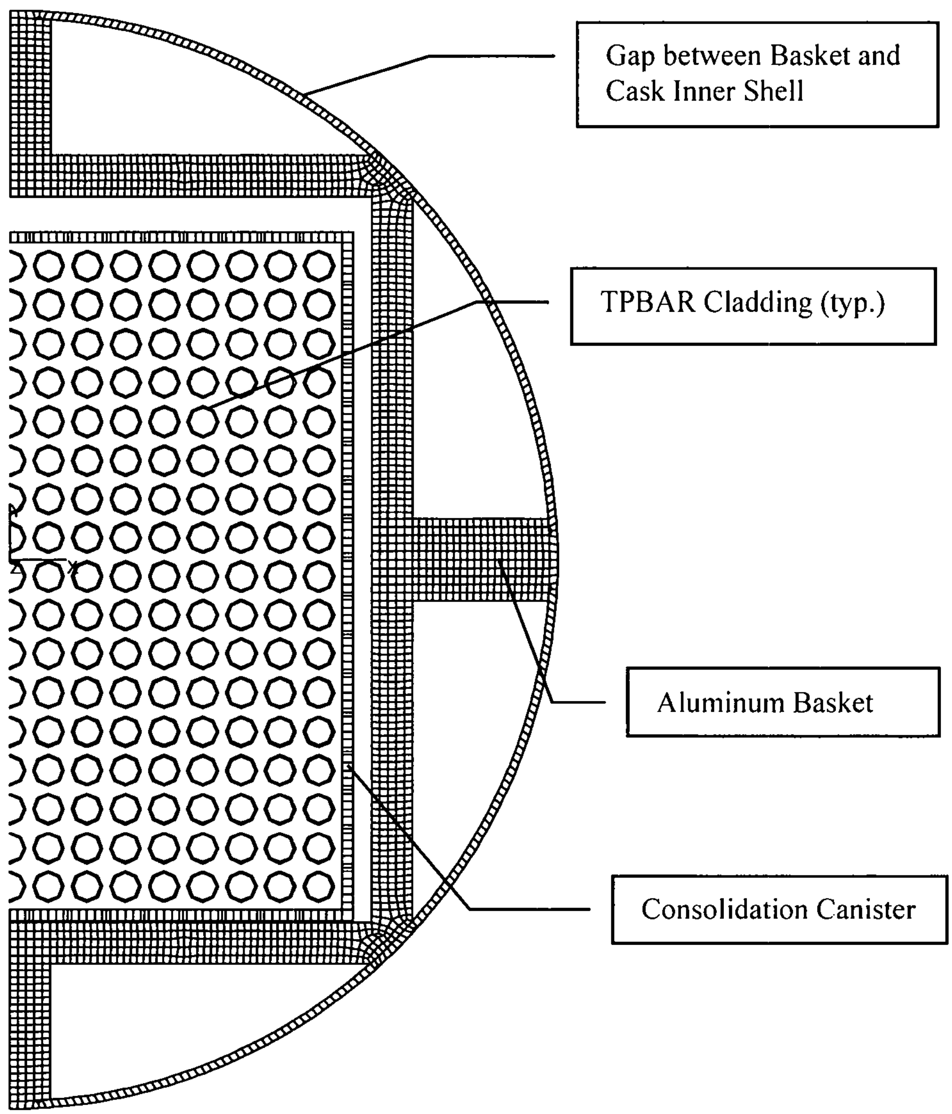


Figure 3.4-18 Finite Element Model for TPBARs



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

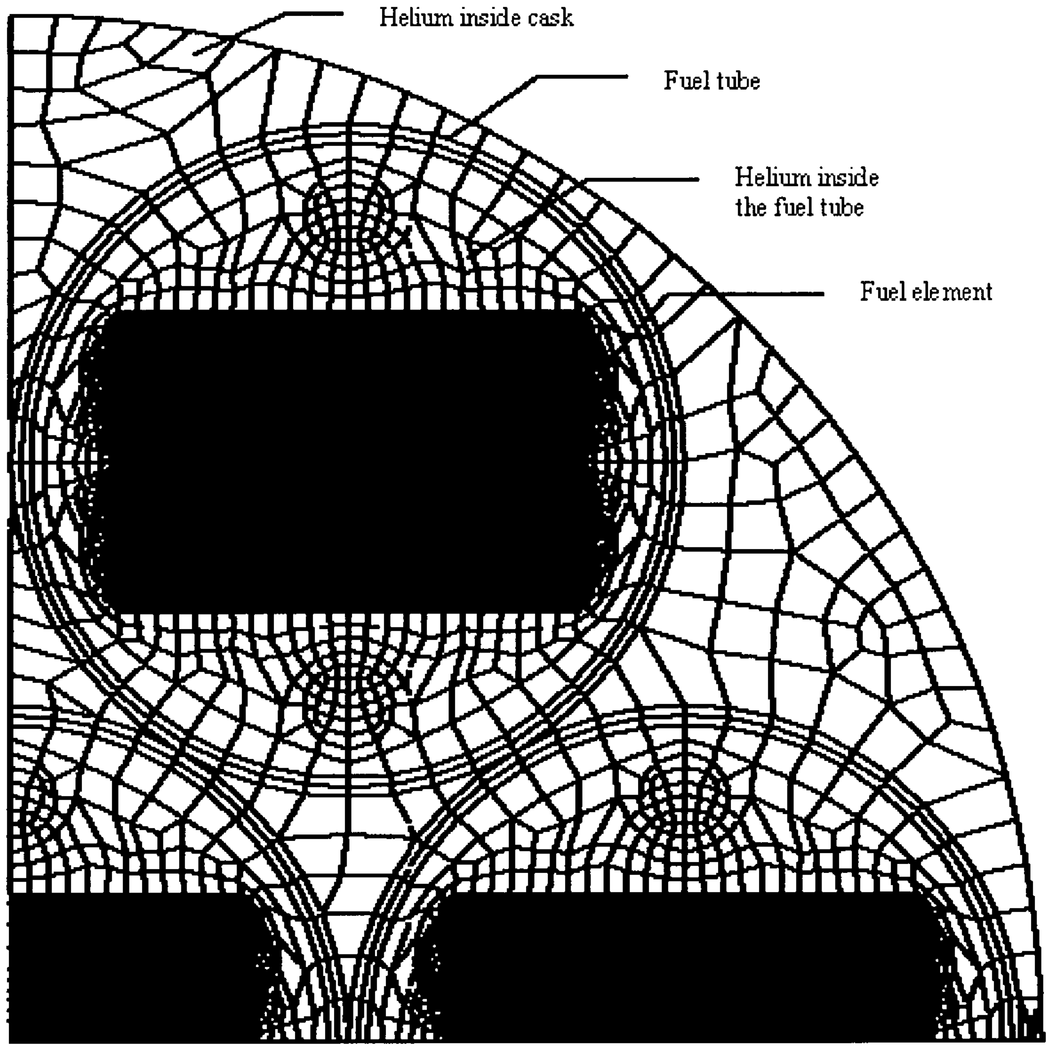


Figure 3.4-20 Finite Element Model for Mark III Spiral Fuel – ANSTO

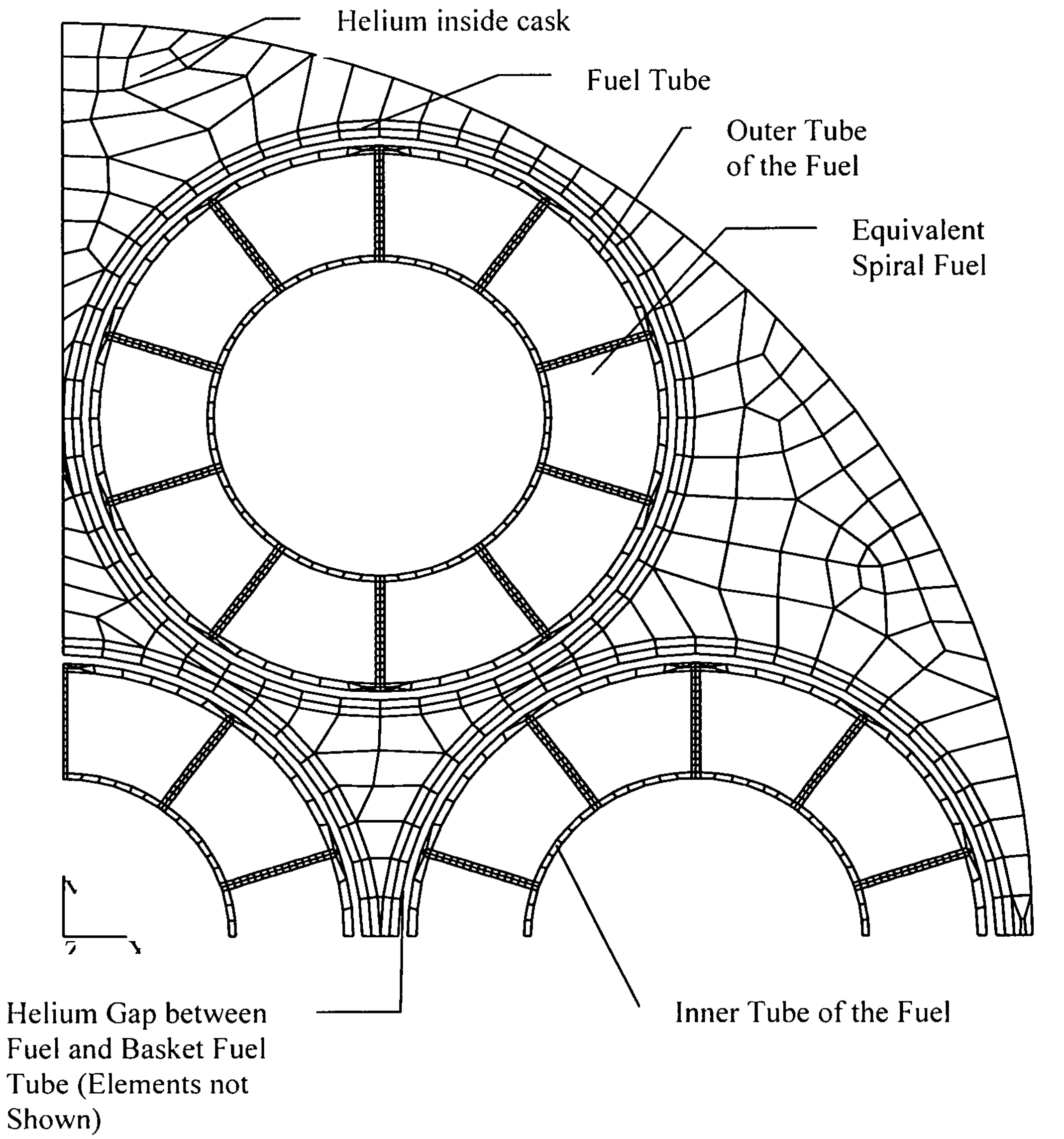
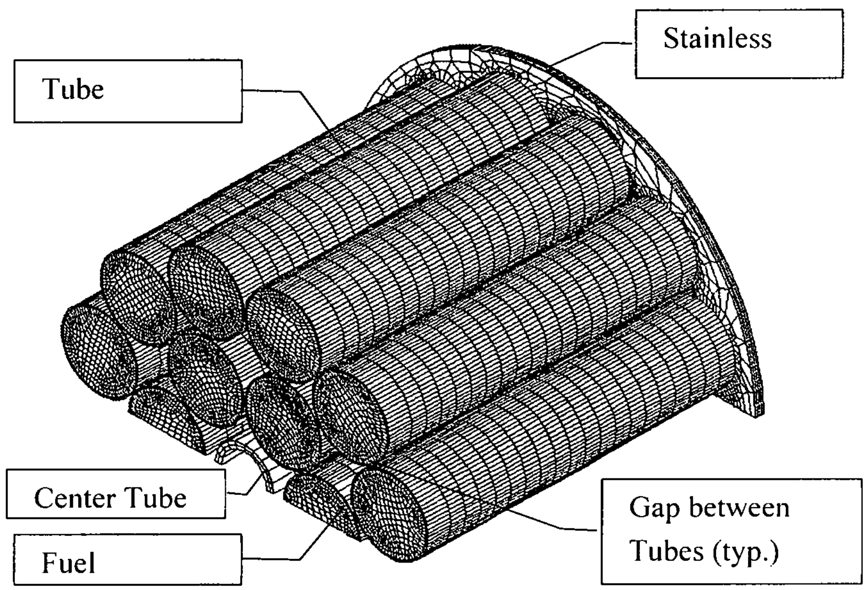


Figure 3.4-21 Finite Element Model for NRU/NRX Fuels in NAC-LWT Cask



Helium elements are not shown for clarity.

Figure 3.4-22 Finite Element Model for HEUNL in NAC-LWT Cask

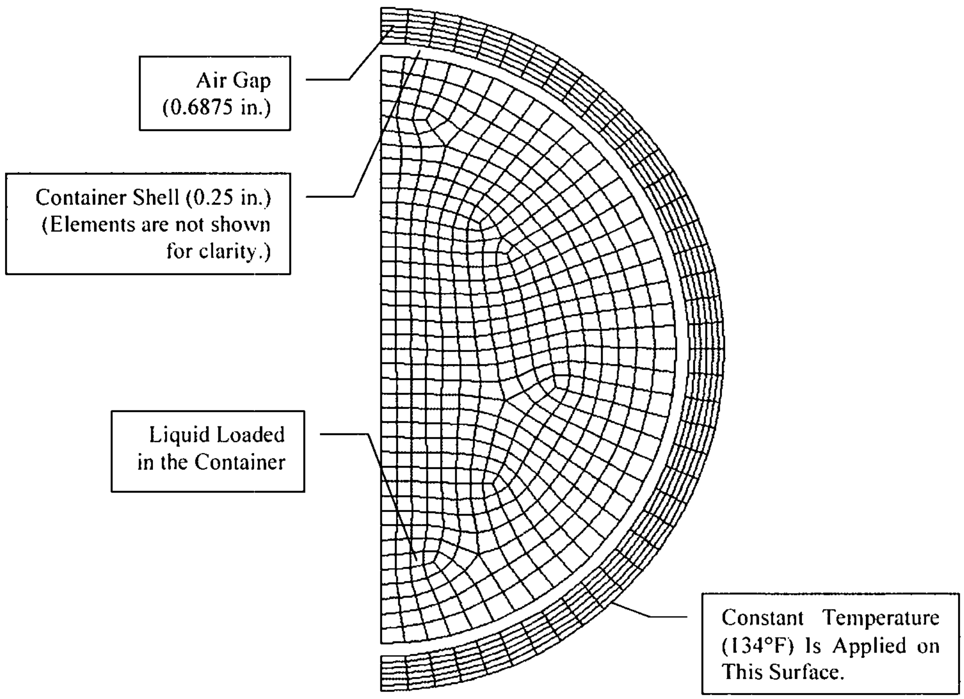


Figure 3.4-23 Finite Element Model for SLOWPOKE Fuel Core Basket

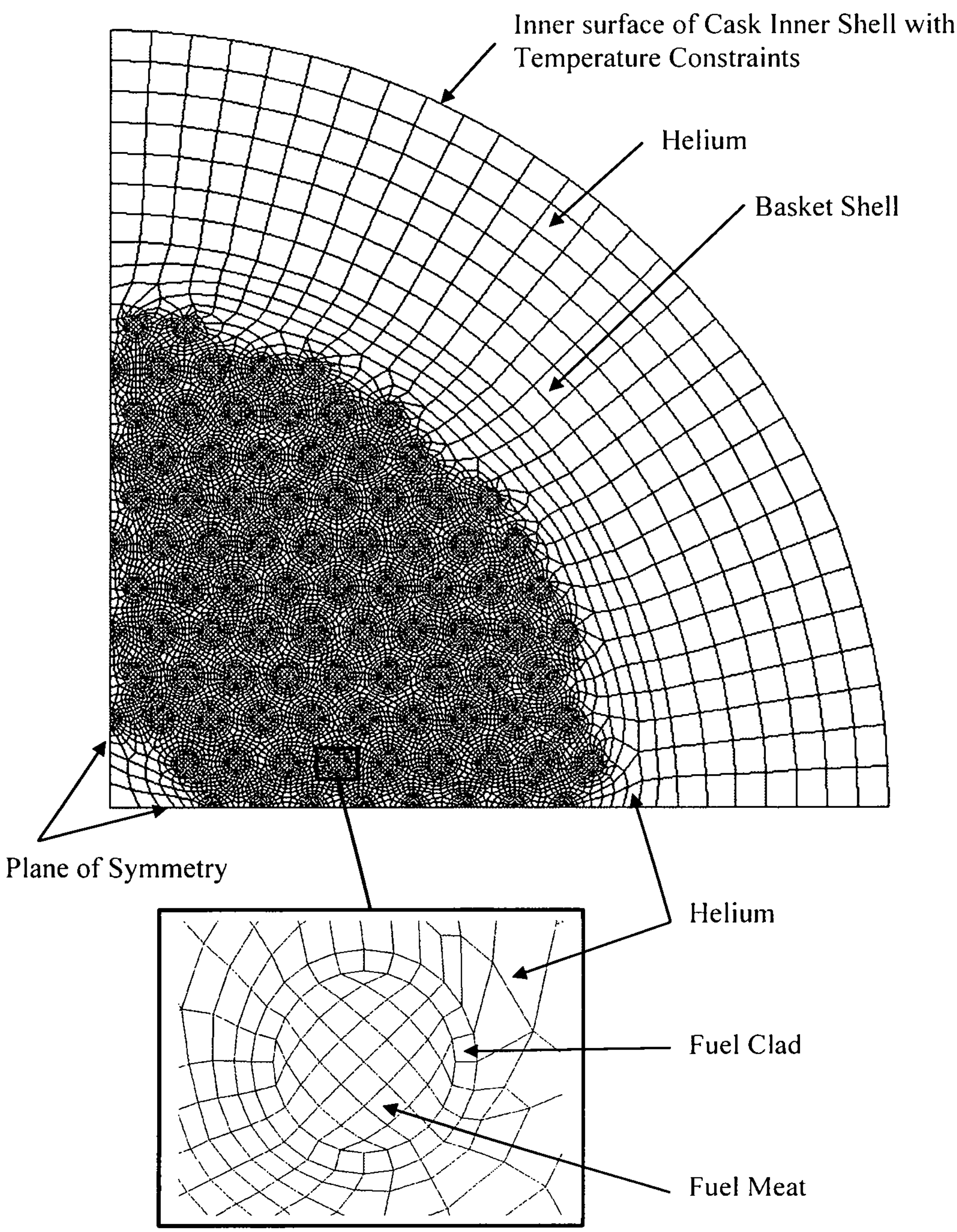


Table 3.4-1 Temperatures for Metallic Fuel Transport

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	Liquid Neutron Shield
Outer Shell	199	Outer Shell
Lead Gamma Shield	212	Lead Gamma Shield
Inner Shell	214	Inner Shell
Basket (maximum)	256	Basket (maximum)
Fuel (maximum)	< 363 [3]	Fuel (maximum)

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. As discussed in Section 3.4.1.3, the loading configuration using the 120-Watt/70-Watt/20-Watt heat load bounds the preferential configuration of 40-Watt maximum heat load per basket slot with the same total heat load of 210-Watt per basket or 1.26 kW for the entire MTR contents.

[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 be 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 Tube2	Fuel Cladding3	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 (liters)	Pressure (atm)
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	Isotope	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

Table 3.4-28 Maximum Component Temperatures – SLOWPOKE Fuel Core

Conditions: 100°F Ambient Temperature
Solar Insolation
45 W Decay Heat Load
NAC-LWT (Transported in an ISO Container)

Component	Temperature (°F)
Liquid Neutron Shield	198[1]
Outer Shell	199[1]
Lead Gamma Shield	212[1]
Inner Shell	214[1]
Basket	239
Fuel	304

[1] Bounding values obtained from Table 3.4-6 for MTR fuel (Condition 1).

Thermal Conductivity	[redacted]
Density	
Specific Heat	

The initial condition for the fire transient analysis is the steady state condition for the normal transport condition, as described in Section 3.4.1.3.1. The LWT cask loaded with HEUNL containers is inside the ISO container. The ambient temperature is 100°F with solar insolation. The gas in the NAC-LWT cask cavity is considered to be air. The same heat load of 12.88W as used for normal transport condition is utilized.

The cask is not in the ISO container as designed for the 30 foot accident drop. The fire is to occur after the drop and pin puncture accidents. At the start of the fire accident, the steam pressure and/or the pin puncture will force the neutron shield to be empty, thus allowing the outer shell of the neutron shield to radiate to the outer surface of the outer cask shell.

[redacted]

[redacted]

3.5.3.19 Evaluation of SLOWPOKE Fuel Core

The maximum heat load for the SLOWPOKE fuel core 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 SLOWPOKE fuel core. 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 basket and fuel temperatures (T_{max}) for the SLOWPOKE fuel core for the accident conditions are determined by adding the increase in steady state temperature from the cask inner shell to the maximum temperature of the component ($\Delta T_{component}$) to the maximum cask inner shell temperature ($T_{inner\ shell}$) obtained from the MTR evaluation. The maximum temperatures of the fuel cladding and basket shell are:

Component	$\Delta T_{component}$ (°F)	$T_{inner\ shell}$(1) (°F)	T_{max} (°F)
Fuel basket	25 = 239(2)-214(2)	334	359
Fuel cladding	90 = 304(2)-214(2)	334	424

(1) Obtained from Table 3.5-2 (Design Basis Heat Loading)
(2) Obtained from Table 3.4-28

The maximum fuel cladding temperature is bounded by those determined for the MTR contents. It is concluded that the structural integrity of the fuel cladding is maintained for the fire accident condition.

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

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

$$P_{\text{fuelrods}} = 79.6 \text{ psia}$$

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

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

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

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

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.

	Free Volume	Pressure		
Description	(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.

$$P_{Accident} = P_{Nonnal} \times \frac{T_{Accident}}{T_{Normals}} = 73.1\text{psia} \times \frac{528\text{K}}{470\text{K}} = 82.1\text{psia} = 67.4\text{psig}$$

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.

3.5.4.8 Maximum Internal Pressure for HEUNL Contents

3.5.4.8.1 Cask Containment

[redacted]

3.5.4.8.2 HEUNL Container

[redacted]

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.

rate for the containment system fabrication verification, periodic and maintenance leak tests described in Section 4.1 and in Chapter 8.

The NAC-LWT leaktight containment criteria, per ANSI N14.5-1997, is 1×10^{-7} ref cm^3/s, which is equivalent to a helium leak rate of less than, or equal to, 2×10^{-7} std cm^3/sec (helium) under test conditions. The minimum test sensitivity is 1×10^{-7} cm^3/s (helium).

Table 4.2-1 Containment Analysis Basis Cask Free Volumes and Pressures

Fuel Type	Pressure (atm) Normal	Accident	Temperature (K)	Free Volume (10^5 cm^3)
PWR	1.99[1]	11.41[1]	517.4	1.471
BWR	1.99[2]	11.4[2]	517.4	1.018
Metallic Fuel	1.99[2]	11.4[2]	405.2	1.018
TRIGA[7]	1.99[2]	11.4[2]	571.4[2]	1.717
GA IFM	N/A[6]	N/A[6]	403.2	3.354
25 PWR Rods – 56% Failed Fuel Fraction	3.0	4.3[5]	588.7[4]	0.9681
25 BWR Rods – 56% Failed Fuel Fraction	3.2	4.5[5]	588.7[4]	0.8932
SLOWPOKE[8]	N/A	N/A	331 (Rods in Canister) 424 (Fuel Core)	N/A

[1] Based on Sections 3.4.4 and 3.5.4, the maximum calculated pressures for the PWR payload are 1.93 atm (28.3 psia) normal condition and 8.56 atm (125.8 psia) accident conditions.

[2] The maximum pressure for the PWR fuel is conservative.

[3] The temperature employed is approximately 4K lower than the maximum fuel clad temperature calculated. The fuel clad temperature is significantly higher than the average gas temperature in the cask.

[4] The normal condition temperature is conservatively applied to the 25 PWR and BWR high burnup rod analysis.

[5] These pressures result from the 100% fuel rod failure plus the design basis fire accident.

[6] Based on the lower temperature and larger free volume of the GA IFM, as compared to the other contents, the pressure, although not explicitly calculated, is lower than that calculated for PWR and BWR fuel.

[7] TRIGA volume and pressure conservatively applied to TRIGA cluster rod analysis. Free volume is higher in the cluster rod configuration.

[8] The SLOWPOKE contents (rods in canister or fuel core) are low mass, low burnup, low heat load materials that produce low temperature and low fission gas quantities. Listed fuel core temperature is conservative as MTR design basis (1.26 kW) inner shell temperatures are applied in the fuel/basket thermal analysis (see Chapter 3). The metal alloy fuel will trap fission gases resulting in minimal gas release. In comparison to other licensed payloads, no significant pressure will result in the cask cavity. No payload specific data was calculated.

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by zirconium alloy-based hardware components such as burnable poison rods (BPRs), provided they are not comprised of highly activated materials (e.g., steel or inconel). The rod lattice is located within a canister placed into an insert located within the NAC-LWT PWR basket. Source terms were calculated using SCALE 5.0 SAS2H, with three-dimensional dose rates calculated using the MCNP5 Monte Carlo transport code.

An analysis of the content condition of 300 production Tritium Producing Burnable Absorber Rods (TPBARs) in a consolidation canister used the ORIGEN-S module of the SCALE package for source terms and the MCNP code package to calculate three-dimensional dose rates. This evaluation is presented in Section 5.3.13 and shows that dose rates are well below regulatory limits for normal and accident conditions. The second TPBAR content condition of 55 segmented TPBARs cooled for a minimum of 90 days is evaluated using the source terms determined by the ORIGEN-S module of the SCALE package. This evaluation readily shows compliance with the previously calculated regulatory dose rates for 300 production TPBARs cooled a minimum of 30 days.

A payload of up to 700 PULSTAR fuel elements is analyzed in Section 5.3.14. Source terms were calculated using SAS2H with three-dimensional dose rates calculated using the MCNP code. PULSTAR fuel elements may be loaded as assemblies in a 5×5 rectangular array; intact elements in a 4×4 fuel rod insert; or intact or damaged elements and nonfuel components of fuel assemblies in a can. Four 7-element MTR basket modules are stacked to form a 28 MTR basket in the cask cavity. The maximum cell loading is 25 elements.

A payload of up to 42 spiral fuel assemblies or 42 MOATA plate bundles in the ANSTO basket is analyzed in Section 5.3.15. Six 7-element ANSTO basket modules are stacked to form a 42-assembly payload in the cask cavity. Source terms were calculated using SAS2H. Due to similarities in the basket design to the DIDO basket and bounding source terms in the DIDO shielding evaluation, no shielding evaluations are required to demonstrate regulatory compliance.

A payload of up to 800 SLOWPOKE fuel elements is analyzed in Section 5.3.20. Source terms were calculated using SCALE 6.1 TRITON with three-dimensional dose rates calculated using MCNP5 v1.3. SLOWPOKE fuel elements may be loaded into the canister assembly with either 5x5 or 4x4 canister inserts, with up to four inserts per canister assembly. The canister assemblies may be loaded into the top two 7-element MTR baskets in a 28 MTR basket stack. The SLOWPOKE elements are found to be well within regulatory limits at the minimum cool time of 14 years.

A payload of up to 18 NRU or NRX fuel assemblies is analyzed in Section 5.3.21. Source terms were calculated using TRITON in SCALE 6.1 with three-dimensional dose rates calculated using the MCNP code. NRU HEU and NRX fuel assemblies are found to be within regulatory limits at

their respective minimum cool times of 18 and 19 years. NRU LEU fuel assemblies are found to be within regulatory compliance at a minimum cool time of 3 years.

A payload of 4 HEUNL containers is analyzed in Section 5.3.20. Source terms were calculated using an inventory of gamma-emitting radionuclides with the actinide and nitrate contents. The ORIGEN-S control module in SCALE 6.1 is used to calculate source spectra. A maximum payload of 64.3 L (17.0 gal) per container is conservatively applied for the source strength (due to void volume in the container that allows HEUNL thermal expansion, actual container capacity is less). Three-dimensional dose rates are calculated using the MCNP v1.60 code. The HEUNL payload is found to be within regulatory limits.

A payload of a SLOWPOKE core, up to 298 rods with 2.81 g ^{235}U per rod with a minimum enrichment of 90 wt% ^{235}U, is analyzed in Section 5.3.23. Source terms were calculated using TRITON in SCALE 6.1 with three-dimensional dose rates calculated using MCNP5 v1.6. The SLOWPOKE core is found to be within regulatory limits after a minimum cool time of 14 days.

5.1 Discussion and Results

The NAC-LWT cask is designed for the safe transport of spent nuclear fuel from various commercial nuclear installations and research reactors.

5.1.1 NAC-LWT Contents

The following contents constitute the design basis for transport in the NAC-LWT cask:

- 1 PWR assembly;
- up to 2 BWR assemblies;
- up to 15 sound metallic fuel rods;
- up to 9 failed metallic fuel rods;
- up to 3 severely failed metallic fuel rods in filters;
- up to 42 MTR fuel elements;
- up to 42 DIDO fuel assemblies;
- up to 25 PWR fuel rods (including up to 14 rods classified as damaged);
- up to 25 BWR fuel rods (including up to 14 rods classified as damaged)[1];
- up to 25 PWR or BWR UO_2 fueled high burnup (up to 80,000 MWd/MTU) rods
- up to 16 PWR MOX or UO_2 rods in any combination (up to 62,500 MWd/MTHM)
- up to 140 TRIGA fuel elements;
- up to 560 TRIGA fuel cluster rods;
- 2 GA IFM packages;
- up to 300 TPBARs (of which two can be prefailed) in a consolidation canister;
- up to 25 TPBARs (of which two can be prefailed) in a rod holder;
- up to 55 TPBARs segmented during 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;
- any combination of individual ANSTO basket modules containing either spiral fuel assemblies or MOATA plate bundles up to a total of 42 assemblies/bundles;
- up to 4,000 lbs of solid, irradiated and contaminated hardware;
- up to 18 NRU (HEU or LEU) or up to 18 NRX fuel assemblies;
- 4 HEUNL containers (empty or filled such that a minimum under filled cavity void of one gallon exists);

[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 800 SLOWPOKE fuel elements contained in up to 8 SLOWPOKE fuel canisters, or
- 1 SLOWPOKE fuel core containing up to 298 SLOWPOKE fuel elements in the SLOWPOKE fuel core basket.

The 25 high burnup PWR and BWR rods may be transported in three configurations: 1) a maximum of 25 intact fuel rods loaded in the rod holder; 2) a maximum of 25 fuel rods with up to 14 damaged fuel rods or rod fragments loaded in the rod holder; and 3) a maximum of 25 intact fuel rods housed in a fuel assembly lattice within the NAC-LWT PWR basket. The fuel assembly lattice may be irradiated up to an equivalent burnup of 80,000 MWd/MTU.

The metallic fuel consists of a single rod of uranium metal clad with aluminum. The intact metallic fuel rods are placed into a transport canister that will hold five intact rods. The cask can hold three transport canisters for a total of 15 intact metallic fuel rods. In the event the metallic fuel has failed or is suspected of having failed, each fuel rod is sealed in its own container. The failed metallic fuel is loaded into either one of the three holes in the metallic fuel basket or into one of the six openings in the failed metallic fuel basket.

MTR research reactor fuel elements are typically 33 to 57 inches long, including lower nozzle and upper handle. The fuel plates typically consist of U-Al, U_3O_8-Al, or USi-Al clad with aluminum. The fuel plates are held in a parallel arrangement with two thick aluminum slotted pieces to form a fuel element. Standard fuel elements have between 10 and 23 fuel plates. The active fuel region is typically 22.75 inches in height, and the fuel meat is typically 0.023-inch thick. The highly enriched uranium (HEU) fuel has been analyzed conservatively with an enrichment of 90 wt % ^{235}U and fuel loading per element up to 380 g ^{235}U, with a separate analysis performed to accommodate up to 460 g ^{235}U. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. The dose rates produced from the design basis 470 g ^{235}U and 640 g ^{235}U LEU and 380 g ^{235}U MEU MTR fuel are bounded by the HEU MTR design basis fuel. Therefore, a mixed loading of LEU, MEU and HEU MTR fuel elements are also bounded by a full HEU MTR fuel element loading.

The source term characteristics of the design basis PWR fuel assembly, BWR fuel assembly, metallic rods, 25 PWR rods, 16 PWR MOX rods, and MTR fuels are given in Table 5.1.1-3. The design basis PWR and BWR fuels require two years of cooling after discharge to meet the neutron and gamma source, and decay heat limits of the cask. The MOX rods require 90 days of cooling. The design basis metallic fuel requires one year cooling. The design basis MTR fuel requires a variable number of years cooling, after discharge, to meet the decay heat limits of the cask. Loading configurations must conform to the limits stated in Section 7.1.5.

DIDO research reactor fuel elements typically consist of U-Al, U_3O_8-Al, or U_3Si_2-Al that is aluminum clad. The fuel elements are held in a concentric arrangement inside an outer

aluminum cylinder to form a fuel assembly. Fuel assemblies have 4 fuel elements. The active fuel region is typically 23.6 inches in height, and the fuel meat is typically 0.026 inch thick. The highly enriched uranium (HEU) fuel has been analyzed with a minimum enrichment of 90 wt % ^{235}U and fuel loading per assembly up to 190 g ^{235}U. Low enriched (LEU) and medium enriched (MEU) assemblies are evaluated at 190 g ^{235}U with minimum enrichments of 19 and 40 wt % ^{235}U, respectively. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. As discussed in Section 5, the dose rates produced from the design basis LEU and MEU DIDO fuel are bounded by the HEU DIDO design basis fuel. Therefore, a mixed loading of LEU, MEU and HEU DIDO fuel assemblies is also bounded by a full HEU DIDO fuel assembly loading.

Two GA IFM Fuel Handling Units (packages) are intended for a single shipment in the NAC-LWT. The first package is composed of Reduced-Enrichment Research and Test Reactor (RERTR) type fuel, which is an Incoloy clad TRIGA fuel. The second is composed of High-Temperature Gas-Cooled Reactor (HTGR) type fuel. Each set of IFM is packaged into stainless steel weld-encapsulated primary and secondary enclosures. Design basis fuel parameters are summarized in Table 5.1.1-1, with fuel characteristics presented in Table 5.1.1-2. Design basis source terms are provided in Table 5.1.1-3. NAC-LWT combined dose rates for GA IFM are bounded by the dose rates for PWR fuel shown in Table 5.1.1-4 through Table 5.1.1-6.

An inventory of up to 300 production TPBARs (of which two can be prefailed) is intended for multiple shipments in the NAC-LWT. A separate content condition is for the transport of up to 55 segmented TPBARs and associated segmentation debris from PIE contained in a waste container. Each TPBAR is a Type 316 stainless steel rod with a 0.381-inch outer diameter and a 0.336-inch inner diameter and a post-irradiation length of approximately 154 inches. Tritium is produced by irradiation of ^{6}Li. Design basis fuel parameters are summarized in Table 5.1.1-1 with characteristics presented in Table 5.1.1-2. Design basis source terms are provided in Table 5.1.1-3. NAC-LWT dose rates for the payloads of up to 300 production TPBARs in a consolidation canister, or up to 55 segmented TPBARs in the waste container, are bounded by the dose rates for PWR fuel shown in Table 5.1.1-4 through Table 5.1.1-6.

Source terms for the high burnup PWR and BWR rods are developed using the SCALE SAS2H code package. Cask dose rates are evaluated using the SCALE SAS1 shielding analysis sequence. Results presented in Section 5.3.8 give the required cool time for PWR and BWR rods as a function of burnup for up to 25 intact fuel rods loaded in the NAC-LWT rod holder. The results presented in Sections 5.3.11 and 5.3.12 demonstrate that dose rate limits are met for the shipment of fuel rods in an irradiated fuel assembly lattice and damaged fuel rods in a rod holder, respectively.

Source terms for the 62,500 MWd/MTHM MOX rods, and the UO_2 rods evaluated in the same section, are developed using the SCALE 5.0 SAS2H code package. Source terms were conservatively calculated for a 70,000 MWd/MTHM burnup. Cask dose rates are evaluated using the MCNP5 Monte Carlo code. Results presented in Section 5.3.17 require the MOX and UO_2 rods to be cooled 90 days prior to shipment (low quality, power grade, MOX fuel requires 120 days to meet heat load limits, but produces dose rates below limits at 90 days).

As can be seen from Table 5.1.1-3, the PWR fuel assembly has the largest source terms and was used as the design basis fuel for shielding analysis of PWR and BWR fuel in the NAC-LWT presented in this section. The metallic fuel shielding analysis is presented in Section 5.3.3. Metallic fuel is shipped with the neutron shield drained and the analysis reflects this. The MTR fuel shielding analysis is presented in Section 5.3.4. The design basis source terms for 25 PWR rods at 60,000 MWd/MTU are well below the design basis PWR fuel assembly. However, the self shielding of 25 PWR rods is less than the 204 rod design basis PWR fuel assembly. Thus, a shielding evaluation of 25 design basis PWR rods is presented in Section 5.3.5. Similarly, the self shielding for either the 25 high burnup PWR or BWR rods at 80,000 MWd/MTU is lower than that of the design basis assemblies. Shielding evaluations for these rods are presented in Sections 5.3.8, 5.3.11 and 5.3.12.

The transport of up to 140 TRIGA fuel elements is evaluated in Section 5.3.6. TRIGA fuel is a solid metal hydride, U-ZrH and may be high enriched (nominal 70 or 93 wt % ^{235}U), or low enriched (nominal 20 wt % ^{235}U). The fuel clad is either aluminum or stainless steel. TRIGA fuel is fabricated in several configurations, as described in Section 1.2.3.1, that vary in weight, active fuel length and overall length. The typical fuel element length and weight is 28.3 inches and 8.82 pounds. The fuel follower control rod element (FFCR) establishes the upper bound weight (13.2 pounds) and length (approximately 45 inches). These elements can only be loaded in the top module of the TRIGA fuel basket. The design basis TRIGA fuel parameters are presented in Table 5.1.1-1 and Table 5.1.1-2. Source term characteristics are presented in Table 5.1.1-3. Cooling time for TRIGA fuel is variable, down to a minimum of 90 days, based on the time required for the decay heat to reach 7.5 watts.

In addition, the transport of TRIGA fuel cluster rods is evaluated in Section 5.3.7. These rods are obtained from the disassembly of the 5×5 (25 rod) arrays comprising the cluster-type TRIGA fuel as shown in Figure 1.2-6. Only the shipment of the fuel cluster rods is analyzed here; no other activated components of the TRIGA cluster assembly are considered for shipment in this analysis. The TRIGA fuel cluster rod is considered to contain a maximum design-basis fuel mass of 50.5 g of U (evaluated at 92 wt % ^{235}U) for HEU cluster rods and 289.4 g of U (19 wt % ^{235}U) for LEU elements. Both elements are modeled with a nominal H to Zr ratio of 1.6. A

manufacturing tolerance produced H to Zr ratio of 1.7 is evaluated in Chapter 6 for criticality. The manufacturing tolerances have no significant effect on the shielding evaluations. The HEU fuel contains 10 wt % uranium in the U-ZrH_x fuel meat, while the LEU material contains 45 wt % uranium. The rods are clad in Incoloy 800 and contain upper and lower stainless steel end plugs with a mass of approximately 60.5 g each. For shipment, each rod is placed inside an aluminum tube (ID 0.625 in, OD 0.750 in), with 16 rods occupying each LWT basket opening for a total of up to 112 rods per basket or 560 rods per cask.

The basis for the dose rate evaluation of the TRIGA fuel cluster rods is a source term and one-dimensional shielding analysis in which the minimum cooling time required for the dose rates produced by the TRIGA fuel cluster rods to fall below the dose rates produced by the design basis TRIGA fuel elements. Cooling time results are determined at a large number of fuel burnup values (at approximately every 2.5% increment in ^{235}U depletion).

PULSTAR fuel elements are zirconium alloy-clad UO_2 pellets with a physical design characteristic as listed in Table 5.1.1-1 and Table 5.1.1-2. PULSTAR fuel assemblies are a 5×5 rectangular array of elements surrounded by a zirconium alloy box, with aluminum upper and lower fittings. The element pitch is nominally 0.524 × 0.606 inch. PULSTAR fuel elements are analyzed at a loading of 32 grams ^{235}U per element, an initial enrichment of 6 wt % ^{235}U, and 45% ^{235}U burnup. For conservatism, a cool time of one year from discharge is employed in the shielding analysis; a cool time of at least 1.5 years is required to meet the basket cell heat load limit of 30 W. Source term characteristics are presented in Table 5.1.1-3.

Spiral fuel assemblies typically consist of 10 curved plates (also referred to as elements) of metallic U-Al fuel meat that is aluminum clad. The fuel elements are held in a spiral arrangement between an inner and outer aluminum cylinder to form a fuel assembly. The active fuel region is typically 60.325 cm in height, and the fuel meat is typically 0.061 cm thick. The elements are nominally enriched to 80 wt % ^{235}U and were conservatively evaluated at 75 wt % ^{235}U. Maximum fuel loading per assembly is evaluated at 160 g ^{235}U. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. Applying MEU DIDO fuel assembly minimum cool time curves, which are based on a 40 wt % ^{235}U enriched 190 g ^{235}U fuel assembly, to the spiral fuel elements produces source terms that are bounded by the DIDO MEU fuel. Given similar basket designs, the dose rates produced by the spiral fuel elements are bounded by the MEU DIDO evaluation set.

MOATA fuel bundles consist of a maximum of 14 flat MTR type fuel plates. The fuel plates are composed of a metallic U-Al fuel meat that is aluminum clad. The fuel elements are held in place with aluminum outer plates and pins through the top and bottom of the plate stack in their shipment configuration. The plates are held in a typical MTR plate (12 plates per assembly)

with a comb side plate configuration during reactor operations. The active fuel region is typically 58.4 cm in height, and the fuel meat is typically 0.1016-cm thick. The elements are nominally enriched to 90 wt % ^{235}U and were conservatively evaluated at 80 wt % ^{235}U. Maximum fuel loading per plate is evaluated at 25 g ^{235}U (nominal loading is 22 g ^{235}U). The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. The gamma radiation source for the 14 fuel plate bundle is approximately 2% of the DIDO MEU assembly. Since the basket designs are similar, the dose rates produced by the plate bundle are bounded by the MEU DIDO evaluation set.

A payload of up to 18 NRU or up to 18 NRX assemblies is analyzed in Section 5.3.21. NRU fuel assemblies are 12 fuel pins arranged in an annular configuration (9 outer pins, 3 inner). The NRU reactor was operated with HEU fuel (93.2 wt % ^{235}U) until 1992 when it was converted to LEU (19.75 wt % ^{235}U). The NRU fuel consists of either U-Al (HEU) or U_3-Si-Al (LEU) with aluminum clad. The HEU NRU fuel has been analyzed for a loading of 43.7 g ^{235}U per pin at a minimum enrichment of 91.0 wt % ^{235}U. The LEU NRU fuel has been analyzed for a loading of 43.7 g ^{235}U per pin at a minimum enrichment of 19.0 wt % ^{235}U. NRX fuel assemblies are 7 fuel pins arranged in an annular configuration (6 outer pins, 1 central pin). The NRX reactor was operated with HEU fuel (93.1 wt % ^{235}U) until shutdown in 1993. The NRX fuel consists of U-Al alloy with aluminum clad. The NRX fuel has been analyzed for a loading of 79.2 g ^{235}U per pin at a minimum enrichment of 91.0 wt % ^{235}U.

The HEUNL material consists of highly enriched uranyl nitrate, various other nitrates, and water. An inventory of gamma-emitting radionuclides is used with the actinide and nitrate contents to calculate gamma and neutron source terms. A bounding HEUNL volume of 64.3 L (17.0 gal) per container is applied for the source strength (due to void volume in the container that allows HEUNL thermal expansion, actual container capacity is less). The design basis material parameters are provided in Table 5.1.1-1. The material characteristics are presented in Table 5.1.1-2. Source terms are presented in Table 5.1.1-3.

A payload up to 800 SLOWPOKE fuel elements contained in SLOWPOKE fuel canisters is evaluated in Section 5.3.20. Up to 100 fuel elements are placed in a SLOWPOKE fuel canister with up to four canisters placed into an MTR-28 basket module. The top two basket modules may contain SLOWPOKE fuel canisters. SLOWPOKE fuel elements are U-AL alloy and HEU. A conservative 90 wt% ^{235}U enrichment is evaluated with a maximum U mass of 3.1 gram per rod. Maximum burnup is 30 GWd/MTU or 4.5% ^{235}U depletion.

A payload of 1 SLOWPOKE fuel core is analyzed in Section 5.3.23. The fuel core contains up to 298 SLOWPOKE fuel elements (rods). The core is placed in a SLOWPOKE fuel core basket which is placed on a stack of empty intermediate and bottom MTR-42 basket modules which

serve as spacers. A maximum ^{235}U mass of 2.81 g per rod is evaluated. Fuel core ^{235}U content evaluated is 837 gram at a core average depletion of 2.12 % ^{235}U.

The shield materials are selected and arranged to minimize cask weight while maintaining overall shield effectiveness. Lead and steel are chosen as effective gamma radiation shields, and a water tank on the outside of the cask is provided to efficiently moderate and absorb the neutron radiation.

The total neutron and gamma dose rates calculated for the normal operations conditions are shown in Table 5.1.1-4. Note that the maximum dose rate is on the cask lid surfaces at the top end of the cask and does not exceed the design limit of 200 mrem/hour for the surface of the cask. The 10 CFR 71 limits of 10 mrem/hour at two meters from the cask surface and the design limit of 200 mrem/hour on the cask surface are met. Table 5.1.1-4 contains the total dose rates for the hypothetical accident conditions. These dose rates are well under the 49 CFR 173 limit of 1000 mrem/hour at one meter from the cask surface. The dose rates for the lead slump accident are shown in Table 5.1.1-5. These dose rates show that even with the lead slumped, the hypothetical accident dose rate limits have not been exceeded and the cask is safe for transport.

The cask surface fuel centerline normal operations and hypothetical accident dose rates calculated include neutrons and gammas originating from the fuel, neutrons and gammas scattered from the ground and secondary gammas resulting from neutron capture in the neutron shield. All of the other dose locations also include the contribution from the ^{60}Co in the end-fittings.

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel

Fuel Type	-	PWR, Assembly
	-	3.7 wt % ^{235}U maximum initial enrichment
	-	35,000 MWd/MTU maximum burnup
	-	2.5 kW per assembly maximum decay heat
	-	2 years (or more) decay time after reactor discharge
Fuel Form	-	Intact assemblies
Quantity	-	1 design basis fuel assembly
Source of Fuel	-	Commercial PWR nuclear power reactors
Transport Index	-	35
Fuel Type	-	BWR, Assembly
	-	4.0 wt % ^{235}U maximum initial enrichment
	-	30,000 MWd/MTU maximum burnup
	-	1.1 kW per assembly maximum decay heat, 2.2 kW per cask for 2 assemblies
	-	2 years (or more) decay time after reactor discharge
Fuel Form	-	Intact assemblies
Quantity	-	2 design basis fuel assemblies
Source of Fuel	-	Commercial BWR nuclear power reactors
Transport Index	-	35
Fuel Type		High Burnup PWR or BWR rods
	-	5.0 wt % maximum ^{235}U initial enrichment
	-	80,000 MWd/MTU maximum average burnup
	-	2.3 kW /cask maximum decay heat
	-	Minimum cool time dependent on burnup (See Table 5.3.8-29)
Fuel Form	-	Intact rods in a fuel assembly lattice or rod holder and intact rods with up to 14 fuel rods classified as damaged in a rod holder
Quantity	-	Up to 25
Source of Fuel	-	Commercial PWR or BWR nuclear power reactor
Transport Index	-	36 (intact rods) 28 (intact rods in a fuel assembly lattice) 37 (intact rods with 14 rods classified as damaged)
Fuel Type	-	Uranium metal fuel rods
	-	Natural wt % ^{235}U
	-	1,600 MWd/MTU maximum burnup
	-	0.0357 kW per sound rod maximum decay heat, 0.54 kW per cask for 15 sound fuel rods
	-	1 year (or more) decay time after reactor discharge
Fuel Form	-	Intact or encapsulated failed fuel rods
Quantity	-	15 design basis fuel rods, or 6 design basis failed fuel rods
Source of Fuel	-	Research reactors
Transport Index	-	25

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

Fuel Type	-	PULSTAR Fuel Elements
	-	6 wt % ^{235}U
	-	32 grams ^{235}U per element
	-	45% ^{235}U depletion (burnup)
	-	210 W per basket decay heat (30 watts per basket cell) × 4 = 840W Minimum cool time from discharge of 1.5 years[3]
Fuel Form	-	Intact assemblies; intact elements in fuel rod insert; canned intact or failed elements
Quantity	-	Up to 700 elements (25 elements per cell)
Sources of Fuel	-	Research reactors
Transport Index	-	25
Fuel Type	-	Spiral Fuel Assemblies
	-	75 wt % ^{235}U, maximum burnup variable up to 70% ^{235}U depletion
	-	18 W per assembly , 126 W per basket (at given cool time and burnup limits, maximum heat load is 15.7 W per assembly or 110 W per basket) Variable cool time down to 270 days using the procedure in Section 7.1.4 for 18 W DIDO MEU fuel
Fuel Form	-	Intact aluminum clad fuel plates within concentric aluminum inner and outer shells
Quantity	-	Up to 42 fuel assemblies
Sources of Fuel	-	Research reactors
Transport Index	-	40.1 (applied bounding MEU DIDO limit)
Fuel Type	-	MOATA Plate Bundles
	-	80 wt % ^{235}U, maximum burnup variable up to a 30,000 MWd/MTU or 4.1% ^{235}U depletion
Fuel Form	-	Intact aluminum-clad fuel plates
Quantity	-	Up to 42 bundles
Sources of Fuel	-	Research reactors
Transport Index	-	40.1 (applied bounding MEU DIDO limit)
Fuel Type	-	PWR MOX or UO_2 rods (including up to 9 BPRAs)
	-	5.0 wt % maximum ^{235}U initial enrichment for UO_2 rods 7.0 wt % fissile Pu for MOX rods
	-	62,500 MWd/MTHM maximum average burnup
	-	2.3 kW/cask maximum decay heat
	-	Minimum cool time 90 days (120 days for Power Grade MOX)
Fuel Form	-	Undamaged rods in a rod holder
Quantity	-	Up to 16 (any combination of UO_2 or MOX) fuel rods plus up to 9 BPRAs
Source of Fuel	-	Commercial PWR nuclear power reactor
Transport Index	-	28

[3] Conservatively evaluated at a one-year cool time and 38 watts per basket cell.

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

Fuel Type	-	NRU HEU Fuel Assemblies
	-	91.0 wt % minimum ^{235}U initial enrichment
	-	364 MWd maximum burnup (approximately 87.4% ^{235}U depletion)
	-	162 W/cask maximum decay heat
	-	Minimum cool time 19 years
Fuel Form	-	Undamaged or collapsed assemblies
Quantity	-	Up to 18 fuel assemblies
Source of Fuel	-	NRU reactor
Transport Index	-	2.3
Fuel Type	-	NRU LEU Fuel Assemblies
	-	19.0 wt % minimum ^{235}U initial enrichment
	-	363 MWd maximum burnup (approximately 83.6% ^{235}U depletion)
	-	641 W/cask maximum decay heat
	-	Minimum cool time 3 years
Fuel Form	-	Undamaged or collapsed assemblies
Quantity	-	Up to 18 fuel assemblies
Source of Fuel	-	NRU reactor
Transport Index	-	30.8
Fuel Type	-	NRX Fuel Assemblies
	-	91.0 wt % minimum ^{235}U initial enrichment
	-	85.1% ^{235}U maximum depletion (at reactor power of 42 MW)
	-	171 W/cask maximum decay heat
	-	Minimum cool time 18 years
Fuel Form	-	Undamaged or collapsed assemblies
Quantity	-	Up to 18 fuel assemblies
Source of Fuel	-	NRX reactor
Transport Index	-	3.0
Type	-	HEUNL
	-	9.0 Ci/L Curie content for gamma-emitting inventory
Form	-	HEUNL material in HEUNL container
Quantity	-	4 HEUNL containers
Source	-	Radioisotope Production
Transport Index	-	1.5
Fuel Type	-	SLOWPOKE Fuel Rods (in Canister)
	-	90.0 wt % minimum ^{235}U initial enrichment
	-	4.5% ^{235}U maximum depletion (30 GWd/MTU)
	-	2.17 W/cask maximum decay heat
	-	Minimum cool time 14 years
Fuel Form	-	Undamaged or damaged, including debris
Quantity	-	Up to 800
Source of Fuel	-	SLOWPOKE reactor
Transport Index	-	1

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

Fuel Type	-	SLOWPOKE Fuel Core (up to 298 fuel rods)
	-	90.0 wt % minimum ^{235}U initial enrichment
	-	2.1% ^{235}U maximum depletion
	-	56.6 W/cask maximum decay heat
	-	Minimum cool time 2 weeks
Fuel Form	-	Undamaged
Quantity	-	1 (One)
Source of Fuel	-	SLOWPOKE reactor
Transport Index	-	15.2

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation

Parameter	PWR	BWR	Metallic	MTR (HEU)	MTR (MEU)	MTR (LEU)	DIDO
Assembly Array	15 × 15	7 × 7	N/A	Parallel Plates	Parallel Plates	Parallel Plates	Fuel Tubes
Assembly or Element Weight (lbs)	1650	750	1805 (15 rods)	13.0 (max)	13.0 (max)	13.0 (max)	15.0 (max)
Assembly/Element/Rod Length (in)	162	176	120.5	25.23[5]	26.14[5]	26.14[5]	24.6
Active Fuel Length (in)	144	144	120.0	24.80	25.59	25.59	23.6
No. Rods per Assembly	204	49	N/A	N/A	N/A	N/A	N/A
No. of Plates per Element	N/A	N/A	N/A	23	23	23	4
Fuel Rod Diameter/Plate Thickness (in)	0.422	0.563	1.36	0.050	0.050	0.050	0.059
Clad Material	Zr-4	Zr-4	Al	Al	Al	Al	Al
Clad Thickness (in)	0.0243	0.032	0.080	0.0150	0.0150	0.0150	0.0167
Pellet Diameter/Meat Thickness (in)	0.3659	0.487	1.36	0.020	0.020	0.020	0.026
Fuel Material	UO_2	UO_2	U metal	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al
Percent Theoretical Density	95	95	100	N/A	N/A	N/A	N/A
Enrichment (wt % ^{235}U)	3.7	4.0	Natural	90[8]	40[8]	19[8]	90 (HEU) 400 (MEU) 199 (LEU)
Maximum Average Burnup (MWd/MTU)	35,000	30,000	1,600	Variable up to 660,000[2,9]	Variable up to 293,300[2]	Variable up to 139,300[2]	Variable up to 577,460 (HEU) 256,650 (MEU) 121,910 (LEU)
Minimum Cool Time	2 Years	2 Years	1 Year	Variable down to 90 days[2]	Variable down to 90 days[2]	Variable down to 90 days[2]	Variable down to 180 days[10]
U Weight (kg/assembly)	475	198	N/A	N/A	N/A	N/A	N/A
U Weight (kg/element)	N/A	N/A	54.5	0.422 0.511	0.950	3.3684	0.2111 (HEU) 0.4750 (MEU) 1.0000 (LEU)
UO_2 Weight (kg/assembly)	538.9	224.3	N/A	N/A	N/A	N/A	N/A

Notes:
1. Up to 2 of the PWR rods may have a maximum average burnup of 65,000 MWd/MTU.
2. Variable cool time down to 90 days using the procedure in Section 7.1.4.
3. Design Basis normal condition source term is for ACPR fuel with 86,100 MWd/MTU (50% ^{235}U depletion) and accident condition source term is for FLIP-LEU-II with 151,100 MWd/MTU (80% ^{235}U depletion).
4. Detailed fuel data is presented in Tables 1.2-1 and 6.2.5-1. The values presented here are the physical values for the bounding source terms of the ACPR and FLIP-LEU-II fuel types.
5. For MTR fuel assemblies, which are cut to remove non-fuel bearing hardware prior to transport, a nominal 0.28 inch of nonfuel hardware will remain above and below the active fuel region to allow for fuel handling operations
6. Minimum cool time varies with burnup such that maximum decay heat is 1.875 watts/rod.
7. Varies with burnup – see Table 5.3.8-29.
8. For the shielding evaluation, lower values are conservatively assumed.
9. Maximum burnup of 660,000 MWd/MTU for 380 g ^{235}U and 577,500 MWd/MTU for 460 g ^{235}U.
10. Variable cool time down to 180 days using the procedure in Section 7.1.4.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	PWR Rods	High B/U PWR Rods	High B/U BWR Rods	PWR MOX/UO_2 Rods	TRIGA[4]	TRIGA Fuel Cluster Rods	TPBARs
Assembly Array	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Assembly or Element Weight (lbs)	N/A	N/A	N/A	N/A	8.82 (nominal) 13.2 (max)		2.655
Assembly/Element/Rod Length (in)	162	162	176.1	162	45	31.0	153.035 (pre-irradiation)
Active Fuel Length (in)	144	150	150	153.5	15	22	N/A
No. Rods per Assembly per Shipment	25	25	25	16	1	1	300 Production or 55 Segmented
No. of Plates per Element	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Fuel Rod Diameter/Plate Thickness (in)	0.422	0.440	0.570 (7×7) 0.4961 (other)	0.440	1.478	0.542	0.381
Clad Material	Zr-4	Zr-4	Zr-2	Zirc Alloy	304SS	Incoloy 800	316 SS
Clad Thickness (in)	0.242	0.026	0.036 (7×7) 0.0343 (other)	0.026	0.02	0.016	0.0225
Pellet Diameter/Meat Thickness (in)	0.3659	0.3805	0.4900 (7×7) 0.4213 (other)	0.3805	1.435 (max)	0.510	N/A
Fuel Material	UO_2	UO_2	UO_2	$UO_2 - PuO_2$/ UO_2	U-ZrH	U-ZrH	N/A
Percent Theoretical Density	97	95	95	95	95	95	N/A
Enrichment (wt % ^{235}U)	5.0	5.0	5.0	5.0 (UO_2) 7.0 fissile Pu (MOX))	20	92 (HEU) 19 (LEU)	N/A
Maximum Average Burnup (MWd/MTHM)	60,000[1]	80,000	60,000 – 80,000	62,500	ACPR 86,100 (50% ^{235}U)[3] FLIP-LEU-II 151,100 (80% ^{235}U)[3]	Variable up to 600,000 (HEU) Variable up to 140,000 (LEU)	N/A
Minimum Cool Time	150 days	150 days	Varies with burnup[7]	90 days (Power Grade MOX – 120 days)	ACPR 231 days FLIP-LEU-II 908 days	Varies with burnup[6]	30 days for production TPBAR; 90 days for PIE TPBAR
U Weight (kg/assembly)	58.2	65.6	108.8 (7×7) 91.3 (other)	N/A	N/A	N/A	N/A
HM Weight (kg/element)	N/A	N/A	N/A	2.63[11]	ACPR 0.280 FLIP-LEU-II 0.824	0.0505 (HEU) 0.2894 (LEU)	N/A
UO_2 Weight (kg/assembly)	66.0	66.0	74.5	N/A	N/A	N/A	N/A

Notes:
1. Up to 2 of the PWR rods may have a maximum average burnup of 65,000 MWd/MTU.
2. Variable cool time down to 90 days using the procedure in Section 7.1.4.
3. Design Basis normal condition source term is for ACPR fuel with 86,100 MWd/MTU (50% ^{235}U depletion) and accident condition source term is for FLIP-LEU-II with 151,100 MWd/MTU (80% ^{235}U depletion).
4. Detailed fuel data is presented in Tables 1.2-1 and 6.2.5-1. The values presented here are the physical values for the bounding source terms of the ACPR and FLIP-LEU-II fuel types.
5. For MTR fuel assemblies, which are cut to remove nonfuel-bearing hardware prior to transport, a nominal 0.28 inch of nonfuel hardware will remain above and below the active fuel region to allow for fuel handling operations.
6. Minimum cool time varies with burnup such that maximum decay heat is 1.875 watts/rod.
7. Varies with burnup – see Table 5.3.8-29.
8. For the shielding evaluation, lower values are conservatively assumed.
9. Maximum burnup of 660,000 MWd/MTU for 380 g ^{235}U and 577,500 MWd/MTU for 460 g ^{235}U.
10. Variable cool time down to 180 days using the procedure in Section 7.1.4.
11. Heavy metal weight per rod.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	GA IFM RERTR	GA IFM HTGR	PULSTAR Fuel	Spiral Fuel Assembly	MOATA Plate Bundle
Assembly Array	N/A	N/A	5×5	Spiral Plates	Parallel Plates
Assembly or Element Weight (lbs)	23.73	23.52	45 (assembly); 1.3 (element)	7.9	13.6[12]
Assembly/Element/Rod Length (in)	29.92	N/A	38 (assembly) 26.2 (element)	63.5 cm	58.4 cm[13]
Active Fuel Length (in)	22.05	N/A	24.1	60.325 cm	58.4 cm
No. Rods per Assembly	13 intact; 7 sectioned	N/A	25	N/A	N/A
No. of Plates per Element	N/A	N/A	N/A	10	maximum 14
Fuel Rod Diameter/Plate Thickness (in)	0.543	N/A	0.47	0.147 cm	0.203 cm
Clad Material	Incoloy	N/A	Zirconium alloy	Al	Al
Clad Thickness (in)	0.031	N/A	0.0185	0.043 cm	N/A
Pellet Diameter/Meat Thickness (in)	0.512	N/A	0.423	0.061 cm	0.1016 cm
Fuel Material	U-ZrH	UC_2; UCO; UO_2; $(Th,U)C_2$; or $(Th,U)O_2$	UO_2	U-Al	U-Al
Percent Theoretical Density	N/A	N/A	94.9% (nominal); 99.5% (analyzed)	N/A	N/A
Enrichment (wt % ^{235}U)	19.7	93.15 (maximum)	6	75	80
Maximum Average Burnup (MWd/MTU)	N/A	N/A	45	70% ^{235}U depletion	30,000 MWd/MTU 4.1% ^{235}U depletion
Minimum Cool Time	None	None	1.0 Year	see MEU DIDO	10 yr
U Weight (kg/assembly)	8.49	0.45	13.33	0.213[14]	0.4375[15]
U Weight (kg/element)	0.42	N/A	0.53	0.0213[16]	0.03125[17]
UO_2 Weight (kg/assembly)	N/A	N/A	15.13	N/A	N/A

Notes: (cont'd)

12. For 14-fuel plate bundle.
13. Not available for in-core configuration. Analysis input restricted to active fuel length.
14. Based on a 160 g ^{235}U fissile material load and listed enrichment.
15. Based on fuel mass per plate multiplied by 14 plates.
16. Based on 10 plates per assembly.
17. Based on 25 g ^{235}U and listed enrichment.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	NRU HEU	NRU LEU	NRX HEU	HEUNL
Assembly Array	Annular	Annular	Annular	N/A
Assembly or Element Weight (lbs)	15.43 (assembly) 0.849 (pin)	19.06 (assembly) 1.15 (pin)	11.75 (assembly) 1.04 (pin)	N/A
Assembly/Element/Rod Length (in)	115 (cropped)	115 (cropped)	120 (cropped)	N/A
Active Fuel Length (in)	108	108	108	N/A
No. Rods per Assembly	12	12	7	N/A
No. of Plates per Element	N/A	N/A	N/A	N/A
Fuel Rod Diameter/Plate Thickness (in)	0.376	0.376	0.409	N/A
Clad Material	Al	Al	Al	N/A
Clad Thickness (in)	0.03 (clad) 0.127 (fin)	0.03 (clad) 0.127 (fin)	0.03 (clad) 0.127 (fin)	N/A
Pellet Diameter/Meat Thickness (in)	0.216	0.216	0.25	N/A
Fuel Material	U-Al	U_3-Si-Al	U-Al	$UO_2(NO_3)_2$ in solution
Percent Theoretical Density	N/A	N/A	N/A	N/A
Enrichment (wt % ^{235}U)	91	19	91	N/A
Maximum Average Burnup (MWd/MTU)	633,000	132,000	615,000	N/A
Minimum Cool Time	19	3	18	None
U Weight (kg/assembly)	0.576	2.76	0.609	N/A
U Weight (kg/element)	0.048	0.230	0.087	N/A
UO_2 Weight (kg/assembly)	N/A	N/A	N/A	N/A
Max HEUNL Payload per Container	N/A	N/A	N/A	64.3 L (17.0 gal)
HEUNL Density	N/A	N/A	N/A	1.3 g/cc

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	SLOWPOKE Fuel Rods	SLOWPOKE Fuel Core
Assembly Array	N/A	Hex Pitch
Active Fuel Length (cm)	22	22
No. Rods	Up to 100 Per Canister (800 per Cask)	Up to 298
Fuel Rod Diameter (cm)	22	22
Clad Material	Al	Al
Clad Thickness (in)	0.051	0.051
Fuel Material	U-Al	U-Al
Enrichment (wt % ^{235}U)	90	90
Maximum Average Burnup (GWd/MTU)	30	N/A
Maximum Average Depletion (%^{235}U)	4.5	2.12
Minimum Cool Time	14 Years	2 weeks
U Weight (gram)	3.1 / Rod	930
^{235}U Weight (gram)	2.8 / Rod	837

Table 5.1.1-3 Nuclear and Thermal Source Parameters

Payload	Decay Heat (kW)	Gamma Source (g/sec)	Neutron Source (n/sec)	Top End-Fitting (g/sec)	Bottom End-Fitting (g/sec)
1 PWR Assembly	2.5	1.27E+16	2.21E+08	1.49E+13	1.25E+13
2 BWR Assemblies	2.2	1.04E+16	1.34E+08	1.16E+12	2.78E+12
15 Sound Metallic Fuel Rods[2]	0.532	4.37E+15	1.61E+05	N/A	N/A
6 Failed Metallic Fuel Rods[1]	0.03	1.75E+15	6.44E+04	N/A	N/A
42 HEU MTR Elements[3,9]	1.26	7.42E+15	1.40E+08	N/A[15]	N/A[15]
42 MEU MTR Elements[3,8]	1.26	7.86E+15	2.88E+07	N/A[15]	N/A[15]
42 LEU MTR Elements[3,8,14]	1.26	7.51E+15	3.96E+07	N/A[15]	N/A[15]
42 DIDO Assemblies[10]	1.05	6.07E+15	9.73E+04	N/A	N/A
25 PWR Rods[2]	1.41	8.39E+15	1.40E+08	N/A	N/A
TRIGA (140 Elements) Normal Condition	1.05	6.52E+15[4]	1.57E+06	Note 6	Note 6
TRIGA (140 Elements) Accident Condition	1.05	5.97E+15[5]	1.06E+08	Note 6	Note 6
HEU TRIGA Cluster Rod[7]	1.875E-03	1.12E+13	4.918E+01	N/A	N/A
LEU TRIGA Cluster Rod[7]	1.875E-03	1.11E+13	4.005E+02	N/A	N/A
General Atomics Irradiated Fuel Material	0.013	3.429E+13	1.279E+04	Note 11	Note 11
300 Production TPBARs	1.005	6.681E+15	N/A	N/A	N/A
55 PIE TPBARs	1,005	5.6E+13	N/A	N/A	N/A
PULSTAR Fuel	1.05[12]	6.206E+15	2.115E+07	N/A	N/A
Spiral Fuel Assembly[13]	0.756	1.07E +14	4.54E+03	N/A	N/A
MOATA Plate Bundle	0.042	2.2E +12	< 1E+03	N/A	N/A
16 PWR MOX Rods	2.3	1.14E+16	1.17E+09	N/A	N/A
4 HEUNL Containers	0.0128	1.17E+14	3.29E+2	N/A	N//A
800 SLOWPOKE Rods	2.17E-03	1.181E+13	7.354E+02	N/A	N/A
1 SLOWPOKE Fuel Core	0.056	4.136E+14	1.047E+02	N/A	N/A

Notes:

1. Gamma and neutron source terms conservatively calculated based on design basis sound metallic fuel rods.
2. 23 rods with 60,000 MWd/MTU burnup and two rods with 65,000 MWd/MTU burnup. Source terms as a function of cool time for the 80,000 MWd/MTU burnup PWR and BWR rods are presented in Section 5.3.8.
3. Bounding values of the gamma and neutron source terms presented for 30W uniform loading for 80% burnup.
4. Based on TRIGA ACPR fuel (86,100 MWd/MTU, 231 days cooling, 50% ^{235}U depletion).
5. Based on TRIGA FLIP-LEU-II fuel (151,100 MWd/MTU, 908 days cooling, 80% ^{235}U depletion).
6. Total hardware gamma is 7.64E+14 gamma/second for ACPR fuel (86,100 MWd/MTU, 231 days cooling, 50% ^{235}U depletion).
7. Source term at TRIGA cluster rods maximum dose rate burnup/cool time combination. For HEU fuel, 150 GWd/MTU, 1.34 years cooled. For LEU fuel, 30 GWd/MTU, 1.5 years cooled. Gamma source includes source from activated inconel clad.
8. Moderator used is light water, H_20.
9. Moderator used is heavy water, D_20.
10. Bounding values of the gamma and neutron source terms presented for 25W uniform loading for 70% burnup HEU fuel.
11. Hardware activation, including end-fitting sources, for the TRIGA elements included in the total gamma source for GA IFM.
12. Cool time required to meet 30 watt per cell heat load limit is 1.5 years.
13. Based on 18 W per assembly heat load.
14. Fuel source represents maximum magnitude gamma source obtained from the 470 g ^{235}U analysis, and the maximum neutron source obtained from the 640 g ^{235}U analysis.
15. A maximum 100 grams of cadmium may be included as part of the MTR fuel element or plate construction. Activation of the cadmium produces no significant source per Section 5.3.4.

Table 5.1.1-4 Combined Dose Rates for Normal Operations Conditions

(1 PWR assembly, 35,000 MWd/MTU, 2-year cool time)

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial at 2 m from personnel barrier, Fuel midplane	1	Neutron	1.25
		Secondary Gamma	0.18
		Primary Gamma	6.71
		TOTAL	8.14
Radial surface, Fuel midplane	2	Neutron	6.53
		Secondary Gamma	1.37
		Primary Gamma	43.44
		TOTAL	51.34
Bottom surface, Axial centerline	3	Neutron	0.33
		Primary Gamma	35.51
		End-fitting Gamma	17.02
		TOTAL	52.86
Bottom at 2 m from impact limiter, Axial centerline	4	Neutron	0.03
		Primary Gamma	2.19
		End-fitting Gamma	0.79
		TOTAL	3.01
Top surface, Axial centerline	5	Neutron	0.12
		Primary Gamma	54.17
		End-fitting Gamma	41.45
		TOTAL	95.74
Top at 2 m from impact limiter, Axial centerline	6	Neutron	0.01
		Primary Gamma	3.82
		End-fitting Gamma	2.17
		TOTAL	6.00
Top at Cab	7	Neutron	0.00135
		Primary Gamma	0.47
		End-fitting Gamma	0.25
		TOTAL	0.72

Table 5.1.1-5 Hypothetical Accident – Loss of Shielding Materials

(1 PWR assembly, 35,000 MWd/MTU, 2-year cool time)

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial surface, Fuel midplane, With neutron shield	8	Neutron	6.53
		Secondary Gamma	1.37
		Primary Gamma	43.44
		TOTAL	51.34
Radial surface, Fuel midplane, Without neutron shield	9	Neutron	177.13
		Secondary Gamma	0.39
		Primary Gamma	75.00
		TOTAL	252.52
Radial at 1 *m* from surface, Fuel midplane, Without neutron shield	10	Neutron	50.93
		Secondary Gamma	1.52
		Primary Gamma	54.59
		TOTAL	107.04

Table 5.1.1-6 Hypothetical Accident – Lead Slump

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial at 1 m from surface, PWR top end-fitting	11	End-fitting Gamma	3.60
		TOTAL	3.60
Radial at 1 m from surface, PWR top end-fitting	12	End-fitting Gamma	1.31
		TOTAL	1.31
Radial at 1 m from surface, PWR top end-fitting	13	End-fitting Gamma	0.80
		TOTAL	0.80
Radial at 1 m from surface, PWR bottom end-fitting	14	End-fitting Gamma	0.01
		TOTAL	0.01
Radial at 1 m from surface, PWR bottom end-fitting	15	End-fitting Gamma	0.35
		TOTAL	0.35
Radial at 1 m from surface, PWR bottom end-fitting	16	End-fitting Gamma	1.48
		TOTAL	1.48
Radial at 1 m from surface, BWR bottom end-fitting	17	End-fitting Gamma	0.10
		TOTAL	0.10
Radial at 1 m from surface, BWR bottom end-fitting	18	End-fitting Gamma	0.54
		TOTAL	0.54
Radial at 1 m from surface, BWR bottom end-fitting	19	End-fitting Gamma	0.84
		TOTAL	0.84

5.3.23 SLOWPOKE Core Configuration

Results of a shielding analysis for one SLOWPOKE core (up to 298 fuel rods and 930 g U) 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 and 10 CFR 71.51 are not exceeded.

Dose rates are calculated using the MCNP (MCNP5, Version 1.60) three-dimensional transport code. Source terms are calculated using the TRITON module of the SCALE package (SCALE 6.1).

5.3.23.1 SLOWPOKE Core Source Term

Source terms are calculated to bound the irradiation history of the SLOWPOKE core. A sketch of the fuel rod is shown in Figure 5.3.23-1 and characteristics are summarized in Table 5.3.23-1. Inputs for irradiation and material parameters required by TRITON are given in Table 5.3.23-2. Key parameters differing between the input and analysis are reduced enrichment, increased fuel mass, and increased irradiation time. All parameters are revised to produce bounding source terms. Each of the modified parameters is described below as to its effect on source:

- Increased fuel mass at a fixed depletion value (% ^{235}U depletion) increases source as the total amount of ^{235}U depleted increases, thereby increasing fission product sources.
- Reduced enrichment has opposing effects on source due to its relative effects on fission product versus higher actinide sources. For a fixed depletion percentage, a reduction in ^{235}U percentage will reduce the amount of material depleted, thereby reducing fission product sources, but increasing source as higher actinides are formed by parasitic absorption at a higher rate, increasing both neutron and gamma sources. Overall, the source effect from enrichment variations is minor, as the enrichment is decreased by only 3% for a high >90% enriched fuel source. This effect is significantly more pronounced for low enrichment fuels.
- Increased irradiation time, in conjunction with a continuous burn at full core power, increases source as it raises the depletion percentage with corresponding increases in both fission products and higher actinides generated.

As the exact configuration of the rods in the core is unknown, two configurations were evaluated; a reference core and a compact core. The configuration shown in Figure 5.3.23-2 is referred to as the reference configuration in which the rods are symmetrically distributed through the core. Figure 5.3.23-3 displays the compact core in which the rods are all shifted towards the center of the core. As the reference core configuration produces maximum gamma source spectra, it is used for the dose rate evaluation.

The SLOWPOKE core is designed to be critical, using fixed Beryllium reflectors surrounding the radial extent of the core and the core bottom. The Beryllium reflector top, also referred to as Beryllium shim, is adjusted to maintain a critical configuration. Top and bottom reflectors are not included within the scope of the 2-D TRITON evaluation.

As a full core was modeled, fuel source was extracted at each ring of the core to determine which location produces maximum source spectra. The maximum gamma source (controlling for shielding) was obtained from the inner ring location (ring 1). This source was then applied to all fuel rods for the dose rate analysis. Gamma source from ring 1 (adjust on a per rod basis) is 24 percent higher for the reference core than the compact core model. The ring 1 per rod source of the reference model is 44 percent higher than the core average per rod source of the reference model.

TRITON input is shown in Figure 5.3.23-4, with the resulting TRITON material model shown in Figure 5.3.23-2. Neutron and gamma source terms for a cool time of 14 days from discharge are presented in Table 5.3.23-3 and Table 5.3.23-4, respectively. The modeled heat load in the dose rate analysis is 56.6 W. The calculated core average heat load at this cool time is 39.3 W or 42.2 kW/MTU.

The effect of subcritical neutron multiplication is directly computed in the MCNP analysis.

5.3.23.2 SLOWPOKE Core Shielding Model

MCNP three-dimensional shielding analysis allows detailed modeling of the fuel, basket, and cask shield configurations. 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.

Fuel Models

The SLOWPOKE core is modeled in MCNP in the same configuration which produced the bounding source spectra. The fuel rods are explicitly modeled.

Cross-section of the VISED model of the source region are shown in Figure 5.3.23-5 and Figure 5.3.23-6 under normal conditions and Figure 5.3.23-8 and Figure 5.3.23-9 under accident conditions. 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. For the SLOWPOKE core analysis, only the top basket containing the SLOWPOKE fuel is modeled. The remaining baskets are modeled as void, conservatively removing material from the shielding model. Similarly, the basket handle structure is modeled as void.

The characteristics of the analyzed SLOWPOKE core basket are summarized in Table 5.3.23-6. The analyzed design for the basket contains a 3-inch steel shield plug attached to the bottom (inside) of the basket lid and a separate spacer to push the fuel down in the basket. The design was updated to incorporate the shield plug and spacer into a single piece. The resulting spacer has a 2.5-inch top plate and a 1.5-inch bottom plate and maintains the 15.75-inch total spacing from the bottom of the lid to the top of the SLOWPOKE core. The modeled basket is conservative as the updated spacer contains an additional inch of shielding material as well as placing shielding directly above the core.

The as modeled basket can be seen in Figure 5.3.23-9, while a sketch of the updated lid design is shown in Figure 5.3.23-10.

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
- Lead slump – Radial and Axial modeled simultaneously

Common to both the normal and accident condition models is a 0.1374 cm gap between the lead outer diameter and the cask outer shell. A lead gap slump is evaluated under hypothetical accident conditions. The lead gap volume is applied to both the axial slump and radial slump simultaneously. No lead slump is expected as the cask lead shield is poured in stages assuring minimal contraction gaps. The modeled radial and axial lead slump can be seen in Figure 5.3.23-8 and Figure 5.3.23-9, respectively. As stated previously, the elevation of the source regions is set at its maximum axial extent. 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. Sample input files are provided in Figure 5.3.23-7 and Figure 5.3.23-11 for normal and accident conditions, respectively.

Tally/Detector Description

MCNP surface (F2) tallies are applied in the calculation of system dose rates. As the normal condition cask model is symmetric around the z-axis, dose rates are calculated as averages around the circumference of the cask. The dose rate profile as a function of z-elevation is generated at the radius of the neutron shield shell for normal conditions. An additional tally is placed in the gap between impact limiter and neutron shield shell on the cask outer shell. Hypothetical accident condition dose rates remove both impact limiter and neutron shield and shield shell. Axial and radial lead slumps are included. As a radial lead slump is evaluated, the tally results are not symmetric around the cask periphery (peaking at the radial slump location). Azimuthal tally divisions are applied to capture peaks around the circumference of the cask. While the plot of dose versus z-elevation for the accident condition displays circumferential average dose rates, the maximum accident condition dose rates reported in the summary table are based on the azimuthal tally results.

Shield Regional Densities

Material compositions for structural and shield materials are shown in Table 5.3.23-5.

5.3.23.3 SLOWPOKE Core Shielding Evaluation

Calculational Methods

The shielding evaluation is performed using MCNP5 v1.6.

The MCNP shielding model described in Section 5.3.23.2 is utilized with the source terms described in Section 5.3.23.1 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 5.3.23-7 provides maximum dose rates for the tabulated distances and transport conditions (normal and accident). Table 5.3.23-8 contains key results. Significant margin is present for all dose rate limits.

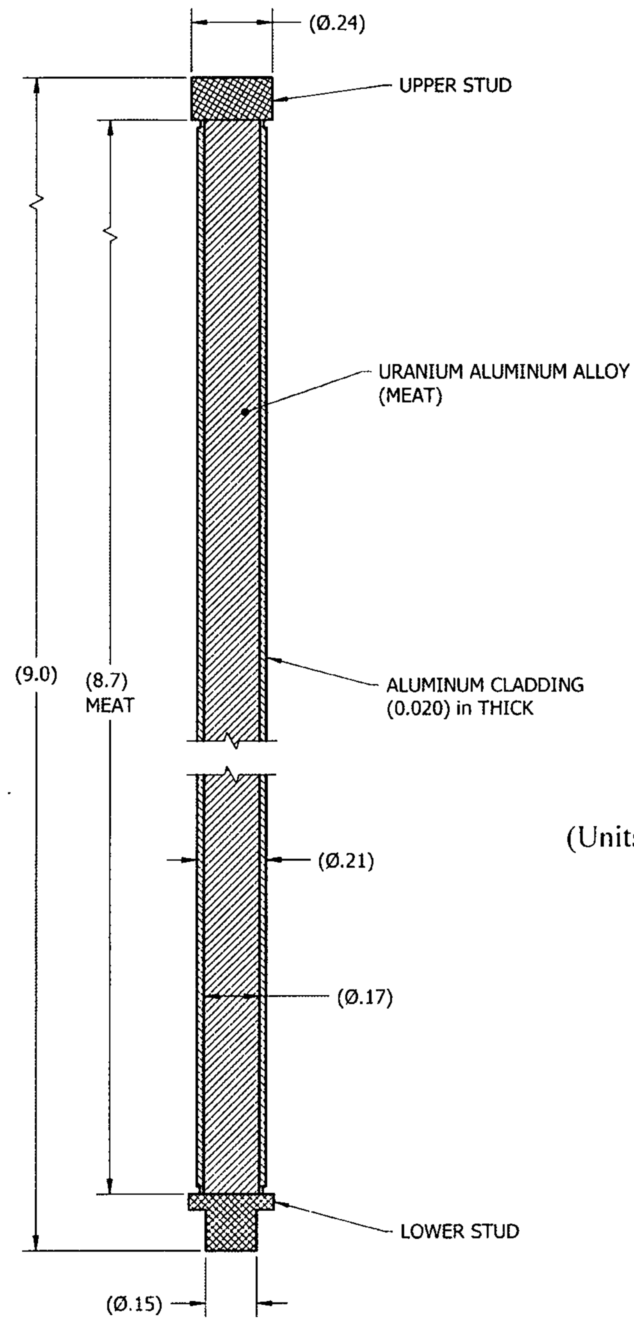
Calculated normal condition radial surface dose rates are below 200 mrem/hr. The Transportation Index (TI) is 15.2 (dose at 1 meter). As the transport index is over 10, an exclusive use designation for the NAC-LWT is used.

The maximum dose rate is dominated by the gamma component. The radial surface dose rate profile is shown in Figure 5.3.23-12. The normal condition maximum radial 2-meter dose rate is 3.1 mrem/hr. As expected, the dose rate profile is skewed towards the top of the cask, as shown Figure 5.3.23-13.

The maximum dose rate at the exposed cask surface above the neutron shield is 42.3 mrem/hr, significantly below the maximum radial dose rate taken from the surface of the neutron shield shell.

Accident condition radial 1-meter dose rates are well below the 1,000 mrem/hr limit. The radial dose rate profile is shown in Figure 5.3.23-14, with the bounding dose rate taken from the azimuthal profile shown in Figure 5.3.23-15.

Figure 5.3.23-1 SLOWPOKE Fuel Element



(Units in inches)

Figure 5.3.23-2 SLOWPOKE Core TRITON Model - Reference

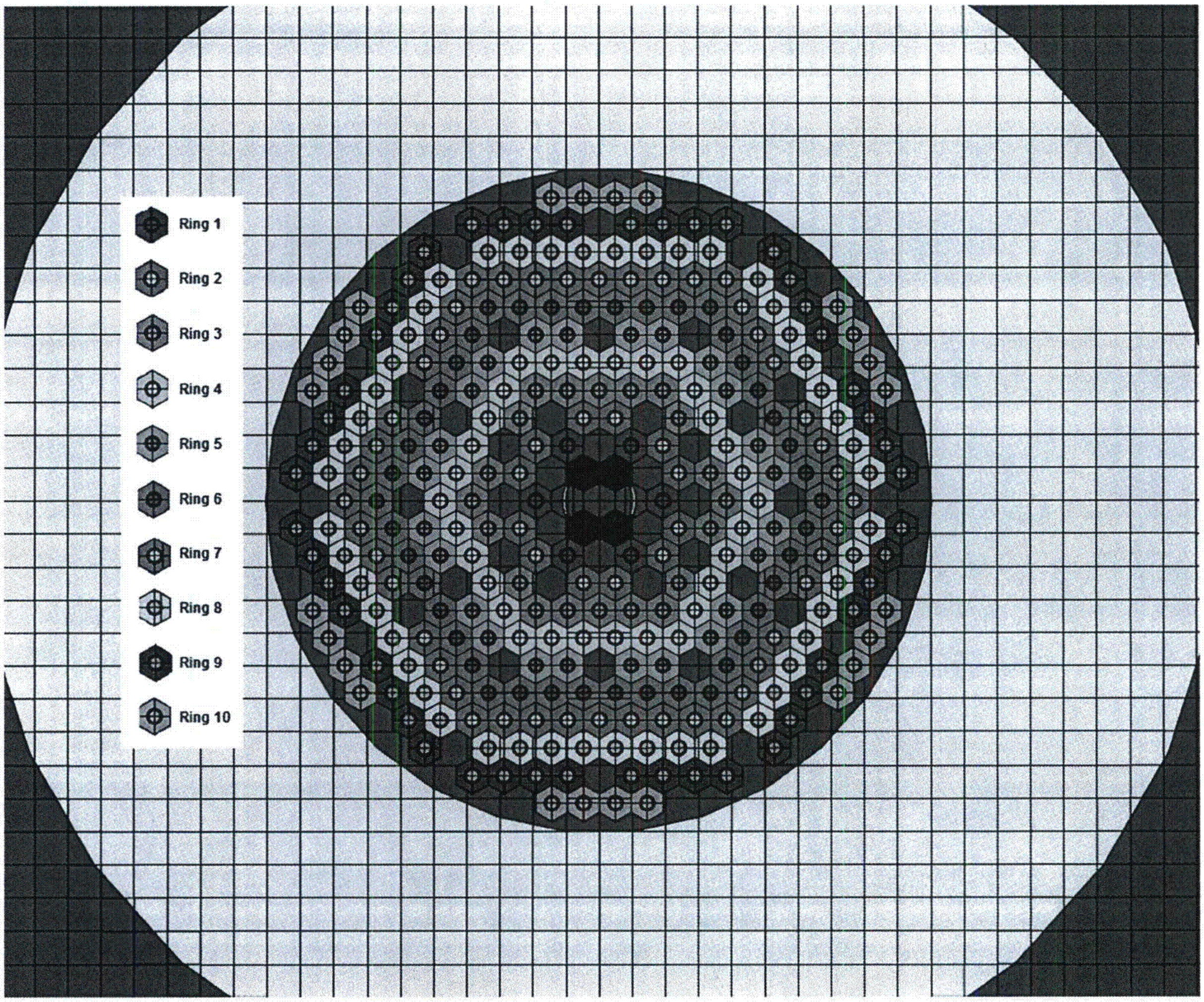


Figure 5.3.23-3 SLOWPOKE Core TRITON Model - Compact

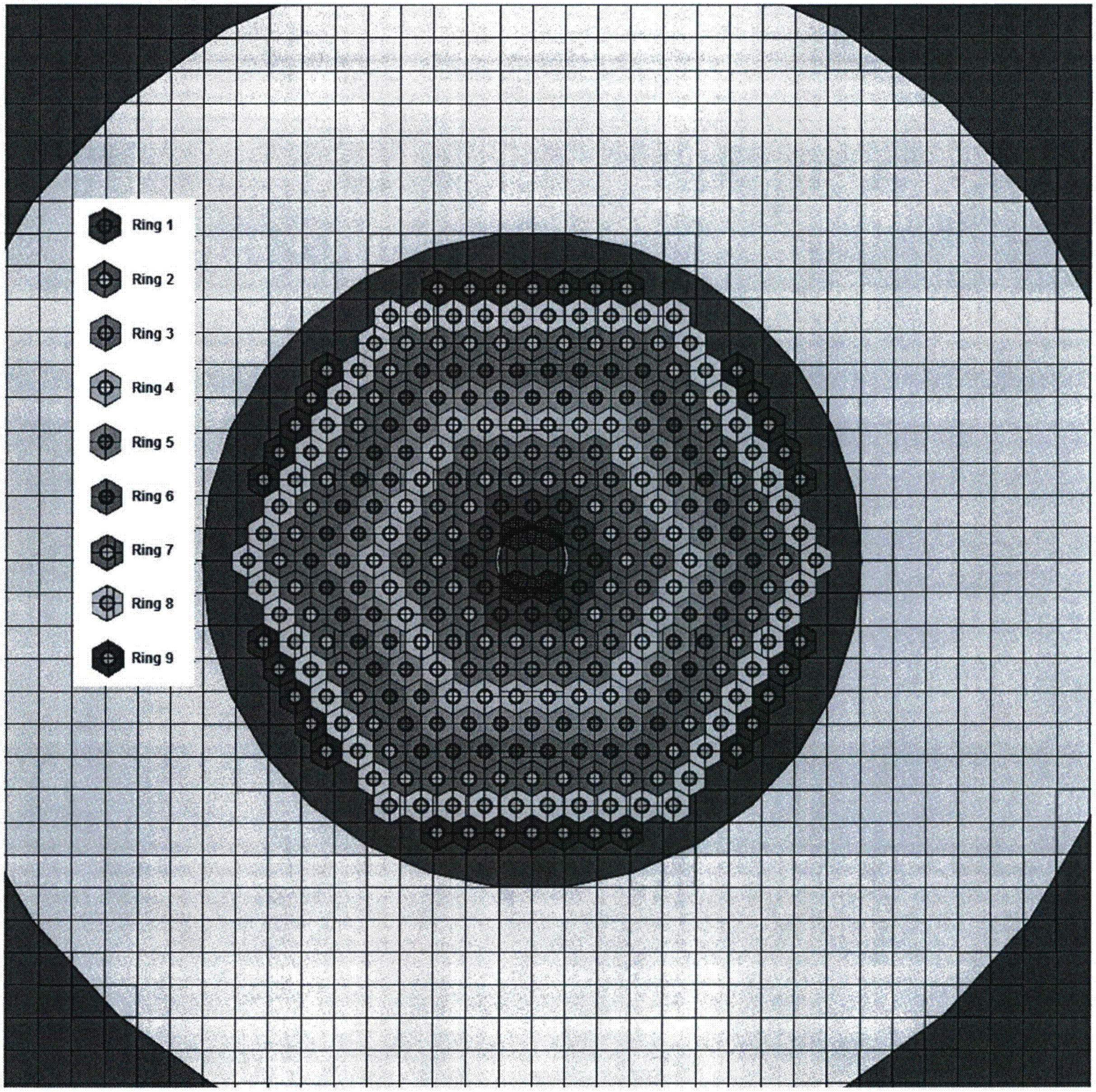


Figure 5.3.23-4 TRITON Input for SLOWPOKE Fuel - Reference

```
=t-depl       (parm=centrm)
SLOWPOKE CORE NEWT / CENTRM Depletion - 1.104 cm Rod Pitch - 15 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    11  1.0  363.0  END
H2O   21  1.0  313.0  END
U      2 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     2 DEN=3.51 0.712 373.0  END
AL    12  1.0  363.0  END
H2O   22  1.0  313.0  END
U      3 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     3 DEN=3.51 0.712 373.0  END
AL    13  1.0  363.0  END
H2O   23  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    14  1.0  363.0  END
H2O   24  1.0  313.0  END
U      5 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     5 DEN=3.51 0.712 373.0  END
AL    15  1.0  363.0  END
H2O   25  1.0  313.0  END
U      6 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     6 DEN=3.51 0.712 373.0  END
AL    16  1.0  363.0  END
H2O   26  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    17  1.0  363.0  END
H2O   27  1.0  313.0  END
U      8 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     8 DEN=3.51 0.712 373.0  END
AL    18  1.0  363.0  END
H2O   28  1.0  313.0  END
U      9 DEN=3.51 0.288 373.0  92235 90.0  92238  10.0  END
AL     9 DEN=3.51 0.712 373.0  END
AL    19  1.0  363.0  END
H2O   29  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    20  1.0  363.0  END
H2O   30  1.0  313.0  END
BE    33  1.0  313.0  END
end comp
read celldata
latticecell triangpitch pitch=1.104  21 fueld=0.422 1 cladd=0.524 11 end
latticecell triangpitch pitch=1.104  22 fueld=0.422 2 cladd=0.524 12 end
latticecell triangpitch pitch=1.104  23 fueld=0.422 3 cladd=0.524 13 end
latticecell triangpitch pitch=1.104  24 fueld=0.422 4 cladd=0.524 14 end
latticecell triangpitch pitch=1.104  25 fueld=0.422 5 cladd=0.524 15 end
latticecell triangpitch pitch=1.104  26 fueld=0.422 6 cladd=0.524 16 end
latticecell triangpitch pitch=1.104  27 fueld=0.422 7 cladd=0.524 17 end
latticecell triangpitch pitch=1.104  28 fueld=0.422 8 cladd=0.524 18 end
latticecell triangpitch pitch=1.104  29 fueld=0.422 9 cladd=0.524 19 end
latticecell triangpitch pitch=1.104  30 fueld=0.422 10 cladd=0.524 20 end
end celldata
read depletion 1 2 3 4 5 6 7 8 9 10 end depletion
read opus
matl= 1 2 3 4 5 6 7 8 9 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
' 298rods-927 gram fuel - 20kW/Core  (21.50MW/MTU)
power=21.50 burn=100 down=0 end
power=21.50 burn=200 down=0 end
power=21.50 burn=200 down=0 end
power=21.50 burn=198 down=14 end
end burndata
read model
SLOWPOKE 298 Rod Assembly - Berylium Reflector - Collapse 44-group
read parm
  prtflux=no  drawit=yes
  xnlib=4 run=yes prtmxsec=no prtbroad=no
  prtmxtab=yes cmfd=no echo=yes
end parm
read materials
  mix=1  pn=1 end
  mix=11 pn=1  end
  mix=21  pn=2  end
  mix=2  pn=1 end
  mix=12 pn=1  end
  mix=22  pn=2  end
  mix=3  pn=1  end
  mix=13 pn=1  end
  mix=23  pn=2  end
  mix=4  pn=1  end
  mix=14 pn=1  end
  mix=24  pn=2  end
  mix=5  pn=1  end
  mix=15 pn=1  end
  mix=25  pn=2  end
  mix=6  pn=1  end
  mix=16 pn=1  end
  mix=26  pn=2  end
  mix=7  pn=1  end
  mix=17 pn=1  end
  mix=27  pn=2  end
  mix=8  pn=1  end
  mix=18 pn=1  end
  mix=28  pn=2  end
  mix=9  pn=1  end
  mix=19 pn=1  end
  mix=29  pn=2  end
  mix=10 pn=1  end
  mix=20 pn=1  end
  mix=30  pn=2  end
  mix=33 pn=2 end
end materials
read geom
' Ring 1
unit 1
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  1 1 10
media 11 1 20 -10
media 21 1 30 -20
boundary 30 2 2
' Ring 2
unit 2
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  2 1 10
media 12 1 20 -10
media 22 1 30 -20
boundary 30 2 2
' Ring 3
unit 3
```

```
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  3 1 10
media 13 1 20 -10
media 23 1 30 -20
boundary 30 2 2
' Ring 4
unit 4
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  4 1 10
media 14 1 20 -10
media 24 1 30 -20
boundary 30 2 2
' Ring 5
unit 5
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  5 1 10
media 15 1 20 -10
media 25 1 30 -20
boundary 30 2 2
' Ring 6
unit 6
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  6 1 10
media 16 1 20 -10
media 26 1 30 -20
boundary 30 2 2
' Ring 7
unit 7
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  7 1 10
media 17 1 20 -10
media 27 1 30 -20
boundary 30 2 2
' Ring 8
unit 8
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  8 1 10
media 18 1 20 -10
media 28 1 30 -20
boundary 30 2 2
' Ring 9
unit 9
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media  9 1 10
media 19 1 20 -10
media 29 1 30 -20
boundary 30 2 2
' Ring 10
unit 10
cylinder 10 0.211
cylinder 20 0.262
hexprism 30 0.552
media 10 1 10
media 20 1 20 -10
media 30 1 30 -20
boundary 30 2 2
' Center Empty
```

```
unit 11
hexprism 30 0.552
media 21 1 30
boundary 30 2 2
' Around Center Empty
unit 12
hexprism 30 0.552
media 21 1 30
boundary 30 2 2
' Right Side Test Unit
unit 14
cylinder 20 1.1684 sides=36 origin x=-1.102
                   chord +x=-0.20 chord +y=-0.55
                   chord -y=+0.55
cylinder 25 1.27   sides=36 origin x=-1.102
                   chord +x=-0.20 chord +y=-0.55
                   chord -y=+0.55
hexprism 30 0.552
media 21 1 20 30
media 11 1 25 30 -20
media 21 1 30 -25 -20
boundary 30 2 2
' Bottom Right Side Test Unit
unit 15
cuboid 20  0.5  0.0  0.2  0.1   rotate a1=35
cuboid 25  -0.0  -0.5  -0.00  -0.10   rotate a1=12
hexprism 30 0.552
media 11 1 20
media 11 1 25
media 21 1 30 -20 -25
boundary 30 15 15
' Bottom Left Side Test Unit
unit 16
cuboid 20   0.5   0.0  -0.0  -0.10   rotate a1=-12
cuboid 25  -0.0  -0.5   0.2   0.1   rotate a1=-35
hexprism 30 0.552
media 11 1 20
media 11 1 25
media 21 1 30 -20 -25
boundary 30 15 15
' Left Side Test Unit
unit 17
cylinder 20 1.1684 sides=36 origin x=1.102
                   chord -x=0.20 chord +y=-0.55
                   chord -y=+0.55
cylinder 25 1.27   sides=36 origin x=1.102
                   chord -x=0.20 chord +y=-0.55
                   chord -y=+0.55
hexprism 30 0.552
media 21 1 20 30
media 11 1 25 30 -20
media 21 1 30 -25 -20
boundary 30 2 2
' Top Right Side Test Unit
unit 18
cuboid 20   0.5   0.0  -0.00  -0.10  rotate a1=-35
cuboid 25  -0.0  -0.5   0.2    0.1   rotate a1=-12
hexprism 30 0.552
media 11 1 20
media 11 1 25
media 21 1 30 -20 -25
boundary 30 15 15
' Top Left Side Test Unit
unit 19
cuboid 20   0.5   0.0   0.2   0.1   rotate a1=12
cuboid 25  -0.0  -0.5  -0.0  -0.10   rotate a1=35
hexprism 30 0.552
media 11 1 20
media 11 1 25
media 21 1 30 -20 -25
boundary 30 15 15
```

```
'
global unit 40
' Disk 8.66 inch - 11 cm radius - use 11.5 cm for clearance
cylinder 110 11.5 sides=30
' 10 cm Be reflector
cylinder 120 21.5 sides=30
cuboid 130 23.0 -23.0 23.0 -23.0
array 1 110 place 12 11 -0.552 -0.9558
media 21 1 110
media 33 1 120 -110
media 21 1 130 -120
boundary 130 40 40
'
end geom
read array
ara=1 typ=shexagonal nux=23 nuy=23
fill
0 0 0 0 0 0 0  0 0 0 10 10 10 10 0 0 0  0 0 0 0 0 0
 0 0 0 0 0 0  0 9 9 9  9  0  9  9 9 9 0  0 0 0 0 0 0
0 0 0 0 0 0  9 0 8 8 8  8  8  8  8 8 0 9  0 0 0 0 0
 0 0 0 0 0  9 8 7 7 7  7  7  7 7  7 7 8 9  0 0 0 0 0
0 0 0 0 10 9 8 7 6 6 6  6  6  6 6  6 7 8 9 10 0 0 0
 0 0 0 10 9 8 7 6 0 5  5  0  5 5 0  6 7 8 9 10 0 0 0
0 0 0 10 0 8 7 6 5 4 4  4  4  4 4 5  6 7 8 0 10 0 0
 0 0 10 9 8 0 6 5 4 3  3  3  3 3 4 5  6 0 8 9 10 0 0
0 0 0  9 8 7 6 0 4 3 0  2  2  0 3 4 0  6 7 8 9  0 0
 0 0  9 8 7 6 5 4 0 2 1  0  1  2 0 4 5  6 7 8 9  0 0
0 0  9 8 7 6 5 4 3 2 0 16 15  0 2 3 4 5  6 7 8 9  0
 0  0 0 7 6 0 4 3 0 1 17 11 14 1 0 3 4 0  6 7 0 0  0
0 0  9 8 7 6 5 4 3 2 0 19 18  0 2 3 4 5  6 7 8 9  0
 0 0  9 8 7 6 5 4 0 2 1  0  1  2 0 4 5  6 7 8 9  0 0
0 0 0  9 8 7 6 0 4 3 0  2  2  0 3 4 0  6 7 8 9  0 0
 0 0 10 9 8 0 6 5 4 3  3  3  3 3 4 5  6 0 8 9 10 0 0
0 0 0 10 0 8 7 6 5 4 4  4  4  4 4 5  6 7 8 0 10 0 0
 0 0 0 10 9 8 7 6 0 5  5  0  5 5 0  6 7 8 9 10 0 0 0
0 0 0 0 10 9 8 7 6 6 6  6  6  6 6  6 7 8 9 10 0 0 0
 0 0 0 0 0  9 8 7 7 7  7  7  7 7  7 7 8 9  0 0 0 0 0
0 0 0 0 0 0  9 0 8 8 8  8  8  8  8 8 0 9  0 0 0 0 0
 0 0 0 0 0 0  0 9 9 9  9  0  9  9 9 9 0  0 0 0 0 0 0
0 0 0 0 0 0 0  0 0 0 10 10 10 10 0 0 0  0 0 0 0 0 0
end fill
end array
read bounds all=vacuum end bounds
end model
end
```

Figure 5.3.23-5 VISED X-Y Slice – SLOWPOKE Core – Normal Conditions

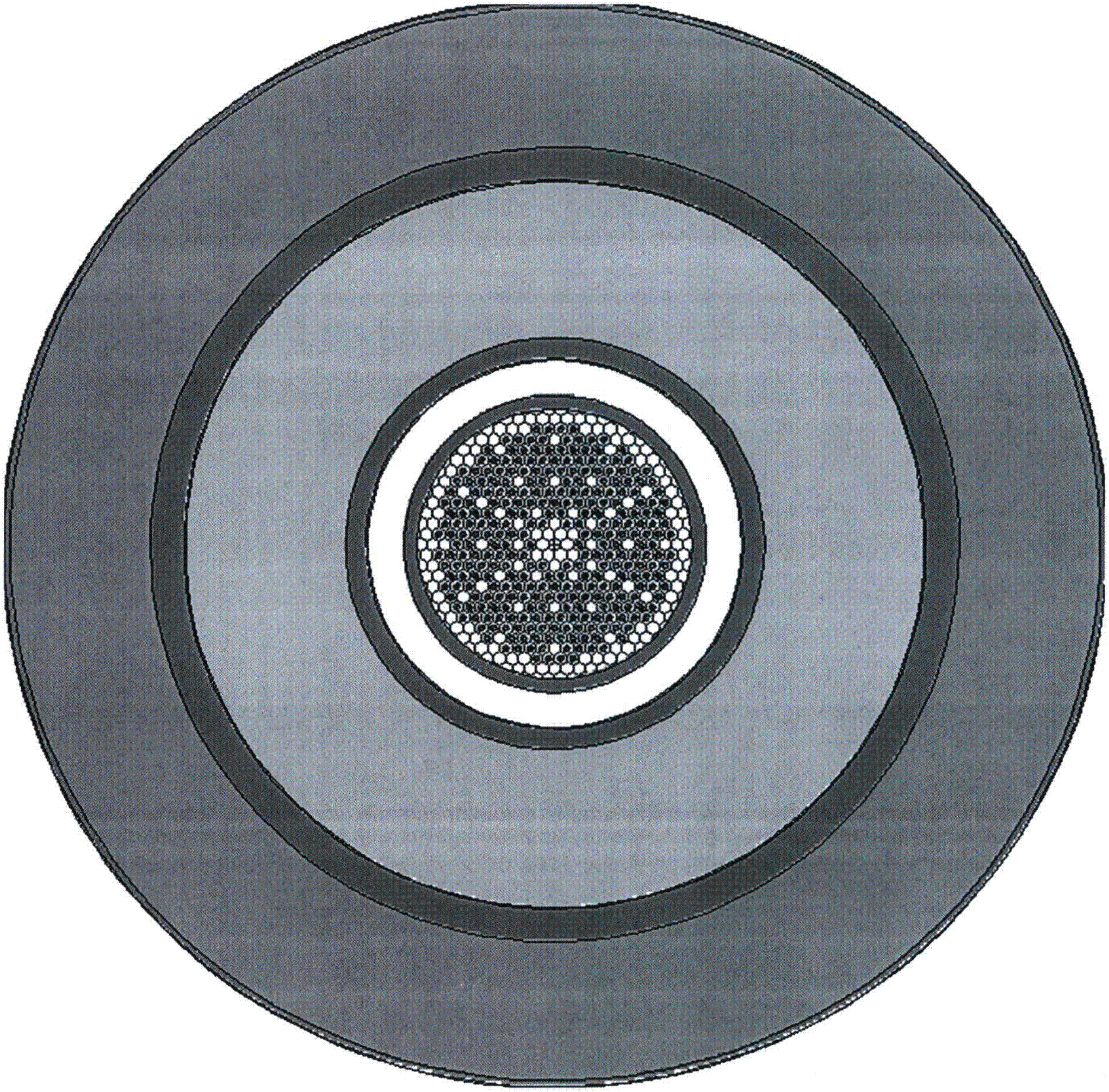
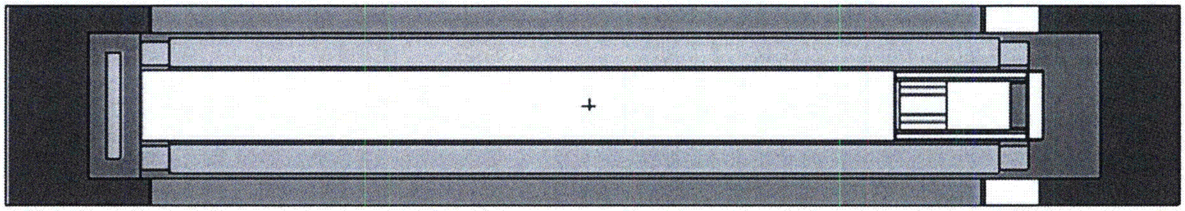


Figure 5.3.23-6 VISED Y-Z Slice – SLOWPOKE Core – Normal Conditions



Note: SLOWPOKE fuel core basket is shifted to the NAC-LWT lid. Void space indicated by model is the space occupied by the basket lid collar whose material, but not spacing, is conservatively removed from the model. Location of the fuel core near the bottom of the basket cavity is maintained by a spacer structurally evaluated to survive both normal and accident conditions of transport.

Figure 5.3.23-7 Sample MCNP Input File – Normal Conditions

```
NAC-LWT Cask - 15b90e14d - Normal Transport Conditions
C Radial Biasing - Fuel Gamma Source
C Fuel Rod Cells
1 1 -3.4819 -1 +3 -4   u=5 $ Fuel Meat
2 4 -2.7000 -2 +1 3 -4  u=5 $ Clad
3 4 -2.7000 -5 : -6     u=5 $ Top Cap
4 4 -2.7000 -7     u=5 $ Bottom Cap
5 0          #1 #2 #3 #4      u=5 $ Outside Rod
C Fuel Core Array Cell
6 0          -8 u=4 lat=2 fill=-15:15 -15:15 0:0  $ Core Array
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 4 5 5 5 5 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 5 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 4 5 5 5 5 5 5 5 5 5 5 5 4 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 5 4 5 5 4 5 5 4 5 5 4 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 5 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 5 5 4 5 5 4 5 4 4 4 5 4 5 5 4 5 5 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 5 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 4 5 5 4 5 5 4 5 5 4 5 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 4 5 5 5 5 5 5 5 5 5 5 5 4 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 5 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 5 5 5 5 4 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
C Cells - SLOWPOKE Core Basket
7 0          -9 fill=4 ( 0.0000 0.0000 3.3274 )   u=3 $ SLOWPOKE Core
8 6 -7.9400 -10     u=3 $ Base Plate
9 6 -7.9400 -11 +12    u=3 $ Basket Tube
10 6 -7.9400 -13     u=3 $ Shield Plug
11 6 -7.9400 -14     u=3 $ Lid Plate
12 0         -15     u=3 $ Lid Spacer
13 0         #7 #8 #9 #10 #11 #12      u=3 $ Void
C Cells - LWT Cavity
14 0         -16 fill=3 ( 0.0000 0.0000 377.6980 )   u=2
15 0         #14     u=2
C Cells - LWT Cask Normal Conditions
16 5 -11.344 -20     u=1 $ BotPb
17 0         -19 fill=2    u=1 $ Cavity
18 6 -7.9400 -17 -18 +20    u=1 $ Bottom
19 6 -7.9400 -17 +18 +22 +25 +19 u=1 $ OuterShell
20 6 -7.9400 -21 +24 +19    u=1 $ InnerShellTaper
21 6 -7.9400 -23 +19    u=1 $ InnerShell
22 5 -11.344 -24 +23    u=1 $ Lead
23 5 -11.344 -22 +21 +24    u=1 $ LeadTaper
24 0         -25 +24    u=1 $ LeadGap
25 3 -0.9400 -27 +17    u=1 $ NeutronShield
26 6 -7.9400 -26 +17 +27    u=1 $ NSShell
27 7 -0.4997 -28 +17    u=1 $ UpperLimiter
28 7 -0.4997 -29 +17    u=1 $ LowerLimiter
29 0         -30 +17 +26 +28 +29 u=1 $ Container
30 0         +30      u=1 $ Outside
C Detector Cells - Radial Biasing
```

```
100 0 -100 fill=1 $ AziTrun
110 0 -110 +100 fill=1 $ Surface
310 0 -310 +110 $ AziSurFuel
329 0 -329 +110 +310 $ 1ft
429 0 -429 +110 +310 +329 $ 1m
529 0 -529 +110 +310 +329 +429 $ Azi1m
548 0 -548 +110 +310 +329 +429 +529 $ 2m
648 0 -648 +110 +310 +329 +429 +529 +548 $ 2m+Convey
748 0 -748 +110 +310 +329 +429 +529 +548 +648 $ Azi2m+Con
848 0 +110 +310 +329 +429 +529 +548 +648 +748 $ Exterior

C Fuel Rod Surfaces
1 CZ 0.2108       $ Fuel Meat OD
2 CZ 0.2616       $ Clad OD
3 PZ 0.4572       $ Lower Fuel Cut Plain
4 PZ 22.4536       $ Upper Fuel Cut Plain
5 RCC 0.0000 0.0000 0.3302 0.0000 0.0000 0.1270 0.3048 $ Lower Stud Rim
6 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 0.3302 0.1930 $ Lower Stud Cap
7 RCC 0.0000 0.0000 22.4536 0.0000 0.0000 0.3810 0.3048 $ Upper Stud
C Fuel Core Lattice Surface
8 RHP 0.0 0.0 0.0 0.0 0.0 22.8346
         0.5518 0.0 0.0    $ Lattice Cell
C Surfaces - SLOWPOKE Core Basket
9 RCC 0.0000 0.0000 3.3274 0.0000 0.0000 22.8346 12.3824 $ Core
10 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 1.2700 16.8466 $ Base Plate
11 RCC 0.0000 0.0000 1.2700 0.0000 0.0000 64.8970 13.6525 $ Tube OD
12 RCC 0.0000 0.0000 1.2700 0.0000 0.0000 64.8970 12.3825 $ Tube ID
13 RCC 0.0000 0.0000 58.5470 0.0000 0.0000 7.6200 10.9982 $ Shield Plug
14 RCC 0.0000 0.0000 66.1670 0.0000 0.0000 1.2700 16.8466 $ Lid Plate
15 RCC 0.0000 0.0000 67.4370 0.0000 0.0000 6.9850 16.84655 $ Lid Spacer
C Surfaces - LWT Cavity
16 RCC 0.0000 0.0000 377.6980 0.0000 0.0000 74.4220 16.8467 $ Basket
C Surfaces - LWT Cask Normal Conditions
17 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 507.3650 36.5189 $ Lwt
18 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 26.6700 36.5189 $ Bottom
19 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 452.1200 16.9863 $ Cavity
20 RCC 0.0000 0.0000 -17.7800 0.0000 0.0000 7.6200 26.3525 $ Bottom gamma shield
21 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 20.1740 $ Lead id - taper
22 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 31.5976 $ Lead od - taper
23 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 18.9103 $ Lead id
24 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 33.3271 $ Lead od
25 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 33.4645 $ Lead gap
26 RCC 0.0000 0.0000 3.8100 0.0000 0.0000 419.1000 49.8183 $ Neutron shield shell
27 RCC 0.0000 0.0000 5.0800 0.0000 0.0000 416.5600 49.2189 $ Neutron shield
28 RCC 0.0000 0.0000 450.2150 0.0000 0.0000 70.5612 49.8183 $ Upper limiter
29 RCC 0.0000 0.0000 -68.0212 0.0000 0.0000 71.8312 49.8183 $ Lower limiter
30 RCC 0.0000 0.0000 -68.0212 0.0000 0.0000 588.7974 49.8183 $ Container
C Radial Detector DRA (AziTrun)
100   RCC  0.0000   0.0000   422.9200   0.0000   0.0000   27.2850   36.6189
101   PZ  425.6485
102   PZ  428.3770
103   PZ  431.1055
104   PZ  433.8340
105   PZ  436.5625
106   PZ  439.2910
107   PZ  442.0195
108   PZ  444.7480
109   PZ  447.4765
C Radial Detector DRB (Surface)
110   RCC  0.0000   0.0000   -68.1212   0.0000   0.0000   588.9974   49.9184
111   PZ  -65.1762
112   PZ  -62.2312
113   PZ  -59.2862
114   PZ  -56.3413
115   PZ  -53.3963
116   PZ  -50.4513
117   PZ  -47.5063
118   PZ  -44.5613
119   PZ  -41.6163
120   PZ  -38.6713
121   PZ  -35.7263
```

```
122  PZ  -32.7814
123  PZ  -29.8364
124  PZ  -26.8914
125  PZ  -23.9464
126  PZ  -21.0014
127  PZ  -18.0564
128  PZ  -15.1114
129  PZ  -12.1664
130  PZ  -9.2215
131  PZ  -6.2765
132  PZ  -3.3315
133  PZ  -0.3865
134  PZ  2.5585
135  PZ  5.5035
136  PZ  8.4485
137  PZ  11.3934
138  PZ  14.3384
139  PZ  17.2834
140  PZ  20.2284
141  PZ  23.1734
142  PZ  26.1184
143  PZ  29.0634
144  PZ  32.0084
145  PZ  34.9533
146  PZ  37.8983
147  PZ  40.8433
148  PZ  43.7883
149  PZ  46.7333
150  PZ  49.6783
151  PZ  52.6233
152  PZ  55.5683
153  PZ  58.5132
154  PZ  61.4582
155  PZ  64.4032
156  PZ  67.3482
157  PZ  70.2932
158  PZ  73.2382
159  PZ  76.1832
160  PZ  79.1282
161  PZ  82.0731
162  PZ  85.0181
163  PZ  87.9631
164  PZ  90.9081
165  PZ  93.8531
166  PZ  96.7981
167  PZ  99.7431
168  PZ  102.6880
169  PZ  105.6330
170  PZ  108.5780
171  PZ  111.5230
172  PZ  114.4680
173  PZ  117.4130
174  PZ  120.3580
175  PZ  123.3030
176  PZ  126.2479
177  PZ  129.1929
178  PZ  132.1379
179  PZ  135.0829
180  PZ  138.0279
181  PZ  140.9729
182  PZ  143.9179
183  PZ  146.8629
184  PZ  149.8078
185  PZ  152.7528
186  PZ  155.6978
187  PZ  158.6428
188  PZ  161.5878
189  PZ  164.5328
190  PZ  167.4778
191  PZ  170.4227
192  PZ  173.3677
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193	PZ	176.3127
194	PZ	179.2577
195	PZ	182.2027
196	PZ	185.1477
197	PZ	188.0927
198	PZ	191.0377
199	PZ	193.9826
200	PZ	196.9276
201	PZ	199.8726
202	PZ	202.8176
203	PZ	205.7626
204	PZ	208.7076
205	PZ	211.6526
206	PZ	214.5976
207	PZ	217.5425
208	PZ	220.4875
209	PZ	223.4325
210	PZ	226.3775
211	PZ	229.3225
212	PZ	232.2675
213	PZ	235.2125
214	PZ	238.1574
215	PZ	241.1024
216	PZ	244.0474
217	PZ	246.9924
218	PZ	249.9374
219	PZ	252.8824
220	PZ	255.8274
221	PZ	258.7724
222	PZ	261.7173
223	PZ	264.6623
224	PZ	267.6073
225	PZ	270.5523
226	PZ	273.4973
227	PZ	276.4423
228	PZ	279.3873
229	PZ	282.3323
230	PZ	285.2772
231	PZ	288.2222
232	PZ	291.1672
233	PZ	294.1122
234	PZ	297.0572
235	PZ	300.0022
236	PZ	302.9472
237	PZ	305.8921
238	PZ	308.8371
239	PZ	311.7821
240	PZ	314.7271
241	PZ	317.6721
242	PZ	320.6171
243	PZ	323.5621
244	PZ	326.5071
245	PZ	329.4520
246	PZ	332.3970
247	PZ	335.3420
248	PZ	338.2870
249	PZ	341.2320
250	PZ	344.1770
251	PZ	347.1220
252	PZ	350.0670
253	PZ	353.0119
254	PZ	355.9569
255	PZ	358.9019
256	PZ	361.8469
257	PZ	364.7919
258	PZ	367.7369
259	PZ	370.6819
260	PZ	373.6269
261	PZ	376.5718
262	PZ	379.5168
263	PZ	382.4618

```
264   PZ   385.4068
265   PZ   388.3518
266   PZ   391.2968
267   PZ   394.2418
268   PZ   397.1867
269   PZ   400.1317
270   PZ   403.0767
271   PZ   406.0217
272   PZ   408.9667
273   PZ   411.9117
274   PZ   414.8567
275   PZ   417.8017
276   PZ   420.7466
277   PZ   423.6916
278   PZ   426.6366
279   PZ   429.5816
280   PZ   432.5266
281   PZ   435.4716
282   PZ   438.4166
283   PZ   441.3616
284   PZ   444.3065
285   PZ   447.2515
286   PZ   450.1965
287   PZ   453.1415
288   PZ   456.0865
289   PZ   459.0315
290   PZ   461.9765
291   PZ   464.9214
292   PZ   467.8664
293   PZ   470.8114
294   PZ   473.7564
295   PZ   476.7014
296   PZ   479.6464
297   PZ   482.5914
298   PZ   485.5364
299   PZ   488.4813
300   PZ   491.4263
301   PZ   494.3713
302   PZ   497.3163
303   PZ   500.2613
304   PZ   503.2063
305   PZ   506.1513
306   PZ   509.0963
307   PZ   512.0412
308   PZ   514.9862
309   PZ   517.9312
C Radial Detector DRBA (AziSurFuel)
310   RCC  0.0000  0.0000  385.0000  0.0000  0.0000  10.0000  49.9185
311      PX  0.0000
312   1  PX  0.0000
313   2  PX  0.0000
314   3  PX  0.0000
315   4  PX  0.0000
316   5  PX  0.0000
317   6  PX  0.0000
318   7  PX  0.0000
319   8  PX  0.0000
320      PY  0.0000
321   10  PX  0.0000
322   11  PX  0.0000
323   12  PX  0.0000
324   13  PX  0.0000
325   14  PX  0.0000
326   15  PX  0.0000
327   16  PX  0.0000
328   17  PX  0.0000
C Radial Detector DRC (1ft)
329   RCC  0.0000  0.0000  -98.6012  0.0000  0.0000  649.9574  80.2984
330   PZ  -92.1016
331   PZ  -85.6021
332   PZ  -79.1025
```

333	PZ	-72.6029
334	PZ	-66.1033
335	PZ	-59.6038
336	PZ	-53.1042
337	PZ	-46.6046
338	PZ	-40.1050
339	PZ	-33.6055
340	PZ	-27.1059
341	PZ	-20.6063
342	PZ	-14.1067
343	PZ	-7.6072
344	PZ	-1.1076
345	PZ	5.3920
346	PZ	11.8916
347	PZ	18.3911
348	PZ	24.8907
349	PZ	31.3903
350	PZ	37.8899
351	PZ	44.3894
352	PZ	50.8890
353	PZ	57.3886
354	PZ	63.8882
355	PZ	70.3877
356	PZ	76.8873
357	PZ	83.3869
358	PZ	89.8864
359	PZ	96.3860
360	PZ	102.8856
361	PZ	109.3852
362	PZ	115.8847
363	PZ	122.3843
364	PZ	128.8839
365	PZ	135.3835
366	PZ	141.8830
367	PZ	148.3826
368	PZ	154.8822
369	PZ	161.3818
370	PZ	167.8813
371	PZ	174.3809
372	PZ	180.8805
373	PZ	187.3801
374	PZ	193.8796
375	PZ	200.3792
376	PZ	206.8788
377	PZ	213.3784
378	PZ	219.8779
379	PZ	226.3775
380	PZ	232.8771
381	PZ	239.3766
382	PZ	245.8762
383	PZ	252.3758
384	PZ	258.8754
385	PZ	265.3749
386	PZ	271.8745
387	PZ	278.3741
388	PZ	284.8737
389	PZ	291.3732
390	PZ	297.8728
391	PZ	304.3724
392	PZ	310.8720
393	PZ	317.3715
394	PZ	323.8711
395	PZ	330.3707
396	PZ	336.8703
397	PZ	343.3698
398	PZ	349.8694
399	PZ	356.3690
400	PZ	362.8686
401	PZ	369.3681
402	PZ	375.8677
403	PZ	382.3673

```
404   PZ   388.8669
405   PZ   395.3664
406   PZ   401.8660
407   PZ   408.3656
408   PZ   414.8651
409   PZ   421.3647
410   PZ   427.8643
411   PZ   434.3639
412   PZ   440.8634
413   PZ   447.3630
414   PZ   453.8626
415   PZ   460.3622
416   PZ   466.8617
417   PZ   473.3613
418   PZ   479.8609
419   PZ   486.3605
420   PZ   492.8600
421   PZ   499.3596
422   PZ   505.8592
423   PZ   512.3588
424   PZ   518.8583
425   PZ   525.3579
426   PZ   531.8575
427   PZ   538.3571
428   PZ   544.8566
C Radial Detector DRD (1m)
429   RCC   0.0000   0.0000   -168.1212   0.0000   0.0000   788.9974   149.8184
430   PZ   -160.2312
431   PZ   -152.3413
432   PZ   -144.4513
433   PZ   -136.5613
434   PZ   -128.6713
435   PZ   -120.7814
436   PZ   -112.8914
437   PZ   -105.0014
438   PZ   -97.1114
439   PZ   -89.2215
440   PZ   -81.3315
441   PZ   -73.4415
442   PZ   -65.5515
443   PZ   -57.6616
444   PZ   -49.7716
445   PZ   -41.8816
446   PZ   -33.9916
447   PZ   -26.1017
448   PZ   -18.2117
449   PZ   -10.3217
450   PZ   -2.4317
451   PZ   5.4582
452   PZ   13.3482
453   PZ   21.2382
454   PZ   29.1282
455   PZ   37.0181
456   PZ   44.9081
457   PZ   52.7981
458   PZ   60.6880
459   PZ   68.5780
460   PZ   76.4680
461   PZ   84.3580
462   PZ   92.2479
463   PZ   100.1379
464   PZ   108.0279
465   PZ   115.9179
466   PZ   123.8078
467   PZ   131.6978
468   PZ   139.5878
469   PZ   147.4778
470   PZ   155.3677
471   PZ   163.2577
472   PZ   171.1477
473   PZ   179.0377
```

```
474   PZ   186.9276
475   PZ   194.8176
476   PZ   202.7076
477   PZ   210.5976
478   PZ   218.4875
479   PZ   226.3775
480   PZ   234.2675
481   PZ   242.1574
482   PZ   250.0474
483   PZ   257.9374
484   PZ   265.8274
485   PZ   273.7173
486   PZ   281.6073
487   PZ   289.4973
488   PZ   297.3873
489   PZ   305.2772
490   PZ   313.1672
491   PZ   321.0572
492   PZ   328.9472
493   PZ   336.8371
494   PZ   344.7271
495   PZ   352.6171
496   PZ   360.5071
497   PZ   368.3970
498   PZ   376.2870
499   PZ   384.1770
500   PZ   392.0670
501   PZ   399.9569
502   PZ   407.8469
503   PZ   415.7369
504   PZ   423.6269
505   PZ   431.5168
506   PZ   439.4068
507   PZ   447.2968
508   PZ   455.1867
509   PZ   463.0767
510   PZ   470.9667
511   PZ   478.8567
512   PZ   486.7466
513   PZ   494.6366
514   PZ   502.5266
515   PZ   510.4166
516   PZ   518.3065
517   PZ   526.1965
518   PZ   534.0865
519   PZ   541.9765
520   PZ   549.8664
521   PZ   557.7564
522   PZ   565.6464
523   PZ   573.5364
524   PZ   581.4263
525   PZ   589.3163
526   PZ   597.2063
527   PZ   605.0963
528   PZ   612.9862
C Radial Detector DRDA (Azilm)
529   RCC   0.0000   0.0000   385.0000   0.0000   0.0000   15.0000   149.8185
530       PX   0.0000
531   1   PX   0.0000
532   2   PX   0.0000
533   3   PX   0.0000
534   4   PX   0.0000
535   5   PX   0.0000
536   6   PX   0.0000
537   7   PX   0.0000
538   8   PX   0.0000
539       PY   0.0000
540   10   PX   0.0000
541   11   PX   0.0000
542   12   PX   0.0000
543   13   PX   0.0000
```

```
544  14  PX  0.0000
545  15  PX  0.0000
546  16  PX  0.0000
547  17  PX  0.0000
C Radial Detector DRE (2m)
548  RCC  0.0000  0.0000  -268.1212  0.0000  0.0000  988.9974  249.8184
549  PZ  -258.2312
550  PZ  -248.3413
551  PZ  -238.4513
552  PZ  -228.5613
553  PZ  -218.6713
554  PZ  -208.7814
555  PZ  -198.8914
556  PZ  -189.0014
557  PZ  -179.1114
558  PZ  -169.2215
559  PZ  -159.3315
560  PZ  -149.4415
561  PZ  -139.5515
562  PZ  -129.6616
563  PZ  -119.7716
564  PZ  -109.8816
565  PZ  -99.9916
566  PZ  -90.1017
567  PZ  -80.2117
568  PZ  -70.3217
569  PZ  -60.4317
570  PZ  -50.5418
571  PZ  -40.6518
572  PZ  -30.7618
573  PZ  -20.8719
574  PZ  -10.9819
575  PZ  -1.0919
576  PZ  8.7981
577  PZ  18.6880
578  PZ  28.5780
579  PZ  38.4680
580  PZ  48.3580
581  PZ  58.2479
582  PZ  68.1379
583  PZ  78.0279
584  PZ  87.9179
585  PZ  97.8078
586  PZ  107.6978
587  PZ  117.5878
588  PZ  127.4778
589  PZ  137.3677
590  PZ  147.2577
591  PZ  157.1477
592  PZ  167.0377
593  PZ  176.9276
594  PZ  186.8176
595  PZ  196.7076
596  PZ  206.5976
597  PZ  216.4875
598  PZ  226.3775
599  PZ  236.2675
600  PZ  246.1574
601  PZ  256.0474
602  PZ  265.9374
603  PZ  275.8274
604  PZ  285.7173
605  PZ  295.6073
606  PZ  305.4973
607  PZ  315.3873
608  PZ  325.2772
609  PZ  335.1672
610  PZ  345.0572
611  PZ  354.9472
612  PZ  364.8371
613  PZ  374.7271
```

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614  PZ  384.6171
615  PZ  394.5071
616  PZ  404.3970
617  PZ  414.2870
618  PZ  424.1770
619  PZ  434.0670
620  PZ  443.9569
621  PZ  453.8469
622  PZ  463.7369
623  PZ  473.6269
624  PZ  483.5168
625  PZ  493.4068
626  PZ  503.2968
627  PZ  513.1867
628  PZ  523.0767
629  PZ  532.9667
630  PZ  542.8567
631  PZ  552.7466
632  PZ  562.6366
633  PZ  572.5266
634  PZ  582.4166
635  PZ  592.3065
636  PZ  602.1965
637  PZ  612.0865
638  PZ  621.9765
639  PZ  631.8664
640  PZ  641.7564
641  PZ  651.6464
642  PZ  661.5364
643  PZ  671.4263
644  PZ  681.3163
645  PZ  691.2063
646  PZ  701.0963
647  PZ  710.9862
C Radial Detector DRF (2m+Convey)
648  RCC  0.0000  0.0000  -269.1212  0.0000  0.0000  990.9974  321.9200
649  PZ  -259.2112
650  PZ  -249.3013
651  PZ  -239.3913
652  PZ  -229.4813
653  PZ  -219.5713
654  PZ  -209.6614
655  PZ  -199.7514
656  PZ  -189.8414
657  PZ  -179.9314
658  PZ  -170.0215
659  PZ  -160.1115
660  PZ  -150.2015
661  PZ  -140.2915
662  PZ  -130.3816
663  PZ  -120.4716
664  PZ  -110.5616
665  PZ  -100.6516
666  PZ  -90.7417
667  PZ  -80.8317
668  PZ  -70.9217
669  PZ  -61.0117
670  PZ  -51.1018
671  PZ  -41.1918
672  PZ  -31.2818
673  PZ  -21.3719
674  PZ  -11.4619
675  PZ  -1.5519
676  PZ  8.3581
677  PZ  18.2680
678  PZ  28.1780
679  PZ  38.0880
680  PZ  47.9980
681  PZ  57.9079
682  PZ  67.8179
683  PZ  77.7279
```

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684  PZ  87.6379
685  PZ  97.5478
686  PZ  107.4578
687  PZ  117.3678
688  PZ  127.2778
689  PZ  137.1877
690  PZ  147.0977
691  PZ  157.0077
692  PZ  166.9177
693  PZ  176.8276
694  PZ  186.7376
695  PZ  196.6476
696  PZ  206.5576
697  PZ  216.4675
698  PZ  226.3775
699  PZ  236.2875
700  PZ  246.1974
701  PZ  256.1074
702  PZ  266.0174
703  PZ  275.9274
704  PZ  285.8373
705  PZ  295.7473
706  PZ  305.6573
707  PZ  315.5673
708  PZ  325.4772
709  PZ  335.3872
710  PZ  345.2972
711  PZ  355.2072
712  PZ  365.1171
713  PZ  375.0271
714  PZ  384.9371
715  PZ  394.8471
716  PZ  404.7570
717  PZ  414.6670
718  PZ  424.5770
719  PZ  434.4870
720  PZ  444.3969
721  PZ  454.3069
722  PZ  464.2169
723  PZ  474.1269
724  PZ  484.0368
725  PZ  493.9468
726  PZ  503.8568
727  PZ  513.7667
728  PZ  523.6767
729  PZ  533.5867
730  PZ  543.4967
731  PZ  553.4066
732  PZ  563.3166
733  PZ  573.2266
734  PZ  583.1366
735  PZ  593.0465
736  PZ  602.9565
737  PZ  612.8665
738  PZ  622.7765
739  PZ  632.6864
740  PZ  642.5964
741  PZ  652.5064
742  PZ  662.4164
743  PZ  672.3263
744  PZ  682.2363
745  PZ  692.1463
746  PZ  702.0563
747  PZ  711.9662
C Radial Detector DRFA (Azi2m+Con)
748  RCC  0.0000  0.0000  390.0000  0.0000  0.0000  20.0000  321.9201
749     PX  0.0000
750  1  PX  0.0000
751  2  PX  0.0000
752  3  PX  0.0000
753  4  PX  0.0000
```

```
754  5  PX  0.0000
755  6  PX  0.0000
756  7  PX  0.0000
757  8  PX  0.0000
758     PY  0.0000
759  10  PX  0.0000
760  11  PX  0.0000
761  12  PX  0.0000
762  13  PX  0.0000
763  14  PX  0.0000
764  15  PX  0.0000
765  16  PX  0.0000
766  17  PX  0.0000

C
C Materials List
C
C U-Al Fuel
m1    92235 -2.5201E-01 92238 -2.8001E-02 13027 -7.1999E-01
C Water
m2    1001 6.6667E-01 8016 3.3333E-01
mt2   lwtr.01
C Water/Glycol
m3    1001 -1.03200E-01 8016 -6.82400E-01 6000 -2.14400E-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 38r 0
C
C Source Definition - Fuel Gamma
C 15 GWd/MTU burnup, 90 wt% U-235, 14-day cool time, 2.786 g U-235 per rod, 39 W/core
sdef RAD=d1 EXT=d2 ERG=d3 cell=d4
     POS= 0.0000 0.0000 0.4572
     AXS= 0.0000 0.0000 1.0000
si1 0 0.2108
sp1 -21 1
si2 0 21.9964
sp2 0 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 3.1684E+11 1.0982E+11 1.3380E+11 2.3964E+10 5.7276E+10
     1.5811E+11 4.3248E+11 4.1310E+10 2.5592E+09 1.0587E+11 3.5978E+08
     2.0105E+09 3.4304E+09 2.6553E+07 7.9396E-04 1.8338E-04 3.4862E-05
     7.4653E-06
# SI4  SP4
     L  D
     110:17:14:7:6:-1  1.0000
mode p
nps 6.40E+07
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 13 402 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 500 541 550 558 568 947 969 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 AziTrun Tally
f2:p +100.1
fm2 4.13583E+14
fs2  -101  -102  -103  -104  -105  -106
     -107  -108  -109  T
tf2
fc12 Radial Surface Tally
f12:p +110.1
fm12 4.13583E+14
fs12  -111  -112  -113  -114  -115  -116
     -117  -118  -119  -120  -121  -122
     -123  -124  -125  -126  -127  -128
     -129  -130  -131  -132  -133  -134
     -135  -136  -137  -138  -139  -140
     -141  -142  -143  -144  -145  -146
     -147  -148  -149  -150  -151  -152
     -153  -154  -155  -156  -157  -158
     -159  -160  -161  -162  -163  -164
     -165  -166  -167  -168  -169  -170
     -171  -172  -173  -174  -175  -176
     -177  -178  -179  -180  -181  -182
     -183  -184  -185  -186  -187  -188
     -189  -190  -191  -192  -193  -194
     -195  -196  -197  -198  -199  -200
     -201  -202  -203  -204  -205  -206
     -207  -208  -209  -210  -211  -212
     -213  -214  -215  -216  -217  -218
     -219  -220  -221  -222  -223  -224
     -225  -226  -227  -228  -229  -230
     -231  -232  -233  -234  -235  -236
     -237  -238  -239  -240  -241  -242
     -243  -244  -245  -246  -247  -248
     -249  -250  -251  -252  -253  -254
     -255  -256  -257  -258  -259  -260
     -261  -262  -263  -264  -265  -266
     -267  -268  -269  -270  -271  -272
     -273  -274  -275  -276  -277  -278
     -279  -280  -281  -282  -283  -284
     -285  -286  -287  -288  -289  -290
     -291  -292  -293  -294  -295  -296
     -297  -298  -299  -300  -301  -302
     -303  -304  -305  -306  -307  -308
     -309  T
tf12
fc22 Radial AziSurFuel Tally Q1 (+x+y)
f22:p +310.1
fm22 4.13583E+14
fs22 -311  -320
     -312  -313  -314  -315  -316  -317
     -318  -319  T
sd22 1.5682E+03  7.8412E+02  8.7124E+01  8r  3.1365E+03
tf22
fc32 Radial AziSurFuel Tally Q2 (-x+y)
f32:p +310.1
fm32 4.13583E+14
fs32 +311  -320
```

```
     +321  +322  +323  +324  +325  +326
     +327  +328  T
sd32 1.5682E+03  7.8412E+02  8.7124E+01  8r  3.1365E+03
tf32
fc42 Radial AziSurFuel Tally Q3 (-x-y)
f42:p +310.1
fm42 4.13583E+14
fs42 +311  +320
     +312  +313  +314  +315  +316  +317
     +318  +319  T
sd42 1.5682E+03  7.8412E+02  8.7124E+01  8r  3.1365E+03
tf42
fc52 Radial AziSurFuel Tally Q4 (+x-y)
f52:p +310.1
fm52 4.13583E+14
fs52 -311  +320
     -321  -322  -323  -324  -325  -326
     -327  -328  T
sd52 1.5682E+03  7.8412E+02  8.7124E+01  8r  3.1365E+03
tf52
fc62 Radial 1ft Tally
f62:p +329.1
fm62 4.13583E+14
fs62  -330  -331  -332  -333  -334  -335
     -336  -337  -338  -339  -340  -341
     -342  -343  -344  -345  -346  -347
     -348  -349  -350  -351  -352  -353
     -354  -355  -356  -357  -358  -359
     -360  -361  -362  -363  -364  -365
     -366  -367  -368  -369  -370  -371
     -372  -373  -374  -375  -376  -377
     -378  -379  -380  -381  -382  -383
     -384  -385  -386  -387  -388  -389
     -390  -391  -392  -393  -394  -395
     -396  -397  -398  -399  -400  -401
     -402  -403  -404  -405  -406  -407
     -408  -409  -410  -411  -412  -413
     -414  -415  -416  -417  -418  -419
     -420  -421  -422  -423  -424  -425
     -426  -427  -428  T
tf62
fc72 Radial 1m Tally
f72:p +429.1
fm72 4.13583E+14
fs72  -430  -431  -432  -433  -434  -435
     -436  -437  -438  -439  -440  -441
     -442  -443  -444  -445  -446  -447
     -448  -449  -450  -451  -452  -453
     -454  -455  -456  -457  -458  -459
     -460  -461  -462  -463  -464  -465
     -466  -467  -468  -469  -470  -471
     -472  -473  -474  -475  -476  -477
     -478  -479  -480  -481  -482  -483
     -484  -485  -486  -487  -488  -489
     -490  -491  -492  -493  -494  -495
     -496  -497  -498  -499  -500  -501
     -502  -503  -504  -505  -506  -507
     -508  -509  -510  -511  -512  -513
     -514  -515  -516  -517  -518  -519
     -520  -521  -522  -523  -524  -525
     -526  -527  -528  T
tf72
fc82 Radial Azilm Tally Q1 (+x+y)
f82:p +529.1
fm82 4.13583E+14
fs82 -530  -539
     -531  -532  -533  -534  -535  -536
     -537  -538  T
sd82 7.0600E+03  3.5300E+03  3.9222E+02  8r  1.4120E+04
tf82
fc92 Radial Azilm Tally Q2 (-x+y)
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```
f92:p +529.1
fm92 4.13583E+14
fs92 +530  -539
     +540  +541  +542  +543  +544  +545
     +546  +547  T
sd92 7.0600E+03  3.5300E+03  3.9222E+02  8r  1.4120E+04
tf92
fc102 Radial Azi1m Tally Q3 (-x-y)
f102:p +529.1
fm102 4.13583E+14
fs102 +530  +539
     +531  +532  +533  +534  +535  +536
     +537  +538  T
sd102 7.0600E+03  3.5300E+03  3.9222E+02  8r  1.4120E+04
tf102
fc112 Radial Azilm Tally Q4 (+x-y)
f112:p +529.1
fm112 4.13583E+14
fs112 -530  +539
     -540  -541  -542  -543  -544  -545
     -546  -547  T
sd112 7.0600E+03  3.5300E+03  3.9222E+02  8r  1.4120E+04
tf112
fc122 Radial 2m Tally
f122:p +548.1
fm122 4.13583E+14
fs122  -549  -550  -551  -552  -553  -554
     -555  -556  -557  -558  -559  -560
     -561  -562  -563  -564  -565  -566
     -567  -568  -569  -570  -571  -572
     -573  -574  -575  -576  -577  -578
     -579  -580  -581  -582  -583  -584
     -585  -586  -587  -588  -589  -590
     -591  -592  -593  -594  -595  -596
     -597  -598  -599  -600  -601  -602
     -603  -604  -605  -606  -607  -608
     -609  -610  -611  -612  -613  -614
     -615  -616  -617  -618  -619  -620
     -621  -622  -623  -624  -625  -626
     -627  -628  -629  -630  -631  -632
     -633  -634  -635  -636  -637  -638
     -639  -640  -641  -642  -643  -644
     -645  -646  -647  T
tf122
fc132 Radial 2m+Convey Tally
f132:p +648.1
fm132 4.13583E+14
fs132  -649  -650  -651  -652  -653  -654
     -655  -656  -657  -658  -659  -660
     -661  -662  -663  -664  -665  -666
     -667  -668  -669  -670  -671  -672
     -673  -674  -675  -676  -677  -678
     -679  -680  -681  -682  -683  -684
     -685  -686  -687  -688  -689  -690
     -691  -692  -693  -694  -695  -696
     -697  -698  -699  -700  -701  -702
     -703  -704  -705  -706  -707  -708
     -709  -710  -711  -712  -713  -714
     -715  -716  -717  -718  -719  -720
     -721  -722  -723  -724  -725  -726
     -727  -728  -729  -730  -731  -732
     -733  -734  -735  -736  -737  -738
     -739  -740  -741  -742  -743  -744
     -745  -746  -747  T
tf132
fc142 Radial Azi2m+Con Tally Q1 (+x+y)
f142:p +748.1
fm142 4.13583E+14
fs142 -749  -758
     -750  -751  -752  -753  -754  -755
     -756  -757  T
```

```
sd142 2.0227E+04  1.0113E+04  1.1237E+03  8r  4.0454E+04
tf142
fc152 Radial Azi2m+Con Tally Q2 (-x+y)
f152:p +748.1
fm152 4.13583E+14
fs152 +749  -758
     +759  +760  +761  +762  +763  +764
     +765  +766  T
sd152 2.0227E+04  1.0113E+04  1.1237E+03  8r  4.0454E+04
tf152
fc162 Radial Azi2m+Con Tally Q3 (-x-y)
f162:p +748.1
fm162 4.13583E+14
fs162 +749  +758
     +750  +751  +752  +753  +754  +755
     +756  +757  T
sd162 2.0227E+04  1.0113E+04  1.1237E+03  8r  4.0454E+04
tf162
fc172 Radial Azi2m+Con Tally Q4 (+x-y)
f172:p +748.1
fm172 4.13583E+14
fs172 -749  +758
     -759  -760  -761  -762  -763  -764
     -765  -766  T
sd172 2.0227E+04  1.0113E+04  1.1237E+03  8r  4.0454E+04
tf172
C
C Print Control
prdmp -30 -60 1 2
print
C Random Number Generator
rand gen=2 seed=33982735979567 stride=152917 hist=1
C
C Rotation Matrix
C
*TR1 0.0 0.0 0.0 10 100 90 -80 10 90 90 90 0
*TR2 0.0 0.0 0.0 20 110 90 -70 20 90 90 90 0
*TR3 0.0 0.0 0.0 30 120 90 -60 30 90 90 90 0
*TR4 0.0 0.0 0.0 40 130 90 -50 40 90 90 90 0
*TR5 0.0 0.0 0.0 50 140 90 -40 50 90 90 90 0
*TR6 0.0 0.0 0.0 60 150 90 -30 60 90 90 90 0
*TR7 0.0 0.0 0.0 70 160 90 -20 70 90 90 90 0
*TR8 0.0 0.0 0.0 80 170 90 -10 80 90 90 90 0
*TR9 0.0 0.0 0.0 90 180 90 0 90 90 90 90 0
*TR10 0.0 0.0 0.0 100 190 90 10 100 90 90 90 0
*TR11 0.0 0.0 0.0 110 200 90 20 110 90 90 90 0
*TR12 0.0 0.0 0.0 120 210 90 30 120 90 90 90 0
*TR13 0.0 0.0 0.0 130 220 90 40 130 90 90 90 0
*TR14 0.0 0.0 0.0 140 230 90 50 140 90 90 90 0
*TR15 0.0 0.0 0.0 150 240 90 60 150 90 90 90 0
*TR16 0.0 0.0 0.0 160 250 90 70 160 90 90 90 0
*TR17 0.0 0.0 0.0 170 260 90 80 170 90 90 90 0
*TR18 0.0 0.0 0.0 180 270 90 90 180 90 90 90 0
*TR19 0.0 0.0 0.0 190 280 90 100 190 90 90 90 0
*TR20 0.0 0.0 0.0 200 290 90 110 200 90 90 90 0
*TR21 0.0 0.0 0.0 210 300 90 120 210 90 90 90 0
*TR22 0.0 0.0 0.0 220 310 90 130 220 90 90 90 0
*TR23 0.0 0.0 0.0 230 320 90 140 230 90 90 90 0
*TR24 0.0 0.0 0.0 240 330 90 150 240 90 90 90 0
*TR25 0.0 0.0 0.0 250 340 90 160 250 90 90 90 0
*TR26 0.0 0.0 0.0 260 350 90 170 260 90 90 90 0
*TR27 0.0 0.0 0.0 270 360 90 180 270 90 90 90 0
*TR28 0.0 0.0 0.0 280 370 90 190 280 90 90 90 0
*TR29 0.0 0.0 0.0 290 380 90 200 290 90 90 90 0
*TR30 0.0 0.0 0.0 300 390 90 210 300 90 90 90 0
*TR31 0.0 0.0 0.0 310 400 90 220 310 90 90 90 0
*TR32 0.0 0.0 0.0 320 410 90 230 320 90 90 90 0
*TR33 0.0 0.0 0.0 330 420 90 240 330 90 90 90 0
*TR34 0.0 0.0 0.0 340 430 90 250 340 90 90 90 0
*TR35 0.0 0.0 0.0 350 440 90 260 350 90 90 90 0
*TR36 0.0 0.0 0.0 360 450 90 270 360 90 90 90 0
```

Figure 5.3.23-8 VISED X-Y Slice – SLOWPOKE Core – Accident Conditions

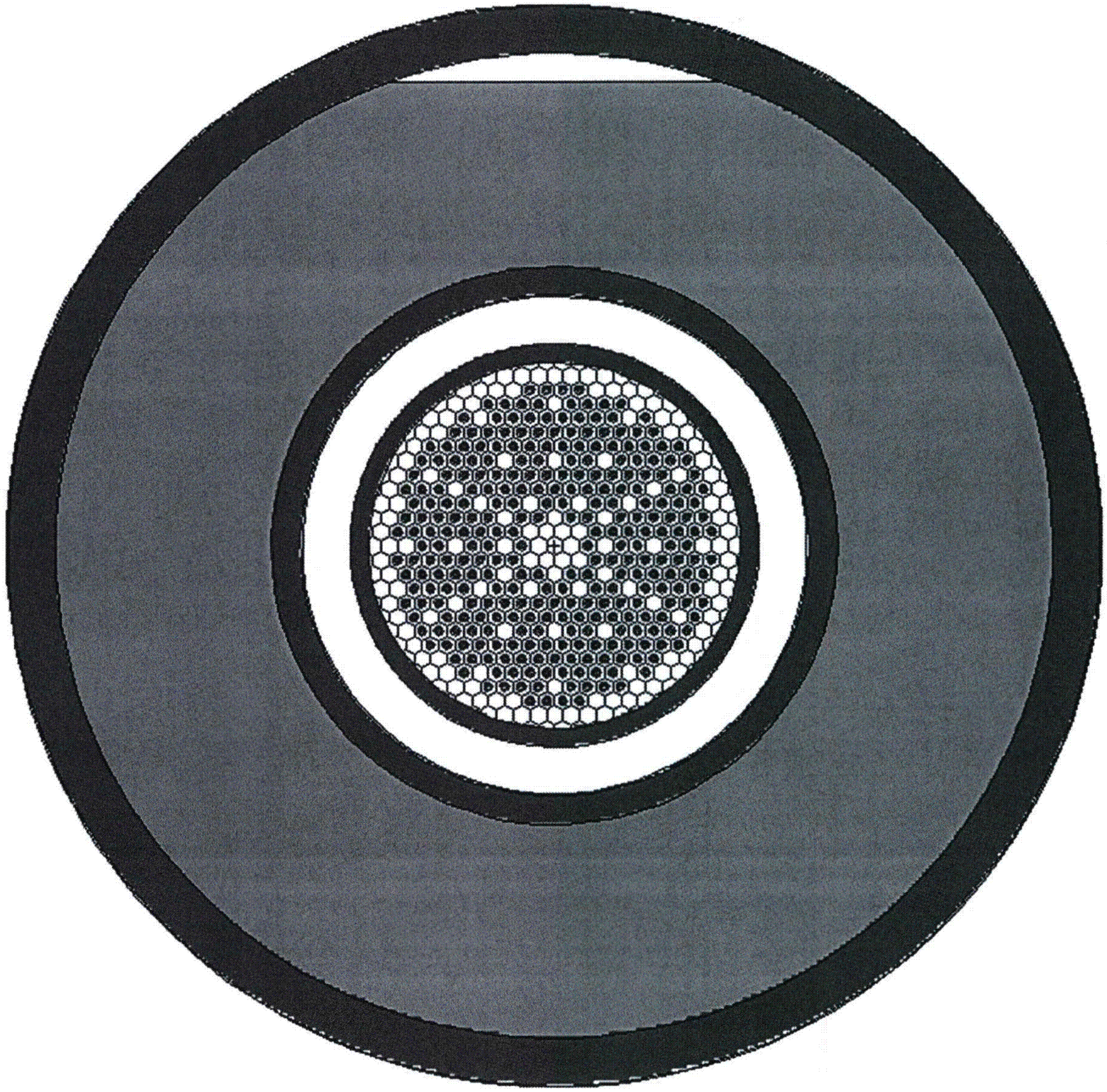
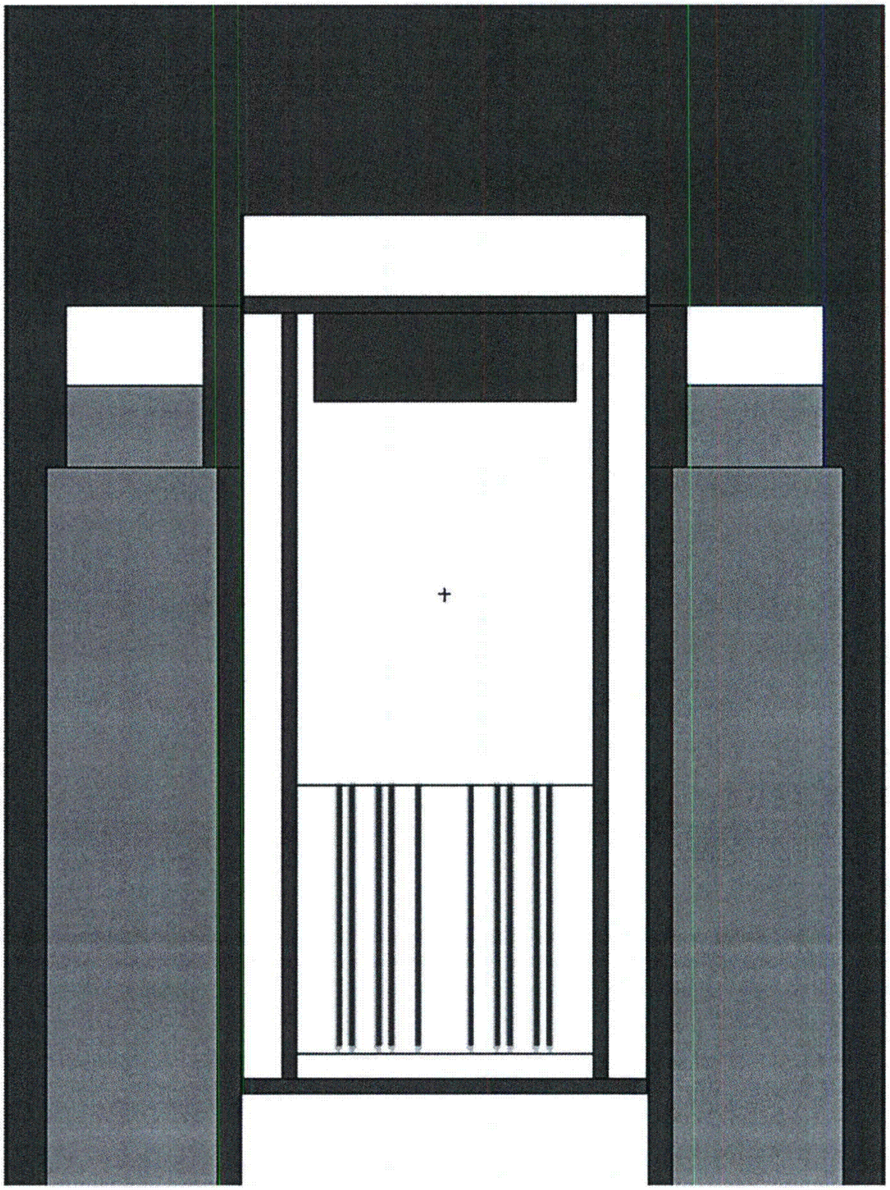


Figure 5.3.23-9 VISED Y-Z Slice – SLOWPOKE – Accident Conditions



Note: SLOWPOKE fuel core basket is shifted to the NAC-LWT lid. Void space indicated by the model is the space occupied by the basket lid collar whose material, but not spacing, is conservatively removed from the model. Location of the fuel core near the bottom of the basket cavity is maintained by a spacer structurally evaluated to survive both normal and accident conditions of transport.

Figure 5.3.23-10 SLOWPOKE Core Basket Sketch

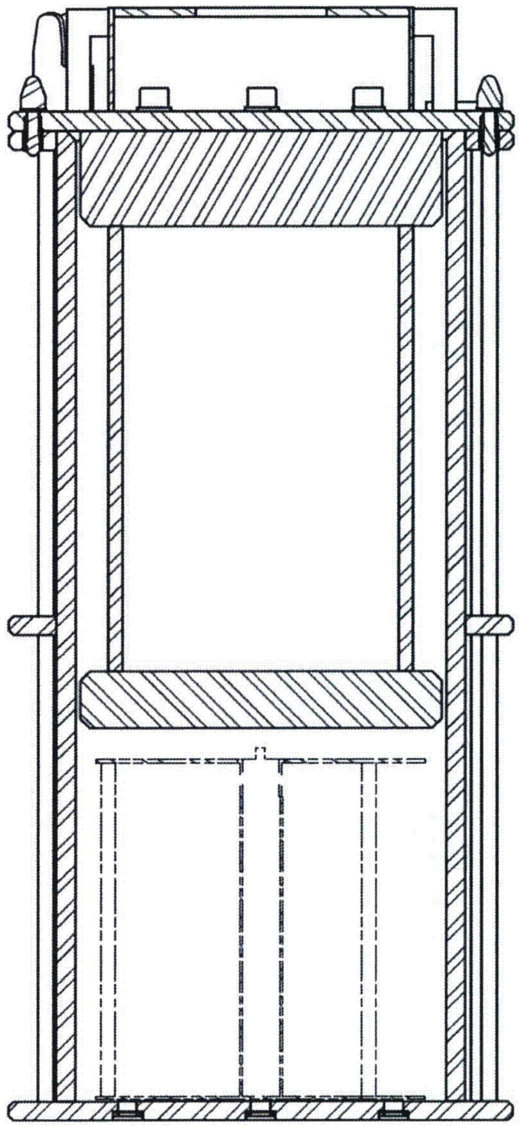


Figure 5.3.23-11 Sample MCNP Input File – Accident Conditions

```
NAC-LWT Cask - 15b90e14d - Accident Transport Conditions
C Top Axial Biasing - Fuel Neutron Source
C Fuel Rod Cells
1 1 -3.4819 -1 +3 -4   u=5 $ Fuel Meat
2 4 -2.7000 -2 +1 3 -4  u=5 $ Clad
3 4 -2.7000 -5 : -6     u=5 $ Top Cap
4 4 -2.7000 -7      u=5 $ Bottom Cap
5 0         #1 #2 #3 #4      u=5 $ Outside Rod
C Fuel Core Array Cell
6 0         -8 u=4 lat=2 fill=-15:15 -15:15 0:0  $ Core Array
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 4 5 5 5 5 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 5 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 4 5 5 5 5 5 5 5 5 5 5 5 4 5 5 5 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 5 4 5 5 4 5 5 4 5 5 4 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 5 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 4
     4 4 4 4 4 4 4 5 5 4 5 5 4 5 4 4 4 5 4 5 5 4 5 5 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 5 5 5 5 5 5 5 5 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 5 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 4 5 5 4 5 5 4 5 5 4 5 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 4 5 5 5 5 5 5 5 5 5 5 5 4 5 5 5 4 4 4 4 4 4 4 4
     4 4 4 4 5 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 5 5 4 5 5 4 5 5 4 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 5 5 5 5 5 5 5 5 5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 5 4 5 5 5 5 5 5 5 5 4 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 5 5 5 4 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
     4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4
C Cells - SLOWPOKE Core Basket
7 0         -9 fill=4 ( 0.0000 0.0000 3.3274 )   u=3 $ SLOWPOKE Core
8 6 -7.9400 -10     u=3 $ Base Plate
9 6 -7.9400 -11 +12    u=3 $ Basket Tube
10 6 -7.9400 -13     u=3 $ Shield Plug
11 6 -7.9400 -14     u=3 $ Lid Plate
12 0         -15     u=3 $ Lid Spacer
13 0         #7 #8 #9 #10 #11 #12     u=3 $ Void
C Cells - LWT Cavity
14 0         -16 fill=3 ( 0.0000 0.0000 377.6980 )   u=2
15 0         #14      u=2
C Cells - LWT Cask Accident Conditions
16 5 -11.344 -20     u=1 $ BotPb
17 0         -19 fill=2    u=1 $ Cavity
18 6 -7.9400 -18 +20    u=1 $ Bottom
19 6 -7.9400 -17 +18 +22 +24 +19 u=1 $ Outer Shell
20 6 -7.9400 -21 +24 +19    u=1 $ Inner Shell Taper
21 6 -7.9400 -23 +19    u=1 $ Inner Shell
22 5 -11.344 -24 +23 -25    u=1 $ Lead
23 5 -11.344 -22 +21 +24 -27 -26 u=1 $ Lead Taper
24 0         -22 +21 +24 +27  u=1 $ Axial Lead Slump
25 0         -24 +25    u=1 $ Radial Lead Slump - Main
26 0         -22 +24 -27 +26  u=1 $ Radial Lead Slump - Taper
27 0         +17     u=1 $ Outside
C Detector Cells - Axial Biasing
100  0  -100  fill=1  $ Surface
200  0  -200  +100  $ 1m
```

```
300  0  -300  +100  +200  $ 2m
400  0  -400  +100  +200  +300  $ 5m
500  0  -500  +100  +200  +300  +400  $ Edge
600  0  -600  +100  +200  +300  +400  +500  $ Driver
700  0  +100  +200  +300  +400  +500  +600  $ Exterior

C Fuel Rod Surfaces
1 CZ 0.2108       $ Fuel Meat OD
2 CZ 0.2616       $ Clad OD
3 PZ 0.4572       $ Lower Fuel Cut Plain
4 PZ 22.4536       $ Upper Fuel Cut Plain
5 RCC 0.0000 0.0000 0.3302 0.0000 0.0000 0.1270 0.3048 $ Lower Stud Rim
6 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 0.3302 0.1930 $ Lower Stud Cap
7 RCC 0.0000 0.0000 22.4536 0.0000 0.0000 0.3810 0.3048 $ Upper Stud
C Fuel Core Lattice Surface
8 RHP 0.0 0.0 0.0 0.0 0.0 22.8346
         0.5518 0.0 0.0    $ Lattice Cell
C Surfaces - SLOWPOKE Core Basket
9 RCC 0.0000 0.0000 3.3274 0.0000 0.0000 22.8346 12.3824 $ Core
10 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 1.2700 16.8466 $ Base Plate
11 RCC 0.0000 0.0000 1.2700 0.0000 0.0000 64.8970 13.6525 $ Tube OD
12 RCC 0.0000 0.0000 1.2700 0.0000 0.0000 64.8970 12.3825 $ Tube ID
13 RCC 0.0000 0.0000 58.5470 0.0000 0.0000 7.6200 10.9982 $ Shield Plug
14 RCC 0.0000 0.0000 66.1670 0.0000 0.0000 1.2700 16.8466 $ Lid Plate
15 RCC 0.0000 0.0000 67.4370 0.0000 0.0000 6.9850 16.84655 $ Lid Spacer
C Surfaces - LWT Cavity
16 RCC 0.0000 0.0000 377.6980 0.0000 0.0000 74.4220 16.8467 $ Basket
C Surfaces - LWT Cask Accident Conditions
17 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 507.3650 36.5189 $ Lwt
18 RCC 0.0000 0.0000 -26.6700 0.0000 0.0000 26.6700 36.5189 $ Bottom
19 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 452.1200 16.9863 $ Cavity
20 RCC 0.0000 0.0000 -17.7800 0.0000 0.0000 7.6200 26.3525 $ Bottom Gamma Shield
21 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 20.1740 $ Lead ID - Taper
22 RCC 0.0000 0.0000 0.0000 0.0000 0.0000 444.5000 31.5976 $ Lead OD - Taper
23 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 18.9103 $ Lead ID
24 RCC 0.0000 0.0000 13.8176 0.0000 0.0000 416.8648 33.3271 $ Lead OD
25 PY 31.4618       $ Radial Slump - Main
26 PY 31.4618       $ Radial Slump - Taper
27 PZ 437.6266       $ Top Lead Slump
C Axial Detector DTA (Surface)
100  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  507.5650  36.5190
101  CZ  7.3038
102  CZ  14.6076
103  CZ  21.9114
104  CZ  29.2152
C Axial Detector DTB (1m)
200  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  607.5650  136.5190
201  CZ  27.3038
202  CZ  54.6076
203  CZ  81.9114
204  CZ  109.2152
C Axial Detector DTC (2m)
300  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  707.5650  236.5190
301  CZ  47.3038
302  CZ  94.6076
303  CZ  141.9114
304  CZ  189.2152
C Axial Detector DTD (5m)
400  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  1007.5650  236.6190
401  CZ  47.3238
402  CZ  94.6476
403  CZ  141.9714
404  CZ  189.2952
C Axial Detector DTE (Edge)
500  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  1091.6650  236.7190
501  CZ  47.3438
502  CZ  94.6876
503  CZ  142.0314
504  CZ  189.3752
C Axial Detector DTF (Driver)
600  RCC  0.0000  0.0000  -26.7700  0.0000  0.0000  1244.0650  236.8190
```

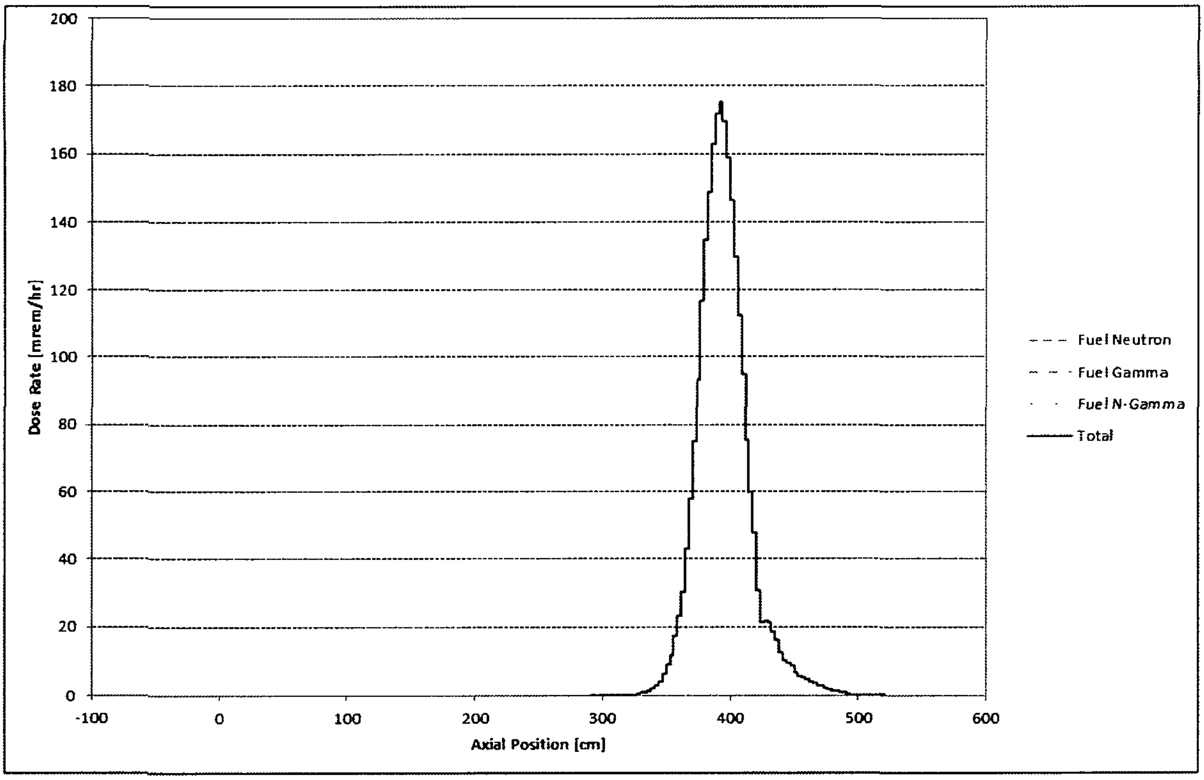
```
601  CZ  47.3638
602  CZ  94.7276
603  CZ  142.0914
604  CZ  189.4552

C
C Materials List
C
C U-Al Fuel
m1    92235 -2.5201E-01 92238 -2.8001E-02 13027 -7.1999E-01
C Water
m2    1001 6.6667E-01 8016 3.3333E-01
mt2   lwtr.01
C Water/Glycol
m3    1001 -1.03200E-01 8016 -6.82400E-01 6000 -2.14400E-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:n 1 32r 0
C
C Source Definition - Fuel Neutron
C 15 GWd/MTU burnup, 90 wt% U-235, 14-day cool time, 2.786 g U-235 per rod, 39 W/core
sdef RAD=d1 EXT=d2 ERG=d3 cell=d4
     POS= 0.0000 0.0000 0.4572
     AXS= 0.0000 0.0000 1.0000
si1 0 0.2108
sp1 -21 1
si2 0 21.9964
sp2 0 1
si3   1.000E-11 1.000E-08 3.000E-08 5.000E-08 1.000E-07 2.250E-07
     3.250E-07 4.140E-07 8.000E-07 1.000E-06 1.130E-06 1.300E-06
     1.860E-06 3.060E-06 1.070E-05 2.900E-05 1.010E-04 5.830E-04
     3.040E-03 1.500E-02 1.110E-01 4.080E-01 9.070E-01 1.420E+00
     1.830E+00 3.010E+00 6.380E+00 2.000E+01
sp3 0.0000E+00 7.7644E-17 1.7532E-16 3.4335E-14 7.9442E-14 2.6321E-13
     4.0690E-13 2.2042E-13 2.1252E-12 1.5718E-12 8.2450E-13 1.3824E-12
     6.4949E-12 1.6664E-11 1.8663E-10 2.4523E-09 1.5003E-08 1.2291E-06
     1.2619E-05 1.5888E-04 5.3354E-03 4.7898E-02 9.6974E-02 1.1373E-01
     6.2345E-02 2.3453E-02 1.4419E-03 8.5813E-05
# SI4  SP4
     L  D
     100:17:14:7:6:-1  1.0000
mode n
nps 6.40E+06
C
C ANSI/ANS-6.1.1-1977 - Neutron Flux-to-Dose Conversion Factors
C (mrem/hr)/(neutrons/cm2-sec)
de0 2.5E-08 1E-07 1E-06 0.00001 0.0001 0.001 0.01
     0.1 0.5 1 2.5 5 7 10
     14 20
df0 3.67E-03 3.67E-03 4.46E-03 4.54E-03 4.18E-03 3.76E-03 3.56E-03
     2.17E-02 9.26E-02 1.32E-01 1.25E-01 1.56E-01 1.47E-01 1.47E-01
     2.08E-01 2.27E-01
C
C Weight Window Generation - Top Axial
wwg 2 0 0 0 0
wwp:n 5 3 5 0 -1 0
mesh geom=cyl ref=0 0 400 origin=0.1 0.1 -527
     imesh 16.8 17.0 18.9 33.3 36.5 536.5
     iints 1 1 1 1 1 1
     jmesh 500 509 517 527 906 928 979 1007 1507
     jints 1 1 1 1 1 12 1 1 1
     kmesh 1
```

```
     kints 1
wwge:n  1e-5 1e-3 1 20
fc2 Axial Surface Tally
f2:n +100.2
fm2 1.04728E+02
fs2  -101  -102  -103  -104  T
tf2
fc12 Axial 1m Tally
f12:n +200.2
fm12 1.04728E+02
fs12  -201  -202  -203  -204  T
tf12
fc22 Axial 2m Tally
f22:n +300.2
fm22 1.04728E+02
fs22  -301  -302  -303  -304  T
tf22
fc32 Axial 5m Tally
f32:n +400.2
fm32 1.04728E+02
fs32  -401  -402  -403  -404  T
tf32
fc42 Axial Edge Tally
f42:n +500.2
fm42 1.04728E+02
fs42  -501  -502  -503  -504  T
tf42
fc52 Axial Driver Tally
f52:n +600.2
fm52 1.04728E+02
fs52  -601  -602  -603  -604  T
tf52
C
C Print Control
prdmp -30 -60 1 2
print
C Random Number Generator
rand gen=2 seed=46929924663793 stride=152917 hist=1
C
C Rotation Matrix
C
*TR1 0.0 0.0 0.0 10 100 90 -80 10 90 90 90 0
*TR2 0.0 0.0 0.0 20 110 90 -70 20 90 90 90 0
*TR3 0.0 0.0 0.0 30 120 90 -60 30 90 90 90 0
*TR4 0.0 0.0 0.0 40 130 90 -50 40 90 90 90 0
*TR5 0.0 0.0 0.0 50 140 90 -40 50 90 90 90 0
*TR6 0.0 0.0 0.0 60 150 90 -30 60 90 90 90 0
*TR7 0.0 0.0 0.0 70 160 90 -20 70 90 90 90 0
*TR8 0.0 0.0 0.0 80 170 90 -10 80 90 90 90 0
*TR9 0.0 0.0 0.0 90 180 90 0 90 90 90 90 0
*TR10 0.0 0.0 0.0 100 190 90 10 100 90 90 90 0
*TR11 0.0 0.0 0.0 110 200 90 20 110 90 90 90 0
*TR12 0.0 0.0 0.0 120 210 90 30 120 90 90 90 0
*TR13 0.0 0.0 0.0 130 220 90 40 130 90 90 90 0
*TR14 0.0 0.0 0.0 140 230 90 50 140 90 90 90 0
*TR15 0.0 0.0 0.0 150 240 90 60 150 90 90 90 0
*TR16 0.0 0.0 0.0 160 250 90 70 160 90 90 90 0
*TR17 0.0 0.0 0.0 170 260 90 80 170 90 90 90 0
*TR18 0.0 0.0 0.0 180 270 90 90 180 90 90 90 0
*TR19 0.0 0.0 0.0 190 280 90 100 190 90 90 90 0
*TR20 0.0 0.0 0.0 200 290 90 110 200 90 90 90 0
*TR21 0.0 0.0 0.0 210 300 90 120 210 90 90 90 0
*TR22 0.0 0.0 0.0 220 310 90 130 220 90 90 90 0
*TR23 0.0 0.0 0.0 230 320 90 140 230 90 90 90 0
*TR24 0.0 0.0 0.0 240 330 90 150 240 90 90 90 0
*TR25 0.0 0.0 0.0 250 340 90 160 250 90 90 90 0
*TR26 0.0 0.0 0.0 260 350 90 170 260 90 90 90 0
*TR27 0.0 0.0 0.0 270 360 90 180 270 90 90 90 0
*TR28 0.0 0.0 0.0 280 370 90 190 280 90 90 90 0
*TR29 0.0 0.0 0.0 290 380 90 200 290 90 90 90 0
*TR30 0.0 0.0 0.0 300 390 90 210 300 90 90 90 0
```

```
*TR31 0.0 0.0 0.0 310 400 90 220 310 90 90 90 0
*TR32 0.0 0.0 0.0 320 410 90 230 320 90 90 90 0
*TR33 0.0 0.0 0.0 330 420 90 240 330 90 90 90 0
*TR34 0.0 0.0 0.0 340 430 90 250 340 90 90 90 0
*TR35 0.0 0.0 0.0 350 440 90 260 350 90 90 90 0
*TR36 0.0 0.0 0.0 360 450 90 270 360 90 90 90 0
```

Figure 5.3.23-12 Normal Condition Radial Surface Dose Rate Profile by Source Type – SLOWPOKE Core



Note: Results based on tally located at the radius of the cask neutron shield shell. Peak dose rates occur at the SLOWPOKE core centerline elevation. Dose rates at the cask surface between impact limiter and neutron shield shell are lower than at the fuel core midplane.

Figure 5.3.23-13 Normal Condition 2-m + Conveyance Radial Surface Dose Rate Profile by Source Type – SLOWPOKE Core

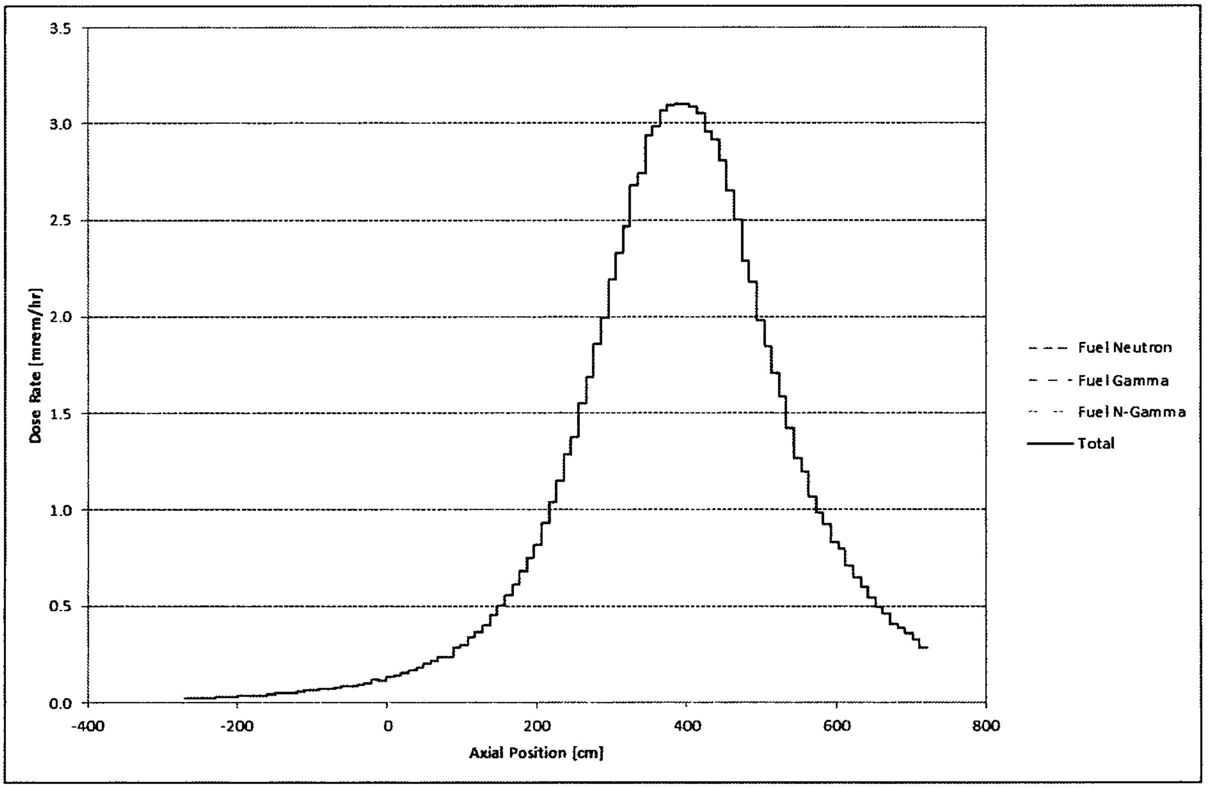
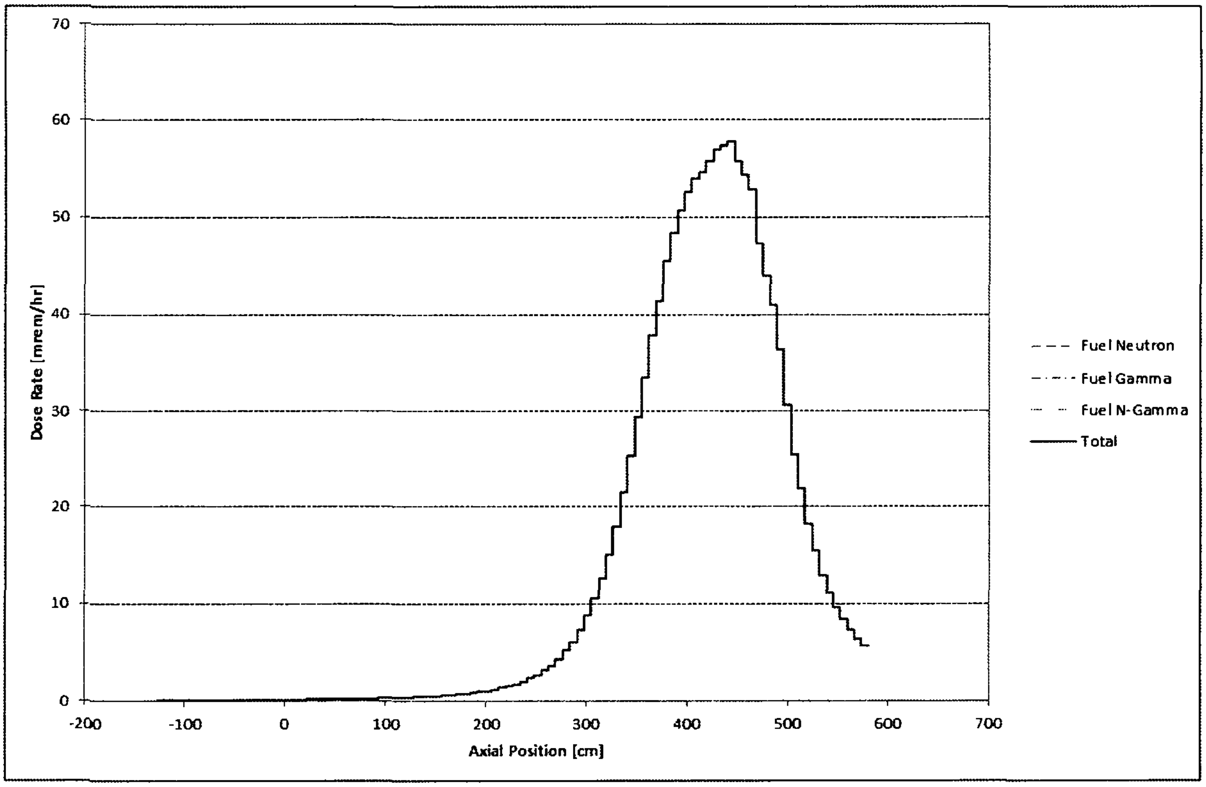


Figure 5.3.23-14 Accident Condition Radial 1m Dose Rate Profile by Source Type – SLOWPOKE Core



Note: Dose rates are circumferential average. Maximum dose rate of 80.7 mrem/hr was calculated using an azimuthal tally at the location of the lead slump.

Figure 5.3.23-15 Accident Condition Radial 1m Dose Rate Azimuthal Profile at Fuel Height – SLOWPOKE Core

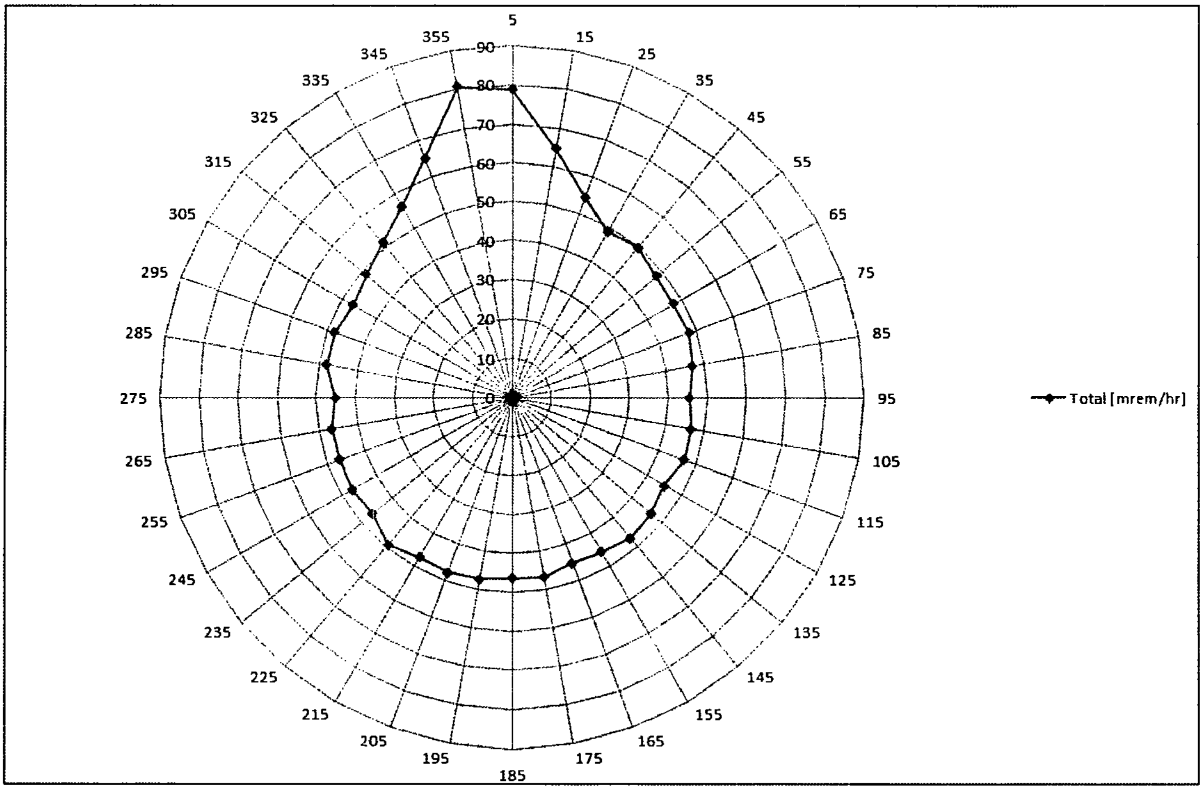


Table 5.3.23-1 SLOWPOKE Fuel Geometry and Materials

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	2.795
Weight of total U	g	3.0
Alloy or compound material weight	g	7.714
Total weight of fuel meat	g	10.714
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	0.52
Total weight of fuel element	g	15.695
Enrichment %	%	93
Core Maximum Power	kW	20
Maximum Burnup (^{235}U depletion)	%	0.65

Table 5.3.23-2 Source Term Generation Parameters for SLOWPOKE Fuel

Parameter	Value
U Mass Per Rod (grams)	3.121
^{235}U per Core (grams)	837.1
Core Power (kW)	20
Number of Hours Burned	16752
Number of Days Cooled	14
Number of Rods / Core	298
Initial Enrichment (wt % ^{235}U)	90
Burnup (% ^{235}U)	2.12
Burnup (GWd/MTU)	15
Moderator/Box Temperature (C)	40
Clad Temperature (C)	90
Fuel Temperature (C)	100

Table 5.3.23-3 SLOWPOKE Neutron Source Term (per rod)

Group	E Lower [MeV]	E Upper [MeV]	Source [neutrons/sec]
1	6.380E+00	2.000E+01	8.581E-05
2	3.010E+00	6.380E+00	1.442E-03
3	1.830E+00	3.010E+00	2.345E-02
4	1.420E+00	1.830E+00	6.234E-02
5	9.070E-01	1.420E+00	1.137E-01
6	4.080E-01	9.070E-01	9.697E-02
7	1.110E-01	4.080E-01	4.790E-02
8	1.500E-02	1.110E-01	5.335E-03
9	3.040E-03	1.500E-02	1.589E-04
10	5.830E-04	3.040E-03	1.262E-05
11	1.010E-04	5.830E-04	1.229E-06
12	2.900E-05	1.010E-04	1.500E-08
13	1.070E-05	2.900E-05	2.452E-09
14	3.060E-06	1.070E-05	1.866E-10
15	1.860E-06	3.060E-06	1.666E-11
16	1.300E-06	1.860E-06	6.495E-12
17	1.130E-06	1.300E-06	1.382E-12
18	1.000E-06	1.130E-06	8.245E-13
19	8.000E-07	1.000E-06	1.572E-12
20	4.140E-07	8.000E-07	2.125E-12
21	3.250E-07	4.140E-07	2.204E-13
22	2.250E-07	3.250E-07	4.069E-13
23	1.000E-07	2.250E-07	2.632E-13
24	5.000E-08	1.000E-07	7.944E-14
25	3.000E-08	5.000E-08	3.433E-14
26	1.000E-08	3.000E-08	1.753E-16
27	1.000E-11	1.000E-08	7.764E-17
Total			3.514E-01

Table 5.3.23-4 SLOWPOKE Fuel Gamma Source Term (per rod)

Group	E Lower [MeV]	E Upper [MeV]	Source [photons/sec]
1	8.00E+00	1.00E+01	7.4653E-06
2	6.50E+00	8.00E+00	3.4862E-05
3	5.00E+00	6.50E+00	1.8338E-04
4	4.00E+00	5.00E+00	7.9396E-04
5	3.00E+00	4.00E+00	2.6553E+07
6	2.50E+00	3.00E+00	3.4304E+09
7	2.00E+00	2.50E+00	2.0105E+09
8	1.66E+00	2.00E+00	3.5978E+08
9	1.33E+00	1.66E+00	1.0587E+11
10	1.00E+00	1.33E+00	2.5592E+09
11	8.00E-01	1.00E+00	4.1310E+10
12	6.00E-01	8.00E-01	4.3248E+11
13	4.00E-01	6.00E-01	1.5811E+11
14	3.00E-01	4.00E-01	5.7276E+10
15	2.00E-01	3.00E-01	2.3964E+10
16	1.00E-01	2.00E-01	1.3380E+11
17	4.50E-02	1.00E-01	1.0982E+11
18	1.00E-02	4.50E-02	3.1684E+11
Total			1.3879E+12

Table 5.3.23-5 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	0.94	5.7965E-02
	O		2.4151E-02
	C		1.0105E-02
Impact Limiter	Al	0.50	1.1153E-02

Table 5.3.23-6 Modeled SLOWPOKE Core Basket Dimensions

Description	Dimension [in]
Core Basket Base Plate Thickness	0.500
Core Basket Base Plate OD	13.265
Core Basket Spacer Height	12.750
Core Basket Lid Height	3.250
Core Basket Lid Plate Thickness	0.500
Core Basket Lid OD	13.265
Core Basket Shield Thickness	3.000
Core Basket Shield OD	8.660
Core Basket Tube Height	25.550
Core Basket Tube OD	10.750
Core Basket Tube Thickness	0.500

Table 5.3.23-7 Maximum Dose Rates for SLOWPOKE Fuel

Transport Condition	Dose Rate Location	Maximum [mrem/hr]	Maximum FSD
Normal	Side Surface of Cask	175.1	0.9%
	Top Surface of Cask	3.1	1.2%
	Bottom Surface of Cask	0.29	8.2%
	Side 1ft	57.2	0.8%
	Side 1m (Transport Index)	15.2	1.0%
	2m from Truck - Radial	3.1	1.1%
	2m from Top	0.24	1.2%
	Edge of Truck - Top	0.045	2.5%
	Edge of Truck - Bottom	0.018	8.4%
	Dose at Cab of Truck	0.031	3.3%
Accident	Side Surface of Cask	2127	1.7%
	Top Surface of Cask	16.3	0.8%
	Bottom Surface of Cask	1.2	9.8%
	Side 1m	80.7	2.1%
	Top 1m	2.0	26.6%
	Bottom 1m	0.15	9.9%

Note: The bounding accident side 1 meter dose rate is taken from the azimuthal tally at the fuel height.

Table 5.3.23-8 Summarized Maximum Dose Rates for SLOWPOKE Fuel

Transport Condition	Dose Rate Location	Maximum [mrem/hr]	Limit [mrem/hr]
Normal	Side Surface of Cask	175.1	1000
	Side 1ft	57.2	200
	Side 1m (Transport Index)	15.2	N/A
	2m from Truck - Radial	3.1	10
Accident	Side 1m	80.7	1000

Note: The side 1 ft detector is closer to the cask than the edge of the conveyance where the 200 mrem/hr limit occurs.

February 2015

Revision LWT-15A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Volume 3 of 3

NON-PROPRIETARY VERSION

Docket No. 71-9225

NAC INTERNATIONAL

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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; up to 800 SLOWPOKE rods; up to 18 NRU or NRX fuel assemblies; 4 HEUNL containers; or one SLOWPOKE fuel core. 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
NRU and NRX	100
HEUNL containers	0.0
SLOWPOKE Fuel Core	100

* 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).

Analyses are performed on the NAC-LWT with 18 NRU or NRX fuel assemblies. Section 6.7.3 presents the methods and MCNP 5 models used in the analyses. Section 6.7.3.3 also presents the

criticality analysis results of the NAC-LWT cask loaded with the NRU/NRX payload. Criticality of the NAC-LWT cask with the most reactive configuration is evaluated. The fuel assemblies are assumed to be unburned. A single cask is analyzed. The results of the analysis show that the $k_{eff} + 2\sigma$ of the NAC-LWT cask with the most reactive NRU/NRX configuration under normal and accident conditions is below the upper safety limit (USL) for highly enriched uranium (HEU) fuel.

Analyses are performed on the NAC-LWT with 4 HEUNL containers. The HEUNL material is permitted with up to 7.40 g/L ^{235}U at a maximum ^{235}U enrichment of 93.4 wt%. The evaluated payload considers a bounding container volume of 64.3 L (17.0 gal). Due to void volume in the container that allows HEUNL thermal expansion, actual container capacity is less. All evaluation detail, including input, method, and analysis results are included in Section 6.7.4. The criticality benchmark for this material is provided in Section 6.5.7. Criticality of the NAC-LWT cask with the most reactive configuration is evaluated. Considered in the most reactive configuration is the uranyl nitrate (other nitrates separated) at optimal H/U. The results show that the bias adjusted k_{eff} of an infinite array of NAC-LWT casks with the most reactive HEUNL configuration under normal and accident conditions is below the upper safety limit (USL) for highly enriched uranyl nitrates.

Analyses are performed on the NAC-LWT with one SLOWPOKE fuel core. The SLOWPOKE fuel core is permitted to contain up to 298 SLOWPOKE fuel rods with up to 95.3 wt % ^{235}U initial enrichment. The payload consists of undamaged fuel. All evaluation details, including input, method, and analysis results, are included in Section 6.7.5. The criticality benchmark analysis for this material is shown in Section 6.5.5. Included in Section 6.7.5 are the fuel core geometry and material description, the MCNP model used in the basket, and the criticality analysis results of the NAC-LWT loaded with one SLOWPOKE fuel core. The fuel is assumed to be fresh, i.e., no burnup credit. With the exception of the normal condition array, a single cask is analyzed. The fuel rod pitch is optimized. Variation of moderator density is considered. This includes preferential flooding evaluations of the basket that contains the fuel core. The results show that the bias adjusted k_{eff} of an NAC-LWT cask at maximum reactivity fuel rod pitch and at maximum reactivity interspersed moderation is below the upper safety limit (USL).

6.2 Package Fuel Loading

The NAC-LWT cask can safely transport 1 PWR assembly, up to 25 intact PWR or BWR rods in a rod holder or fuel assembly lattice, up to 25 PWR or BWR rods with up to 14 of the fuel rods classified as damaged in a rod holder, 2 BWR assemblies, 15 sound metallic fuel rods, 6 failed metallic fuel rods, up to 42 MTR fuel elements, up to 140 TRIGA fuel elements, up to 560 TRIGA fuel cluster rods, up to 42 DIDO fuel assemblies, two General Atomics Irradiated Fuel Material packages, up to 300 TPBARs (of which two can be damaged), up to 700 PULSTAR fuel elements, up to 42 spiral fuel assemblies, up to 42 MOATA plate bundles, up to 800 SLOWPOKE rods, up to 18 AECL NRU or NRX fuel assemblies, or one SLOWPOKE fuel core. The characteristics for payloads containing fissile material are presented in the following sections. Fresh fuel is conservative because the fuel becomes less reactive as burnup increases. Burnable poisons, such as the gadolinium rods sometimes used in BWR assemblies, are ignored for conservatism.

TPBARs are stainless steel clad rods containing $LiAlO_2$ absorber pellets and nickel-plated Zircaloy getter tube or nickel-plated zirconium (NPZ) alloy spacer tubes with no absorber pellets. The TPBARs do not contain any fissile material.

6.7.5 SLOWPOKE Fuel Core

This section includes input, analysis method, results, and criticality benchmark evaluations for the NAC-LWT cask containing a payload of the SLOWPOKE fuel core. The SLOWPOKE fuel core may contain up to 298 undamaged SLOWPOKE fuel rods. SLOWPOKE rods are aluminum clad and contain highly enriched uranium in an aluminum matrix material. Rods are located in a hexagonal pitch configuration using an upper and lower aluminum plate into which the rod end plugs are inserted. This section evaluates criticality for a fuel core in a SLOWPOKE fuel core basket. Loose SLOWPOKE rods in canisters are evaluated in Section 6.7.2.

6.7.5.1 Package Fuel Loading

The NAC-LWT cask may transport a SLOWPOKE fuel core. The core is packaged inside the SLOWPOKE fuel core basket. Characteristics of the SLOWPOKE fuel core are presented in Table 6.7.5-1. Key characteristics for the criticality safety evaluation of the SLOWPOKE fuel core package are initial enrichment and fissile material (^{235}U) mass. Maximizing the number of fuel rods, ^{235}U content per rods and ^{235}U enrichment will produce maximum fissile material mass while reducing parasitic absorption in ^{238}U.

6.7.5.2 Criticality Model Specifications

This section describes the models that are used in the criticality analyses for the NAC-LWT cask containing a SLOWPOKE fuel core. The models are analyzed separately under normal conditions and hypothetical accident conditions to ensure that all possible configurations are subcritical. The input file for the maximum reactivity case is provided in Section 6.6.19.

Each model uses the MCNP5 Version 1.30 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 SLOWPOKE fuel core payload is built using only the SLOWPOKE fuel rod array and basket. Fuel core components not modeled include the center tube and upper and lower plates. These components are considered removed/damaged for modifying the fuel rod pitch and re-arrangement of the rods for maximum reactivity. The upper and lower plates are thin (0.1-inch) aluminum plates that will not significantly influence system reactivity. The center tube is a thin wall (0.043-inch wall thickness) perforated aluminum tube which will also not significantly influence system reactivity.

The fuel rods, basket, cask cavity, and radial shields are explicitly modeled as shown in Figure 6.7.5-1 model sketch. The axial geometry of the basket with SLOWPOKE fuel core is assumed for all studies. The licensed basket includes a stainless steel lid with two thick plates with a spacer in-between that distances the core from the cask top. The lid mounting plate is 2.5 inches thick, and the lid shield plate is 1.5 inches thick. The modeled lid is 3.5 inches of stainless steel with the fuel core located directly below. This positions the core near the LWT lid which is 11.25 inches of stainless steel reflection. Based on the results of the moderator density studies in Section 6.3.3, there is no moderator between the LWT lid and basket. The stainless steel bottom plate is modeled away from the core. The reflection from the plate (0.50-inch thick) is not significant to the model. The modeled basket is shown to be conservative in Section 6.7.5.3.9.

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. Each universe is developed independently as surfaces and cells. The basket 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 basket cavity).

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. Cask exterior conditions for a single cask evaluation are shown to have negligible impact in Section 6.7.5.3.1. For the 10 CFR 71.59 normal condition array evaluation, the reduced diameter allows a more closely packed array.

Sample VISED sketches of the assembled geometry are shown in Figure 6.7.5-2 and Figure 6.7.5-3. Additional fuel rod arrangements are evaluated in Section 6.7.5.3.2. The cask outer surface is surrounded by a cylindrical body with the option of applying reflecting boundary conditions. For single cask analysis, the cask is surrounded by 30 cm of water to apply full water reflection. A single cask is evaluated for all evaluations with the exception of the 10 CFR 71.59 (a.1) evaluation.

Package Regional Densities

The composition densities (g/cc) and nuclide number densities (atoms/b-cm) used in the subsequent criticality analyses are shown in Table 6.7.5-2. The NAC-LWT neutron shield contains soluble boron. Modeled neutron shield material does not include boron (removal of neutron absorber).

Figure 6.7.5-1 MCNP Model Sketch of the NAC-LWT Cask with SLOWPOKE Fuel Core

(Dimensions in inches)

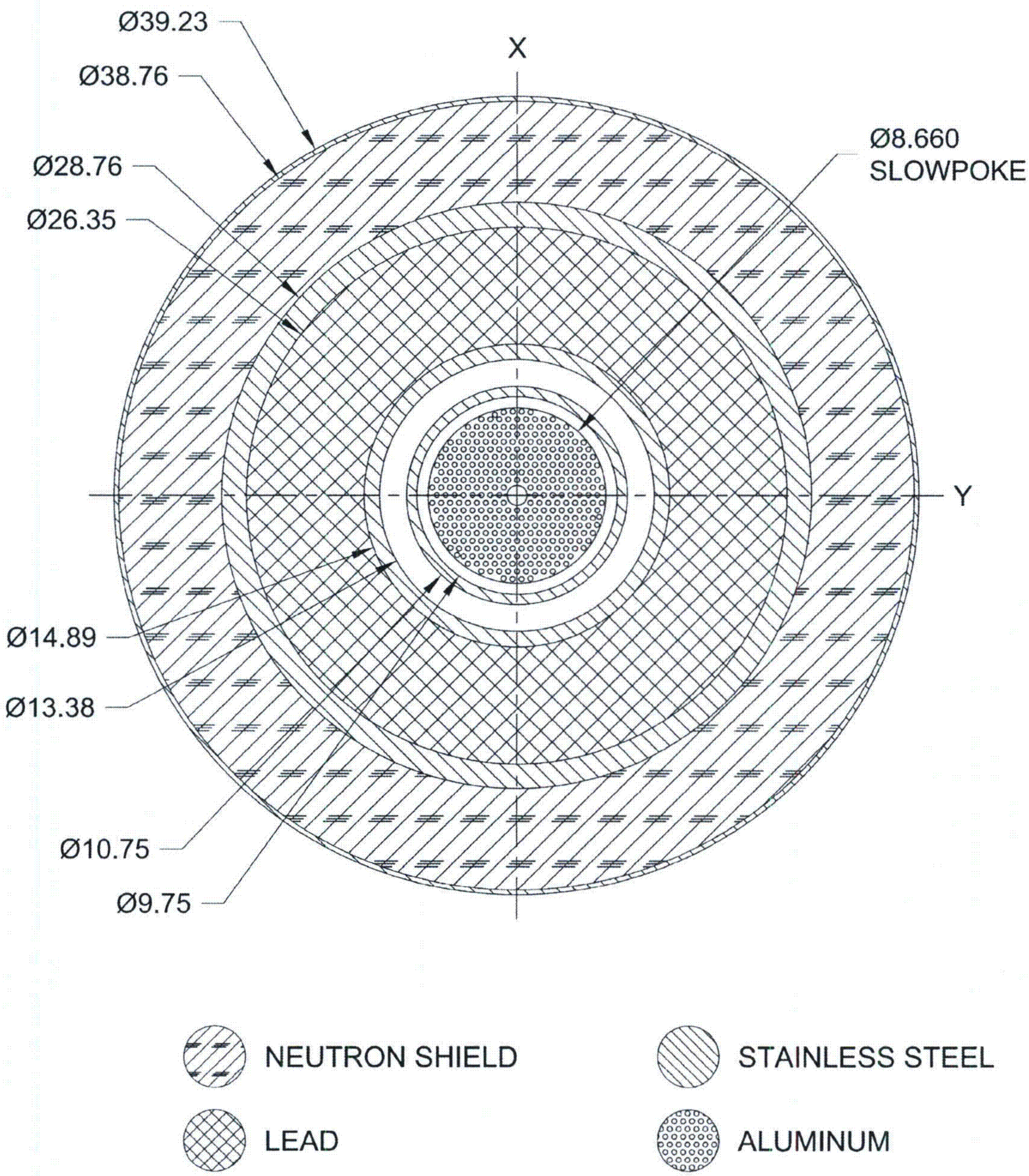


Figure 6.7.5-2 VISED Sketch of LWT Radial View – SLOWPOKE Fuel Core

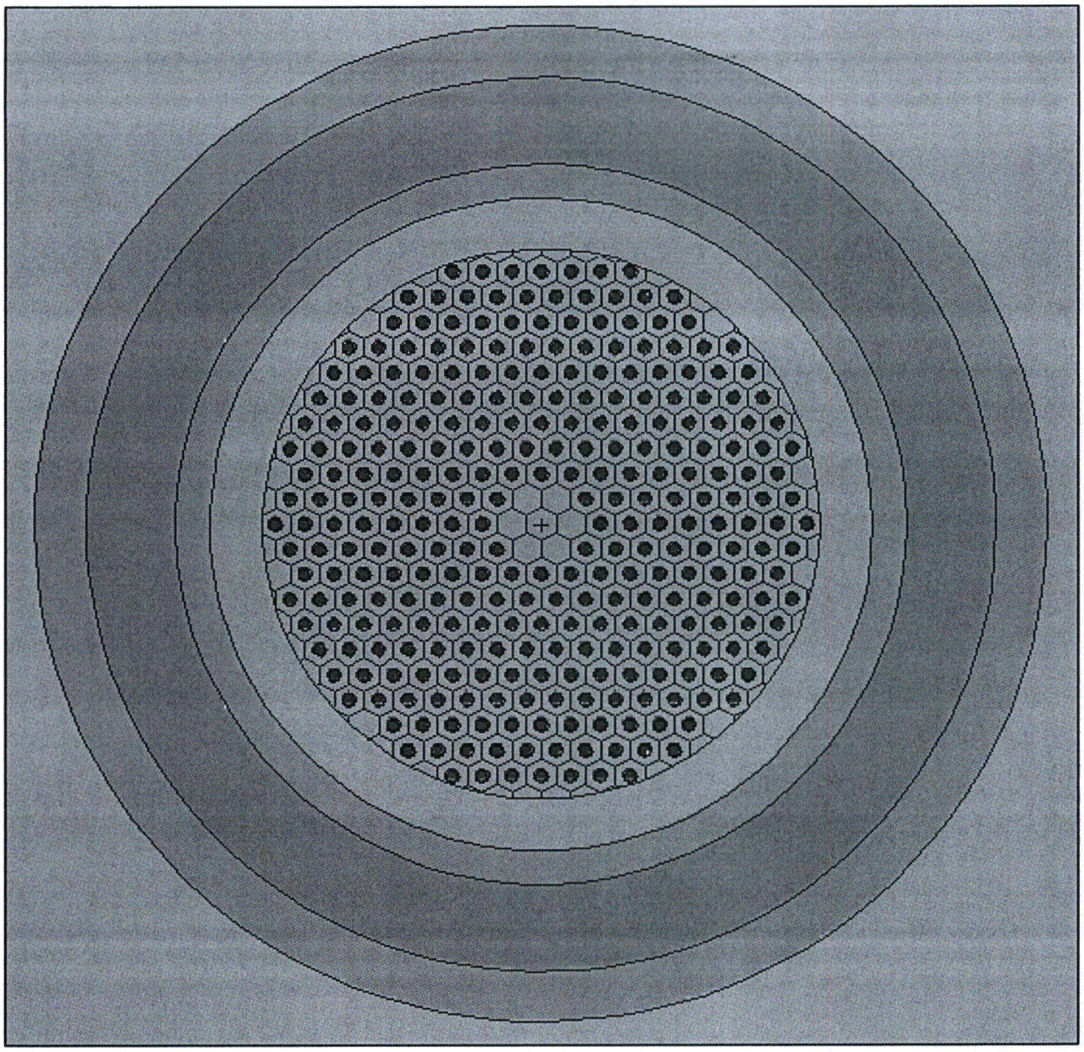
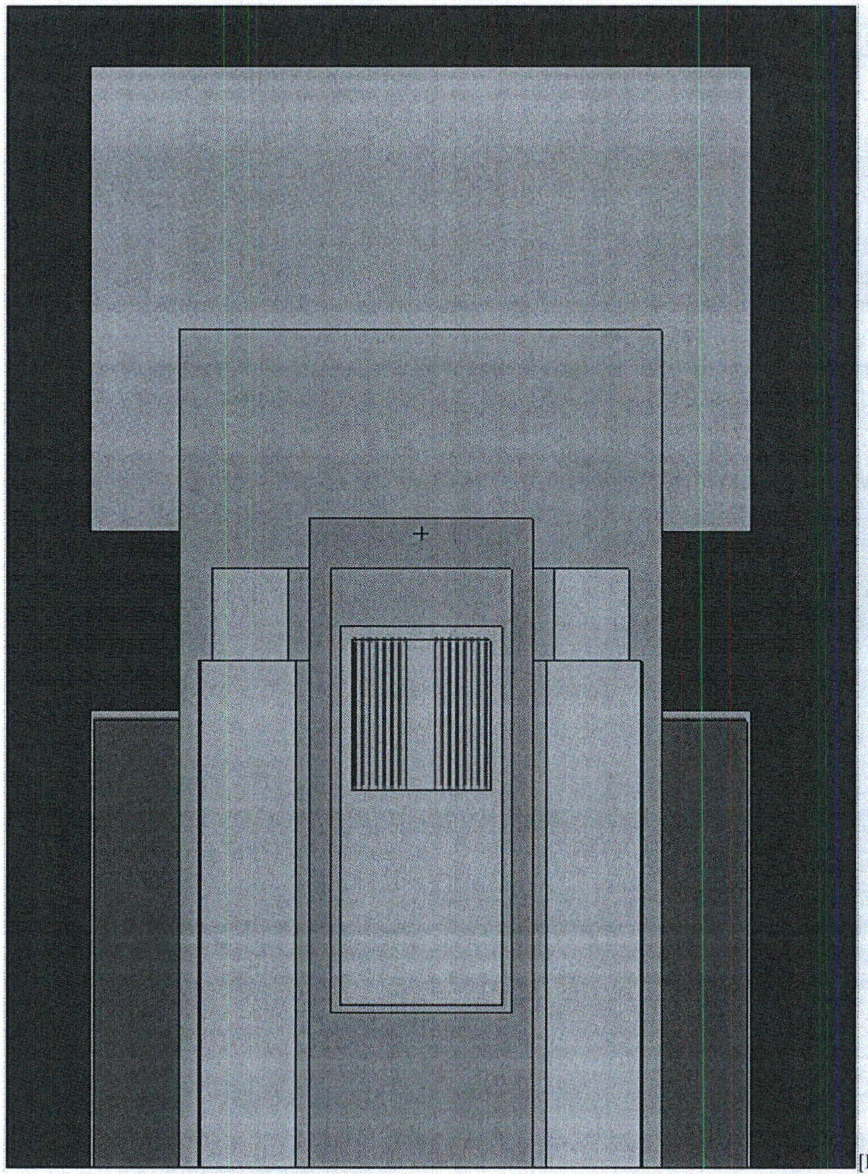


Figure 6.7.5-3 VISED Sketch of LWT Axial View – SLOWPOKE Fuel Core – Normal Conditions – Cask Top Portion



[PH1]

Table 6.7.5-1 SLOWPOKE Fuel Core Design Basis Characteristics

Parameter	Value
Rods per Assembly	298
Initial ^{235}U per Rod, g	2.830
U per Rod, g	2.970
Initial ^{235}U Enrichment, wt. %	95.30
Aluminum in Fuel Matrix per Rod, g	7.714
Clad Material	Aluminum
Weight of Fuel per Rod, g	10.684
Total ^{235}U in Core, g	843.340
Fuel Volume, cm^3	3.071
Fuel Density, g/cm^3	3.478

Table 6.7.5-2 SLOWPOKE Fuel Core Analysis Compositions and Number Densities

Material	U-Al	Al	H_2O	304 Stainless Steel	Lead Shield	Neutron Shield
Density, g/cm3	3.4785	2.7020	0.9982	7.9400	11.3440	0.9400
Atom Density	atoms/b-cm					
^{235}U	2.361E-03					
^{238}U	1.150E-04					
Aluminum	5.602E-02	6.027E-02				
Hydrogen			6.679E-02			5.840E-02
Oxygen			3.340E-02			2.414E-02
Carbon				3.185E-04		1.010E-02
Silicon				1.703E-03		
Phosphorus				6.941E-05		
Chromium				1.747E-02		
Manganese				1.739E-03		
Iron				5.854E-02		
Nickel				7.739E-03		
Lead					3.297E-02	

6.7.5.3 Criticality Calculations

The maximum reactivity configuration is determined by performing a series of studies on a single, fully water-reflected cask (30-cm full density water reflector). Criticality evaluation of the SLOWPOKE fuel core payload is limited to a single cask, with the exception of the low reactivity normal condition array (10 CFR 71.59) calculation included in Section 6.7.5.3.7.

6.7.5.3.1 Normal versus Accident Conditions Cask Configuration

An initial scoping evaluation considers both normal and accident conditions under a preferential flooding scenario (basket fully flooded, cask cavity dry). As shown in Table 6.7.5-3, cask condition has no statistically resolvable effect on reactivity ($\Delta k_{eff}/\sigma < 3$). Accident conditions are applied in the following studies.

6.7.5.3.2 Fuel Rod Arrangement Study

For this study, the effects of re-arranging rods in the fuel rod array are considered. The first configuration moves fuel rods to the outer ring of the SLOWPOKE fuel core. This configuration is denoted as "spread" in the results shown in Table 6.7.5-4. In the second configuration for this evaluation, the center tube and upper and lower grid plates are assumed to be damaged, resulting in the re-arrangement of the fuel rods. Rods from the exterior are moved to the empty center cells. This configuration is denoted as "center filled" in Table 6.7.5-4. The hexagonal fuel rod pitch is retained for this study. The maximum fuel rod pitch is also applied in this study. Additional pitches are examined in Section 6.7.5.3.3.

Both configurations confirm that the lattice is undermoderated. For both configurations there are reactivity tradeoffs between moderation and leakage.

For the spread configuration, reactivity is increased for both nominal and maximum pitch. Additional moderator for fuel rods adjoining the open lattice locations increases reactivity. Additional moderation from increasing the pitch to the maximum possible shows no effects as reactivity increases from additional moderation are being offset by additional leakage.

For the center filled configuration, filling the center cells (7 open equivalent lattice locations) shows no significant change at nominal pitch. Removal of moderator in the center decreases available moderator to adjoining lattice locations while adding fuel to the center, low leakage region increases reactivity. At maximum pitch (near the highest reactivity lattice H/U) the center filled arrangement is significantly more reactive than all other configurations as moderation is increased without an increase in leakage.

Remaining studies will include the maximum reactivity configuration with the center cells filled. An example VISED XY cross-section of the configuration with rods in the outer ring and

nominal fuel rod pitch is shown in Figure 6.7.5-4. An example VISED XY cross-section of the configuration with center cells filled and maximum fuel rod pitch is shown in Figure 6.7.5-5.

6.7.5.3.3 Fuel Rod Pitch Study

A pitch study is performed to determine maximum reactivity fuel rod pitch. The results are shown in Table 6.7.5-5 and plotted in Figure 6.7.5-6. The results demonstrate that peak reactivity is reached prior to the fuel rod array extending beyond the basket cavity radius. As seen in Table 6.7.5-5, after rising rapidly, the slope of reactivity versus rod pitch (H/U ratio) levels out between P4 and Max pitch with no statistically significant variation. Therefore, the lattice is at maximum reactivity pitch within the constraints of the basket. Given the statistically constant reactivity across a range of rod pitches, no basket tolerance study is necessary for this evaluation. The maximum pitch of 1.3417 cm is applied in the remaining criticality calculations.

Previous evaluations of the array containing a center tube/center opening or rods in the outer ring were performed at nominal and max pitch. To confirm that maximum reactivity was not achieved at an intermediate pitch, a pitch study was also run for these configurations. The results are plotted in Figure 6.7.5-6 and demonstrate that the center filled configuration provides the maximum reactivity configuration at the maximum pitch.

6.7.5.3.4 Moderator Density Study (Including Preferential Flooding)

As the pitch study places the rods at maximum reactivity by increasing the H/U ratio, decreases in interior moderator density are expected to reduce system reactivity. This is confirmed by evaluations of variable basket interior moderator densities with dry cask cavity, variable cask cavity moderator densities with wet basket interior, and simultaneous basket and cask cavity density changes. All results are plotted in Figure 6.7.5-7. The preferential flood scenario with a full density basket and dry cask cavity remains the maximum reactivity case.

6.7.5.3.5 Axial Shift Study

The basket and fuel core are shifted towards the cask cavity top to evaluate potential reactivity increases due to reflection changes. The fuel core shift permitted by the model is 0.81 inches (2.0574 cm), and the basket shift permitted by the model is 0.20 inches (0.5080 cm). The small shifts permitted will result in statistically non-resolvable changes in reactivity, as seen in Table 6.7.5-6. Remaining studies retain the shifted down axial location of the basket and fuel core.

6.7.5.3.6 Maximum Reactivity Configuration

Based on the previous analyses, the following conditions are bounding for the maximum reactivity configuration:

- Loss of center tube (fuel rods shifted to center locations)

- Increased fuel rod pitch (loose fuel rods)
- Preferential flooding of basket (dry cask cavity with a flooded basket assembly)

6.7.5.3.7 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. The 10 CFR 71.55 and 10 CFR 71.59 requirements are satisfied and transport of the SLOWPOKE fuel core is acceptable. The results for the configurations stipulated in the CFR requirements are shown in Table 6.7.5-7. The maximum reactivity configuration produced a reactivity of 0.87189, which is under the USL of 0.9171 (USL documented in Section 6.5.5). The CSI for the package is 100.

6.7.5.3.8 Area of Applicability

The SLOWPOKE fuel core is compared in Table 6.7.5-8 to the set of validated experiments used in Section 6.5.5 to establish the USL. The EALCF is within the validated range. The design basis enrichment of the SLOWPOKE fuel core is outside the validated range. There is no trend found for enrichment in the validation that would result in a lower USL.

6.7.5.3.9 Licensed SLOWPOKE Fuel Core Basket versus Modeled Basket

The licensed SLOWPOKE fuel core basket differs from the basket model in previous evaluations as described in Section 6.7.5.2. The modeled basket increases reflection by removing moderator between the fuel and cask lid, which provides 11.25 inches of stainless steel reflection. The increased reflection is expected to increase reactivity. This is confirmed by the results provided in Table 6.7.5-9. For the licensed basket, the space in the lid spacer is modeled as flooded, as drain holes lead to the basket interior. A preferential flood elevation in the actual basket would increase reflection by removing moderator similar to the previously evaluated configuration. Therefore, no additional evaluation is needed to confirm that the analysis is conservative. A VISED slice of the modeled licensed SLOWPOKE fuel core basket is shown in Figure 6.7.5-8.

Figure 6.7.5-4 VISED XY Slice of SLOWPOKE Fuel Core with Spread, Nominal Rod Pitch Configuration

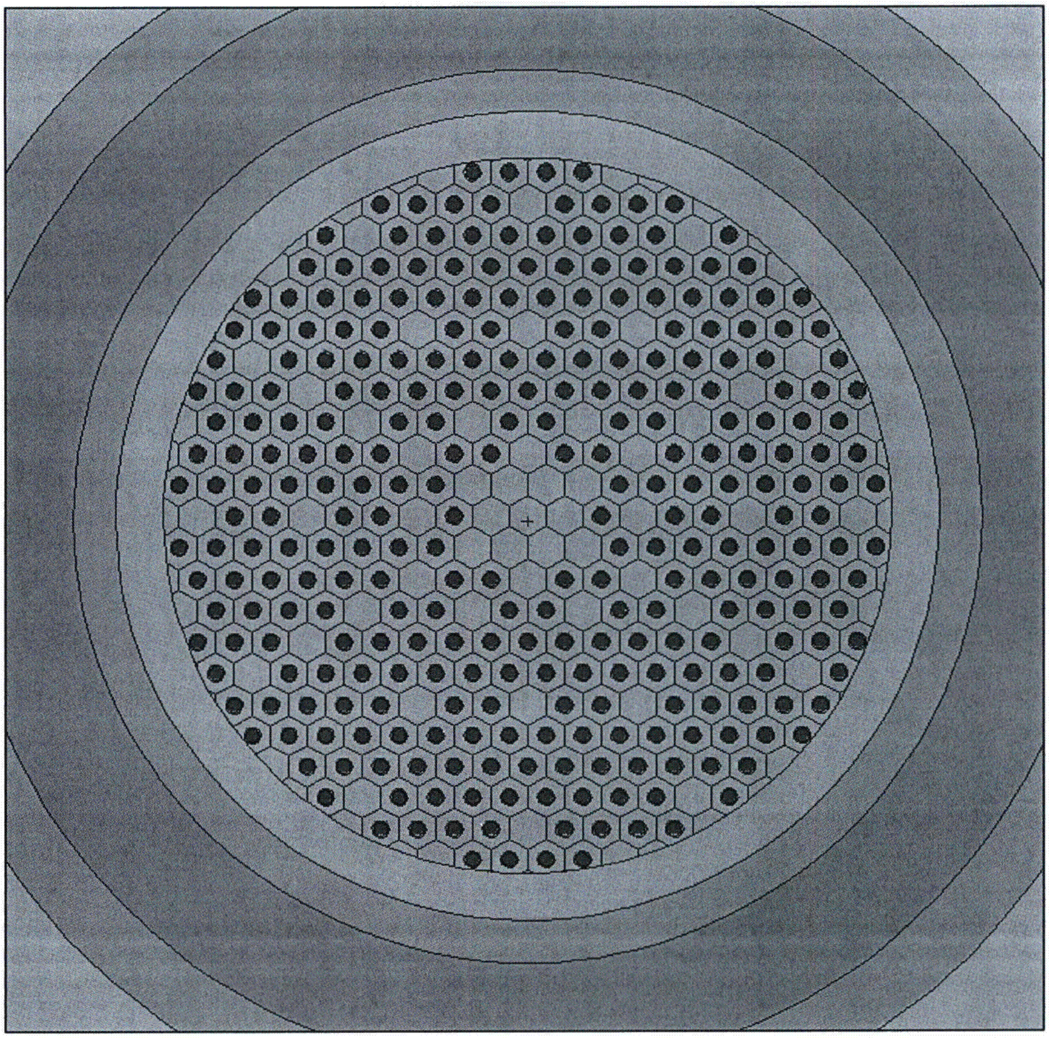


Figure 6.7.5-5 VISED XY Slice of SLOWPOKE Fuel Core with Center Cells Filled, Maximum Rod Pitch Configuration

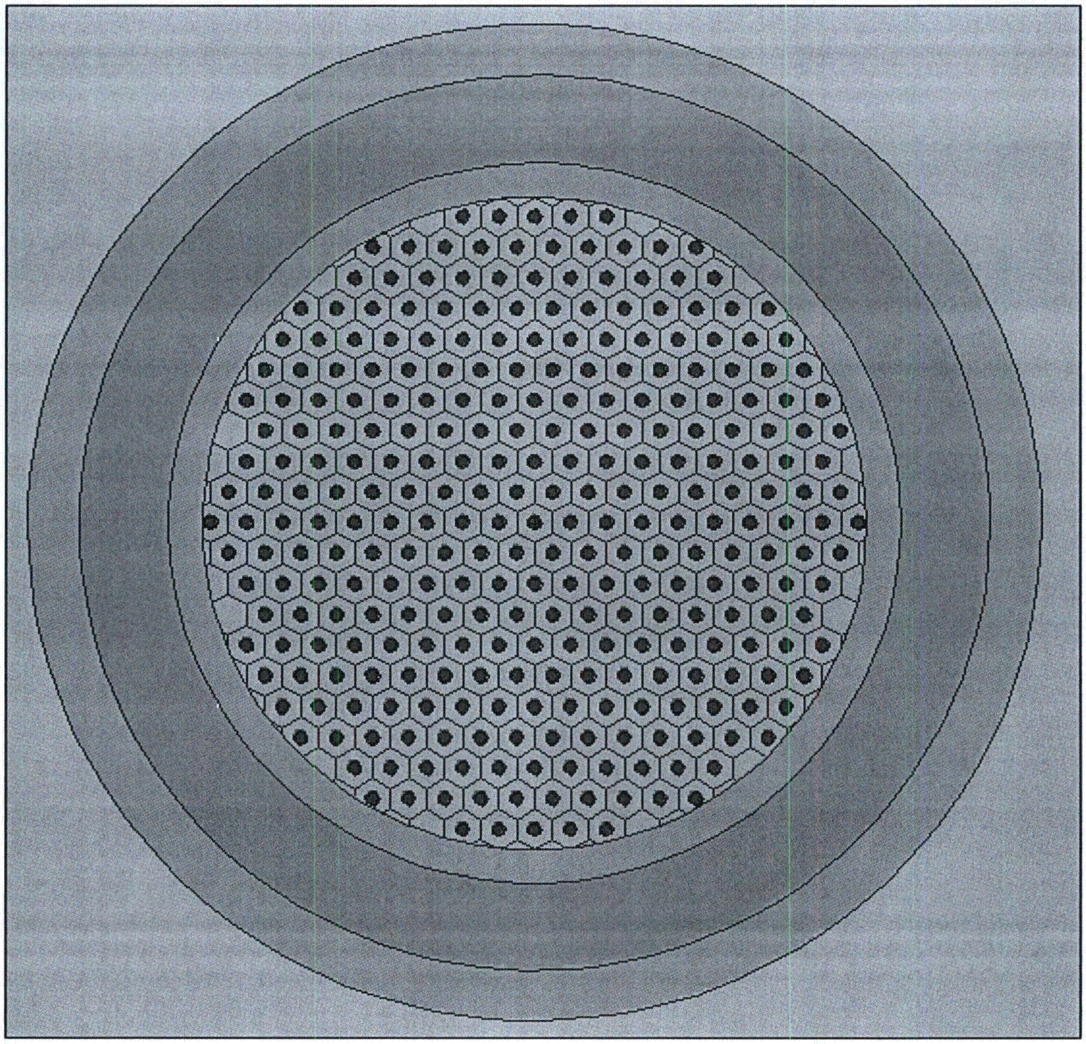


Figure 6.7.5-6 SLOWPOKE Fuel Core, Fuel Rod Pitch Study Results

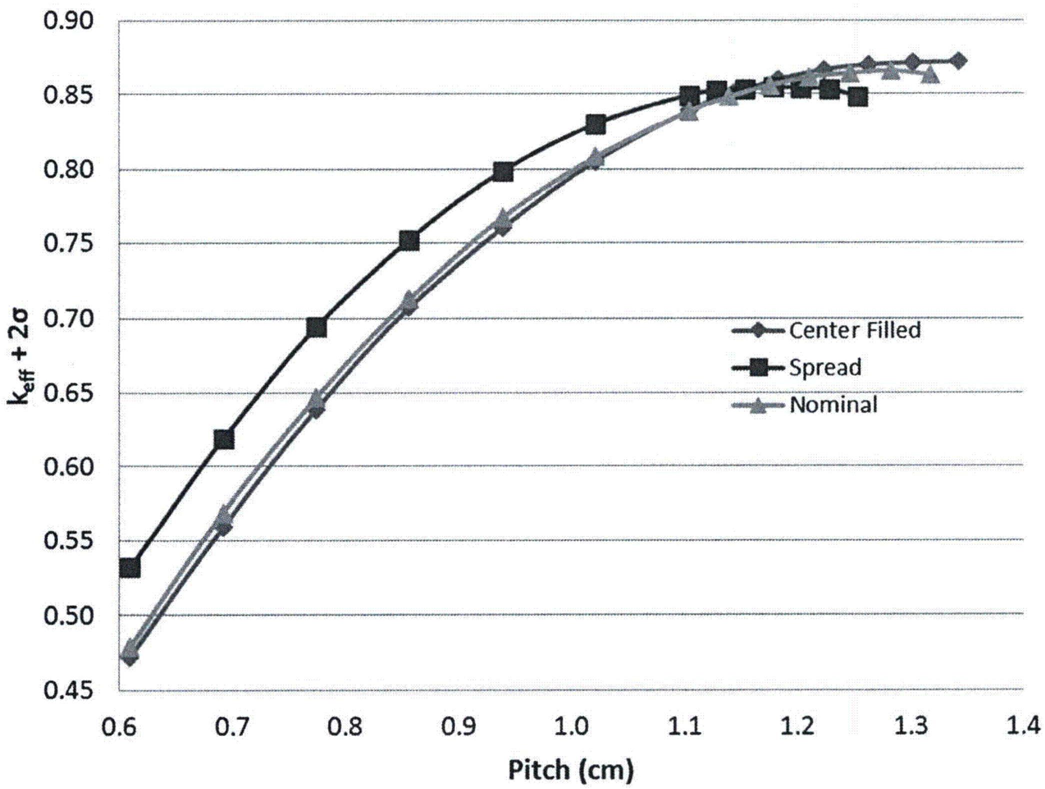


Figure 6.7.5-7 SLOWPOKE Fuel Core Moderator Density Study Results (Percent Full Density Water)

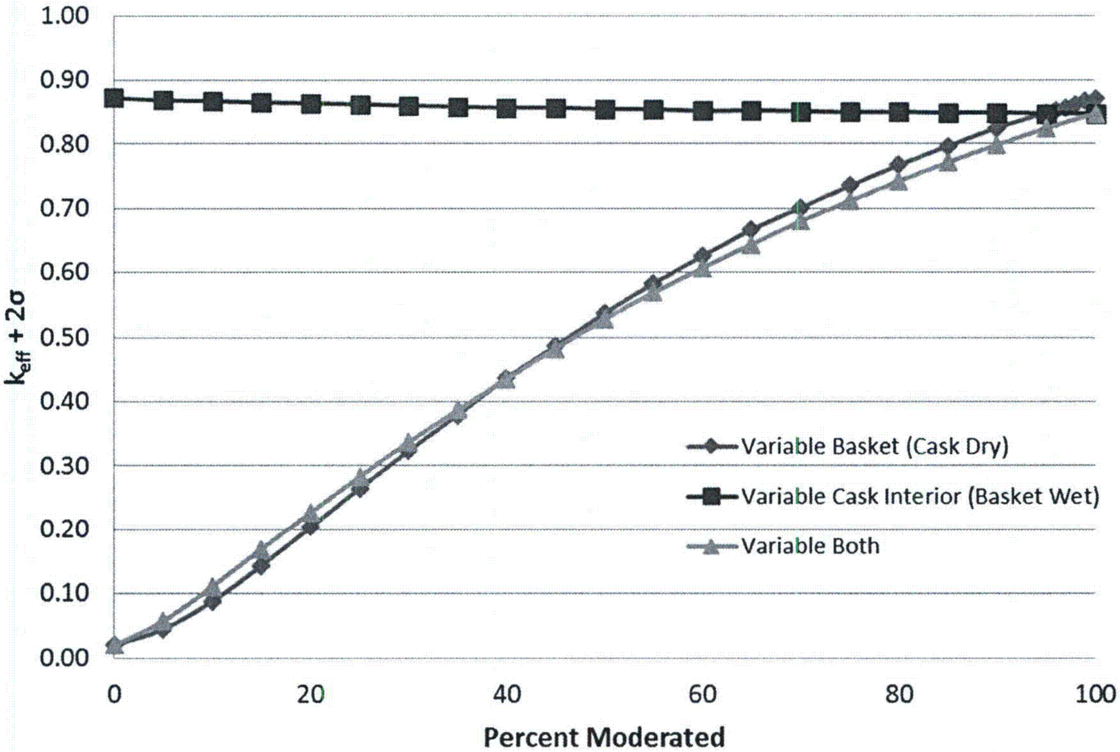


Figure 6.7.5-8 VISED XZ Slice of SLOWPOKE Fuel Core with Licensed Basket Model

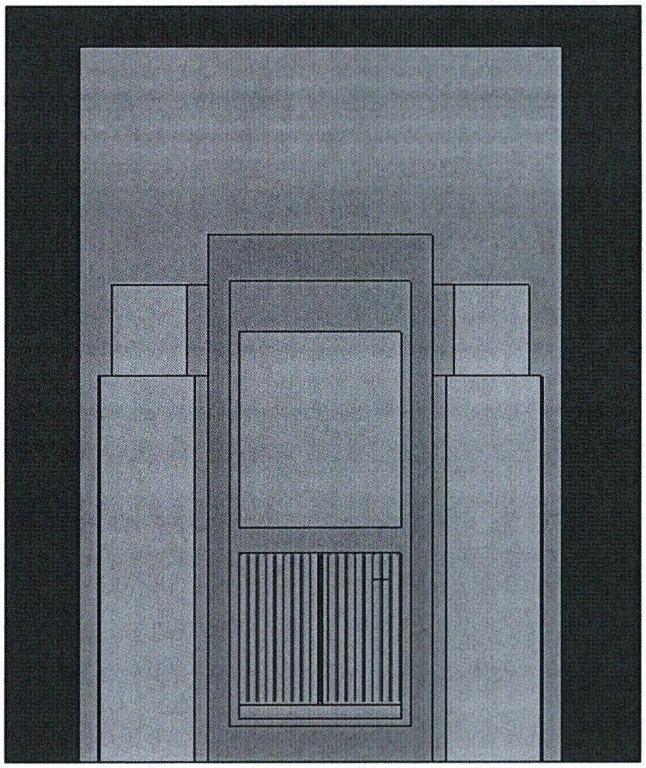


Table 6.7.5-3 Reactivity Results for SLOWPOKE Fuel Core under Normal and Accident Conditions

Cask	k_{eff}	σ	$k_{eff}+2\sigma$	$\Delta k_{eff}/\sigma$
Normal	0.83671	0.00073	0.83817	--
Accident	0.83669	0.00074	0.83817	0.0

Table 6.7.5-4 Reactivity Results for SLOWPOKE Fuel Core Alternate Rod Arrangements

Rod Configuration	Rod Pitch	k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Nominal	Nom	0.83669	0.00074	0.83817	--	--
Nominal	Max	0.86169	0.00074	0.86317	0.02500	23.9
Spread	Nom	0.84776	0.00077	0.84930	0.01107	10.4
Spread	Max	0.84697	0.00072	0.84841	0.01028	10.0
Center Filled	Nom	0.83723	0.00073	0.83869	0.00054	0.5
Center Filled	Max	0.87049	0.00070	0.87189	0.03380	33.2

Table 6.7.5-5 Reactivity Results for SLOWPOKE Fuel Core, Fuel Rod Pitch Study

Description	Pitch (cm)	k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Max	1.3417	0.87049	0.00070	0.87189	--	--
P5	1.3020	0.86970	0.00077	0.87124	-0.0008	-0.8
P4	1.2623	0.86855	0.00073	0.87001	-0.0019	-1.9
P3	1.2227	0.86520	0.00072	0.86664	-0.0053	-5.3
P2	1.1830	0.85829	0.00077	0.85983	-0.0122	-11.7
P1	1.1433	0.84924	0.00073	0.85070	-0.0213	-21.0
Nom	1.1036	0.83723	0.00073	0.83869	-0.0333	-32.9
N1	1.0213	0.80382	0.00075	0.80532	-0.0667	-65.0
N2	0.9390	0.75868	0.00073	0.76014	-0.1118	-110.6
N3	0.8566	0.70568	0.00073	0.70714	-0.1648	-163.0
N4	0.7743	0.63758	0.00074	0.63906	-0.2329	-228.7
N5	0.6919	0.55783	0.00065	0.55913	-0.3127	-327.3
Min	0.6096	0.47131	0.0006	0.47251	-0.3992	-433.0

Table 6.7.5-6 Reactivity Results for SLOWPOKE Fuel Core, Axial Shift Study

Description	k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Bottom Shifted (Nom.)	0.87049	0.00070	0.87189	--	--
Top Shifted	0.86786	0.00070	0.86926	-0.0026	-2.7

Table 6.7.5-7 10 CFR 71.55 and 10 CFR 71.59 Case Matrix for the NAC-LWT with SLOWPOKE Fuel Core

Regulation	Cask Condition	Cask Interior Moderator Conditions	Array Size	Reflection	$k_{eff}+2\sigma$	CSI
71.55	Containment	Pref. Flood	1	30 cm Water	0.82225	--
71.55	Normal	Pref. Flood	1	30 cm Water	0.87110	--
71.55	Accident	Pref. Flood	1	30 cm Water	0.87189	--
71.59 (a.1)	Normal	Dry	Infinite	Cask Surface	0.02308	0
71.59 (a.2)	Accident	Pref. Flood	1	30 cm Water	0.87189	100

Table 6.7.5-8 Area of Applicability for SLOWPOKE Fuel Core Validation

Parameter	Minimum	Maximum	Evaluated
Enrichment (wt.% ^{235}U)	17.0	93.2	95.3
EALCF (eV)	0.0503	1.120	0.0660

Table 6.7.5-9 Comparison of Modeled and Licensed Basket Reactivity for the SLOWPOKE Fuel Core

Basket Description	Fuel Core and Basket Shift	k_{eff}	σ	$k_{eff}+2\sigma$	Δk_{eff}	$\Delta k_{eff}/\sigma$
Modeled	Nom	0.87049	0.00070	0.87189	--	--
Licensed	Nom	0.84503	0.00071	0.84645	-0.0255	-25.5
Licensed	Top	0.84924	0.00071	0.85066	-0.0213	-21.3

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6.6.19 SLOWPOKE Fuel Core MCNP Input

This section contains a sample input file from the evaluation of a SLOWPOKE fuel core in the LWT cask. The input file is shown in Figure 6.6.19-1.

Figure 6.6.19-1 Maximum Reactivity Input for the SLOWPOKE Fuel Core Payload

```
NAC-LWT Cask - Accident Transport Conditions
C SlowPoke Fuel - Core Configuration
C Fuel Rod Cells
1  1  -3.4785  -1  +3  -4      u=3  $ Fuel Meat
2  2  -2.7020  -2  +1  +3  -4     u=3  $ Clad
3  2  -2.7020  -5 : -6           u=3  $ Lower Stud
4  2  -2.7020  -7          u=3  $ Upper Stud
5  3  -0.9982  #1 #2 #3 #4            u=3  $ Outside Rod
C Fuel Core Array Cell
10  3  -0.9982  -10  u=2  lat=2  fill=-15:15 -15:15 0:0    $ Core Array
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2
       2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 3 3 3 3 3 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
       2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
C Basket Cells
20  3  -0.9982  -20  -21  fill=2  ( 0.0000 0.0000 33.6550 )    u=1  $ Core
21  3  -0.9982  -21  +20         u=1  $ Basket Cavity
22  5  -7.9400  -22  +21         u=1  $ Basket
23  4  -0.0001  +22          u=1  $ Outside Basket
C Cells - LWT Cask Accident Conditions
30  6  -11.344  -33            $ BotPb
31  4  -0.0001  -32  fill=1  ( 0.0000 0.0000 377.1900 )        $ Cavity
32  5  -7.9400  -31  +33          $ Bottom
33  5  -7.9400  -30  +31  +35  +38  +32     $ OuterShell
34  5  -7.9400  -34  +37  +32          $ InnerShellTaper
35  5  -7.9400  -36  +32          $ InnerShell
36  6  -11.344  -37  +36          $ Lead
37  6  -11.344  -35  +34  +37         $ LeadTaper
38  0           -38  +37          $ LeadGap
39  8  -0.9982  +30  -39          $ Gap To Reflector
40  0           39           $ Outside

C Fuel Rod Surfaces
1  CZ  0.2108             $ Fuel Meat
2  CZ  0.2616             $ Clad
3  PZ  0.4572             $ Lower Meat Elevation
4  PZ  22.4536             $ Upper Meat Elevation
5  RCC  0.0  0.0  0.3302  0.0  0.0  0.1270  0.3048  $ Lower Stud Rim
6  RCC  0.0  0.0  0.0  0.0  0.0  0.3302  0.1930  $ Lower Stud Cap
7  RCC  0.0  0.0  22.4536  0.0  0.0  0.3810  0.3048  $ Upper Stud
C Fuel Core Lattice Surface
10  RHP  0.0  0.0  0.0010  0.0  0.0  22.8326
       0.6709  0.0  0.0       $ Lattice Cell
C Basket Surfaces
20  RCC  0.0000  0.0000  33.6570  0.0000  0.0000  22.8306  12.6027  $ Core Window
21  RCC  0.0000  0.0000  1.2700  0.0000  0.0000  57.2770  12.3825  $ Basket Cavity
22  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  67.4370  13.6525  $ Basket
C Surfaces - LWT Cask Accident Conditions
30  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  507.3650  36.5189  $ Lwt
31  RCC  0.0000  0.0000  -26.6700  0.0000  0.0000  26.6700  36.5189  $ Bottom
32  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  452.1200  16.9863  $ Cavity
33  RCC  0.0000  0.0000  -17.7800  0.0000  0.0000  7.6200  26.3525  $ Bottom gamma shield
34  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  20.1740  $ Lead id - taper
35  RCC  0.0000  0.0000  0.0000  0.0000  0.0000  444.5000  31.5976  $ Lead od - taper
36  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  18.9103  $ Lead id
37  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.3271  $ Lead od
38  RCC  0.0000  0.0000  13.8176  0.0000  0.0000  416.8648  33.4645  $ Lead gap
39  RCC  0.0000  0.0000  -56.6700  0.0000  0.0000  567.3650  66.5189  $ Container


C
C Materials List
C
C U-Al (Fuel Meat)
m1  92235.69c  -2.6489E-01
       92238.69c  -1.3064E-02
       13027.62c  -7.2204E-01
```

```
C Aluminum / Clad
m2  13027.62c  -1.0
C Canister Water
m3  1001.62c  6.6667E-01  8016.62c  3.3333E-01
mt3  lwtr.60t
C Cask Cavity Water
m4  1001.62c  6.6667E-01  8016.62c  3.3333E-01
mt4  lwtr.60t
C Stainless Steel 304
m5  6000.66c  -8.0000E-04  14000.60c  -1.0000E-02  15031.66c  -4.5000E-04
       24000.50c  -1.9000E-01  25055.62c  -2.0000E-02  26000.55c  -6.8375E-01
       28000.50c  -9.5000E-02
C Lead
m6  82000.50c  -1.0
C Aluminum Honeycomb Impact Limiter
m7  13027.62c  -1.0
C Water/Glycol - Cask Neutron Shield
m9  1001.62c  -1.03171E-01  6000.66c  -2.14392E-01    8016.62c  -6.82437E-01
C Cask Exterior (Water at Various Densities)
m8  1001.62c  6.6667E-01  8016.62c  3.3333E-01
mt8  lwtr.60t
C
C Cell Importances
imp:n 1 19r 0
c
c Criticality Controls
c
kcode  2000  1.00  300  1000
c
c Source Distribution for Initial Generation
ksrc  -2.6834  0.0000  422.2623
       0.0000  2.3239  422.2623
       0.0000  -2.3239  422.2623
       2.6834  0.0000  422.2623
C Print Control
print
C Random Number Generator
rand  GEN=2  SEED=461360
```

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7.1.18 Procedure for the Dry Loading of AECL SLOWPOKE Fuel Core into the NAC-LWT Cask

This section describes the procedures for loading the NAC-LWT cask with a SLOWPOKE fuel core. One SLOWPOKE fuel core assembly can be loaded into a SLOWPOKE fuel core basket. The SLOWPOKE basket assembly consists of a basket weldment with one cylindrical opening and a bolted lid. The SLOWPOKE basket assembly is designed to sit atop a stack of five MTR-42 baskets (one base basket and four intermediate baskets) to position the SLOWPOKE basket assembly at the top of the cask cavity.

The SLOWPOKE fuel core is required to be intact and placed into a SLOWPOKE basket assembly. Loose SLOWPOKE fuel rods are addressed in Section 7.1.15.

The maximum decay heat load of a loaded SLOWPOKE basket shall be ≤ 45 Watts.

The maximum content weight per basket shall be ≤ 15 lbs.

The NAC-LWT cask will be loaded dry, utilizing a transfer cask to place the loaded SLOWPOKE basket into the NAC-LWT cask cavity. The five empty MTR-42 baskets will be preloaded into the cask cavity prior to loading the SLOWPOKE fuel basket.

The procedure for dry-loading and preparation for transport of the NAC-LWT with a SLOWPOKE fuel core 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(s) of Conformance 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.

   Note: Lifting loaded containers from the top corner fitting with forces applied other than vertically is not permitted; use of an approved container lifting spreader, frame or bottom lift container slings is 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 rear 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.

Note: Verify that the package nameplate displays the correct package identification number in accordance with the CoC.

4. Remove the roof from the ISO container and cross members, if installed.
5. Remove the top and bottom impact limiters, and remove 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 alternate vent and drain port covers. Store the alternate port covers to protect the seal surfaces. Visually inspect the vent valve quick-disconnect nipples and replace if necessary. Prior to installation, inspect the Viton® O-ring seals on the alternate port covers, and if any O-ring shows any damage, replace it.
8. Install the cask lifting yoke with the guides removed 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, at the intended loading station. Connect the base plate to the cask's attachment points using chains and take up slack with the tensioners. Disengage the lifting yoke.
10. Visually inspect the neutron shield tank fill, drain, and level inspection plugs for signs of neutron shield fluid leakage. If leakage is detected or suspected, verify shield tank fluid level and correct, as required.
11. Loosen and remove all closure lid bolts. Prior to installation, inspect the lid bolts and replace any that are damaged.
12. Attach lid lifting slings, or equivalent lid removal fixture, 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 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.
13. 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, including the lid sealing surface. Install, or verify the presence of the drain tube and drain alignment ring.
14. Verify the proper installation of, or install, the five empty MTR-42 baskets.

15. Install the required dry transfer system components on the top of the cask.
16. Position the Dry Transfer System (DTS) components for fuel loading, as appropriate.
17. Identify the SLOWPOKE fuel core to be loaded, and verify that it complies with the authorized content, heat load and quantity conditions of the CoC.
18. Place the SLOWPOKE basket weldment in the Intermediate Transfer System (ITS) inner shield.
19. Move the ITS inner shield into position near the reactor for the transfer of the loaded SLOWPOKE fuel core.
20. Lift the SLOWPOKE fuel core out of the reactor using site supplied tooling and equipment and place it into the SLOWPOKE fuel core basket weldment in the ITS inner shield. Disengage the fuel core handling tooling.
21. Install the SLOWPOKE basket lid assembly and torque the lid bolts to 60 +/- 10 in-lbs.
22. Install the inner shield lid.
23. Move the ITS inner shield assembly containing the loaded SLOWPOKE basket assembly to the pre-staged transfer system location.
24. Lift the inner shield assembly containing the loaded SLOWPOKE basket assembly and place it through the ITS shield assembly adapter and into the outer shield of the ITS.
25. Disengage the inner shield lid. Lift and remove the inner shield lid through the shield assembly adapter and close the shield assembly adapter gate.
26. Place the DTS transfer cask onto the ITS shield assembly adapter.
27. Open the DTS transfer cask gate.
28. Open the ITS shield assembly adapter gate.
29. Lower the transfer cask grapple into the ITS and engage the SLOWPOKE basket assembly.
30. Retract grapple and loaded SLOWPOKE basket assembly into the transfer cask.
31. Close the DTS transfer cask shield gate.
32. Lift the DTS transfer cask and place it on the cask adapter assembly positioned on top of the NAC-LWT cask.
33. Open the cask adapter shield gate.
34. Open the DTS transfer shield cask gate and lower the loaded SLOWPOKE basket assembly into the NAC-LWT cask cavity.
35. Disengage grapple and retract back into the transfer cask.

    Note: Grapple release can be verified by checking cable for tension.
36. Verify grapple is fully retracted.

    Note: Indication will be physical indicator attached to cable.
37. Close cask adapter shield gate.

38. Remove the transfer cask from the dry transfer system adapter.
39. Using the dry transfer system adapter components, install temporary shield plug. Remove shield ring/plug assembly through the dry transfer system adapter.
40. Install the closure lid onto the cask using the dry transfer system. Visually verify that the lid is properly seated.
41. Install lid bolts hand tight.
42. Remove dry transfer system components from the top of the cask.
43. Tighten all 12 closure lid bolts to 260 ± 20 ft-lbs in three passes using the torque sequence indicated on the closure lid.
44. Connect a gas supply line to the vent valve and the drain line to the drain valve.
45. 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.

    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 40.
46. Connect the Vacuum Drying System (VDS) to the cask vent valve and evacuate the cask cavity by vacuum pump to less than or equal to 10 torr (13 mbar) and continue vacuum pumping for a minimum of 15 minutes.
47. At the end of the evacuation period, isolate the cask cavity from the vacuum pump and 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), resume vacuum drying until the dryness verification results are satisfactory.
48. Backfill the cask cavity with helium to 0 psig (1 atmosphere, absolute), +1, -0 psi and disconnect the VDS from the vent valve.
49. Perform a helium leakage test of the closure lid containment O-ring using a Helium Mass Spectrometer Leak Detector (MSLD) in accordance with the requirements of SAR Section 8.1.3.1.
50. Install the vent and drain alternate port covers and torque the bolts to 100 ±10 inch-pounds.
51. If an alternate port cover containment O-ring seal was replaced, perform a helium leakage test on the affected port cover using a He MSLD in accordance with the requirements of SAR Section 8.1.3.2.2.
52. If the alternate port cover containment seal was inspected and accepted for reuse, perform an air pressure drop leakage test on the affected port cover as follows.
    a. Install a pressure test fixture to the port cover test port, including a calibrated pressure gauge with a minimum sensitivity of 0.25 psi.

b. Pressurize the port cover seal annulus to 15 psig, +1, -0 psi.

c. Isolate the gas supply and observe the pressure gauge for a minimum of five minutes.

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.

53. Survey the cask surface for removable contamination 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.

54. Using the cask lifting yoke with guides removed, 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.

55. Disengage the cask lifting yoke from the cask lifting trunnions and set it aside.

56. Install and attach the cask tie-down strap. Install the cask top and bottom impact limiters.

57. Install a TID to one of the top impact limiter ball lock pins. Record TID identification number on the loading/shipping documentation.

58. Install roof cross-members, if used and replace ISO container roof.

59. Complete a Health Physics survey on the external surfaces of the package and record the results. Complete dose rate measurements at the package surface, at 1 meter from the package 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 package is the transport index (TI). Ensure compliance with 10 CFR 71.87(i) and observe the following criteria.

a. If the dose rate is less than 2 mSv/h (200 mrem/hr) at all accessible points on the external surface of the package, and the TI is less than 10, the package meets the requirements of 10 CFR 71.47 (a).

b. 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.

c. If the dose rate is > 10 mSv/h (1000 mrem/hr) at any point on the external surface of the package, the package exceeds the limits of 10 CFR 71.47 and cannot be shipped.

60. 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.

61. Complete the shipping documents, carrier instructions (as required), and apply appropriate placards and labels.