June 2015

NAC-LWT

Legal Weight Truck Cask System

LWT-15C SLOWPOKE Fuel Core RAI Response Submittal

NON-PROPRIETARY VERSION

Docket No. 71-9225

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
Phone 770-447-1144, Fax 770-447-1797, www.nacintl.com

ATTACHMENT 1

NAC International Affidavit Pursuant to 10 CFR 2.390

NAC INTERNATIONAL
AFFIDAVIT PURSUANT TO 10 CFR 2.390

Mr. Kent Cole (Affiant), President and CEO, NAC International, hereinafter referred to as NAC, at 3930 East Jones Bridge Road, Norcross, Georgia 30092, being duly sworn, deposes and says that:

1. Affiant has reviewed the information described in Item 2 and is personally familiar with the trade secrets and privileged information contained therein, and is authorized to request its withholding.

2. The information to be withheld includes the following NAC Proprietary Information that is being provided to support the technical review of NAC's Request for a Certificate of Compliance (CoC) (No. 9225) for the NAC-LWT Package.

   - NAC Proprietary Calculation Data Disks
     - Shielding Calculation Data Input/Output Files, Disk 1 of 1
   - NAC-LWT SAR, Revision 15C, - Proprietary Version

   NAC is the owner of the information contained in the above documents. Thus, all of the above identified information is considered NAC Proprietary Information.

3. NAC makes this application for withholding of proprietary information based upon the exemption from disclosure set forth in: the Freedom of Information Act ("FOIA"); 5 USC Sec. 552(b)(4) and the Trade Secrets Act; 18 USC Sec. 1905; and NRC Regulations 10 CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial financial information obtained from a person, and privileged or confidential" (Exemption 4). The information for which exemption from disclosure is herein sought is all "confidential commercial information," and some portions may also qualify under the narrower definition of "trade secret," within the meanings assigned to those terms for purposes of FOIA Exemption 4.

4. Examples of categories of information that fit into the definition of proprietary information are:

   a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by competitors of NAC, without license from NAC, constitutes a competitive economic advantage over other companies.

   b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product.

   c. Information that reveals cost or price information, production capacities, budget levels or commercial strategies of NAC, its customers, or its suppliers.

d. Information that reveals aspects of past, present or future NAC customer-funded development plans and programs of potential commercial value to NAC.

e. Information that discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information that is sought to be withheld is considered to be proprietary for the reasons set forth in Items 4.a, 4.b, and 4.d.

5. The information to be withheld is being transmitted to the NRC in confidence.

6. The information sought to be withheld, including that compiled from many sources, is of a sort customarily held in confidence by NAC, and is, in fact, so held. This information has, to the best of my knowledge and belief, consistently been held in confidence by NAC. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements, which provide for maintenance of the information in confidence. Its initial designation as proprietary information and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in Items 7 and 8 following.

7. Initial approval of proprietary treatment of a document/information is made by the Vice President, Engineering, the Project Manager, the Licensing Specialist, or the Director, Licensing – the persons most likely to know the value and sensitivity of the information in relation to industry knowledge. Access to proprietary documents within NAC is limited via "controlled distribution" to individuals on a "need to know" basis. The procedure for external release of NAC proprietary documents typically requires the approval of the Project Manager based on a review of the documents for technical content, competitive effect and accuracy of the proprietary designation. Disclosures of proprietary documents outside of NAC are limited to regulatory agencies, customers and potential customers and their agents, suppliers, licensees and contractors with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

8. NAC has invested a significant amount of time and money in the research, development, engineering and analytical costs to develop the information that is sought to be withheld as proprietary. This information is considered to be proprietary because it contains detailed descriptions of analytical approaches, methodologies, technical data and/or evaluation results not available elsewhere. The precise value of the expertise required to develop the proprietary information is difficult to quantify, but it is clearly substantial.

9. Public disclosure of the information to be withheld is likely to cause substantial harm to the competitive position of NAC, as the owner of the information, and reduce or eliminate the availability of profit-making opportunities. The proprietary information

is part of NAC's comprehensive spent fuel storage and transport technology base, and its commercial value extends beyond the original development cost to include the development of the expertise to determine and apply the appropriate evaluation process. The value of this proprietary information and the competitive advantage that it provides to NAC would be lost if the information were disclosed to the public. Making such information available to other parties, including competitors, without their having to make similar investments of time, labor and money would provide competitors with an unfair advantage and deprive NAC of the opportunity to seek an adequate return on its large investment.

STATE OF GEORGIA, COUNTY OF GWINNETT

Mr. Kent Cole, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated herein are true and correct to the best of his knowledge, information and belief.

Executed at Norcross, Georgia, this 5th day of June, 2015.

Kent Cole

Kent Cole
President and CEO
NAC International

Subscribed and sworn before me this 5th day of June, 2015.

Deborah R Chapman
Notary Public

DEBORAH R CHAPMAN
MY COMMISSION EXPIRES
NOTARY PUBLIC
02-27-2017
GWINNETT COUNTY, GEORGIA

Enclosure 1 – RAI
Responses and Supporting
Documents

Enclosure 1

No. 71-9225 for NAC-LWT Cask

RAI Responses and Supporting Documents

NAC-LWT SAR, Revision LWT-15C

Enclosure 1 Sections:

1.1 RAI Responses

1.2 Supporting Documents:

a. Shielding Calculation Data Input/Output Files, Disk 1 of 1

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9225, REVISION FOR THE MODEL NO. NAC-LWT PACKAGE TO INCORPORATE SLOWPOKE FUEL CORE

(TAC NO. L24708 DOCKET NO. 71-9225)

JUNE 2015

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NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

3.1 Justify the use of steady state component temperatures calculated for normal conditions of transport when calculating the maximum component temperatures for hypothetical accident conditions.

It is unclear to the NRC staff why the use of normal conditions of transport component temperatures calculated from a shell temperature of 214°F and subsequent change in component temperatures (ΔTs) reported in the unlabeled table in Section 3.5.3.19, page 3.5-16, is appropriate when calculating component temperatures with a hypothetical accident conditions shell temperature of 334°F. Provide additional text and/or calculations that support the information presented in the safety analysis report.

This information is necessary to determine compliance with 10 CFR 71.73.

NAC International Response to Thermal Evaluation RAI 3.1:

The duration of the analyzed fire accident is short relative to the thermal mass of the cask. The basket and contents will increase in temperature as a result of the fire's heat flux into the cask surface, but the inner shell will not increase in temperature due to the thermal inertia and limited conductance of the basket and contents. If the thermal mass of the contents was assumed to be zero, a change in shell temperature could be achieved; however, the cask has significant thermal mass. Thus, applying the change in temperature of the inner shell directly to the maximum initial temperature of the contents is a conservative method of determining maximum contents temperature.

NAC INTERNATIONAL RESPONSE

TO

REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

5.1 Clarify the text in Section 5.3.23.1 that explains why the parameters used in the TRITON analyses produce bounding source terms.

The discussion in Section 5.3.23.1 of the application is not clear. It states that: "*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.*" Section 1.2.3.15 of the application states: "Key physical, radiation protection and thermal characteristics of the SLOWPOKE fuel core are listed in Table 1.2-17." When comparing Table 5.2.23-2 with that of Table 1.2-17, the values are the same. Inspection of the TRITON input file in Figure 5.3.23-2 appears to show that these same values were used in the TRITON input file to generate the source term. The discussion at the beginning of Section 5.3.23.1 seems to indicate that parameters used to generate the source term are changed to produce a more bounding source term, however this does not appear to be the case. Explain what is meant in the statement "key parameters differing between the input and analysis" means.

This information is needed for the staff to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

NAC International Response to Shielding Evaluation RAI 5.1:

The first paragraph of Section 5.3.23.1 states that SLOWPOKE rod characteristics are listed in Table 5.3.23-1. The remainder of the paragraph then describes how the parameters input into the TRITON depletion, those in Table 5.3.23-2, are conservative when compared to the SLOWPOKE rod characteristics presented in Table 5.3.23-1. Table 1.2-17 contains the data applied in the licensing calculation, as these parameters were shown to be safe for transport. As such, values presented in Table 1.2-17 are the same as those presented in Table 5.3.23-2, where applicable.

To clarify the various sets of data, the text in Sections 5.23.1 and 1.2.3.15 was revised. In the revised Section 5.3.23.1, text data is referred to as reference fuel design characteristics for data obtained from reference information, and as depletion and shielding evaluation code input characteristics for the data applied in the TRITON and shielding models. Revised the language in Section 1.2.3.15 to clarify that the listed characteristics is not reference data.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

5.2 Clarify the lead slump assumptions for the NAC-LWT SLOWPOKE Core model.

Section 5.3.23.2 of the application states: *"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."* Revise the application to clarify the extent of axial lead slump after an end drop and radial lead slump after a side drop. Section 5.3.23.2 appears to state that the 0.1374 cm gap includes axial lead slump after an end drop. Either ensure consistency with the lead slump calculations in Chapter 2 or justify that the amount of lead slump is appropriate.

This information is needed to determine compliance with regulations in 10 CFR 71.51(a)(2).

NAC International Response to Shielding Evaluation RAI 5.2:

Section 5.3.23.2 is revised to clarify that the radial and axial lead slumps are based on the 0.1374 cm radial lead gap for the normal condition and that lead void/slump was calculated separately for the axial and radial cask drop conditions. The accident shielding model applied the drop-generated voids simultaneously on the radial and top axial region of the cask, therefore accounting for all drop conditions in one run set. Included in the text revision are the sizes of the slump-generated voids being applied in the MCNP model.

Note that the end drop slump void being applied in Section 5.3.23 is significantly larger than that reported in Chapter 2. Chapter 2, Section 2.10.5, documents an end drop produced slump of 0.33 inches, Section 2.7.1.4 lists a more conservative 1.63 inch slump, and the value applied in the Section 5.3.23.2 shielding model is 2.71 inches. Section 2.6.11, which documents fabrication procedures of the radial lead shield, indicates that heaters are applied to the cask shells during lead pour to assure no significant radial void exists in the cask under normal conditions. The values being applied in void calculations of Chapter 2 and Section 5.3.23 conservatively assume that a lead gap exists.

NAC INTERNATIONAL RESPONSE

TO
REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

5.3 Discuss the axial burnup profile of the SLOWPOKE Core.

Although Section 5.3.23.1 of the application discusses how the most conservative radial location was chosen for the SLOWPOKE Core fuel model, it does not include a discussion on the axial burnup profile. Discuss axial burnup and axial peaking of the SLOWPOKE core and how this was accounted for in the shielding model of the SLOWPOKE core fuel model.

This information is needed to determine compliance with dose rate regulations in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2).

<u>NAC International Response to Shielding Evaluation RAI 5.3:</u>

An axial profile of the payload will not significantly affect the safety conclusion drawn in Chapter 5. A paragraph is added to Section 5.3.23.1 to discuss a potential profile and hypothetical effects of a profile.

NAC INTERNATIONAL RESPONSE

TO

REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

5.4 Provide MCNP output files and weight windows input files.

The staff reviews output files from the dose rate calculation to ensure such things as proper convergence and that the calculated radiation levels agree with those reported in the application. The staff cannot run the submitted input files in Figures 5.2.23-7 and 5.2.23-11 of the application without the weight windows input file (wwinp). In addition, the staff needs to verify that the package dimensions and shielding features are modeled appropriately. The NAC-LWT SLOWPOKE core MCNP input file is complex and staff would prefer to use MCNP visualization tools to reduce review time rather than deconstruct the input file. Provide (1) the weight windows input file(s) and (2) associated output files for the input files in Figures 5.2.23-7 and 5.2.23-11.

This information is needed to verify compliance with 10 CFR 71.47(b) and 10 CFR 71.51(a)(2).

NAC International Response to Shielding Evaluation RAI 5.4:

The requested weight window input and output files are included on the proprietary data disk contained in this submittal. Also included on the disk are the text version of the input files, the weight window output files, and the tally summary file generated by the cases.

Input files in the SAR are intended to provide a cross-section of information. As such, the normal condition radial fuel gamma and the accident top fuel neutron case is included in the SAR. To assure that the reviewer has all information generated, also included on the data disk is the full set of cases, normal and accident condition for radial, top, and bottom cases for fuel gamma, fuel neutron and n-gamma runs. Files are divided into directories based on normal or accident condition, direction of radiation transport, and radiation type.

NAC Supporting Documents are Withheld
in their Entirety Per 10 CFR .390

Enclosure 2

No. 71-9225 for NAC-LWT Cask

List of SAR Changes

NAC-LWT SAR, Revision LWT-15C

SLOWPOKE Fuel Core Amendment

RAI Response Package

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

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 1.2-20, modified the last sentence of the second paragraph of Section 1.2.3.15, "SLOWPOKE Fuel Core."

Chapter 2

- No changes.

Chapter 3

- No changes

Chapter 4

- No changes

Chapter 5

- Page 5.3.20-2, made editorial changes to the first two sentences of the last paragraph in Section 5.3.20.1.
- Pages 5.3.23-1 thru 5.3.23-2, modified Section 5.3.23.1, "SLOWPOKE Core Source Term," throughout.
- Pages 5.3.23-3, text flow changes.
- Page 5.3.23-4, modified the first full paragraph on the page in Section 5.3.23.2, "SLOWPOKE Core Shielding Model."
- Page 5.3.23-5, text flow changes.

Chapter 6

- No changes.

Chapter 6 Appendices

- No changes.

List of SAR Changes, NAC-LWT SAR, Revision LWT-15C (continued)

Chapter 7

- No changes.

Chapter 8

- No changes.

Chapter 9

- No changes.

Enclosure 3

No. 71-9225 for NAC-LWT Cask

Proposed Changes for Revision 63 of Certificate of Compliance

NAC-LWT SAR, Revision LWT-15C

SLOWPOKE Fuel Core Amendment

RAI Response Package

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 32:

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 32:

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 31 of 32:

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 32 of 32:

21. Revision 63 of this certificate may be used until February 28, 2015.

22. Expiration Date: April 30, 2020.

Enclosure 4

No. 71-9225 for NAC-LWT Cask

LOEP and SAR Page Changes

NAC-LWT SAR, Revision LWT-15C

SLOWPOKE Fuel Core Amendment

RAI Response Package

June 2015

Revision LWT-15C

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 EFFECTIVE PAGES (Continued)

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, i.e., parameters documented in the analytical chapters to be safely transported, are listed in Table 1.2-17.

June 2015

Revision LWT-15C

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

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5.3.20 SLOWPOKE Fuel Configuration

Results of a shielding analysis for up to 800 fuel rods in the LWT cask are presented in this section. Maximum dose rates are calculated to demonstrate that dose rate limits of 10 CFR 71.47 are not exceeded.

Dose rates are calculated using the MCNP (MCNP5, Version 1.30) three-dimensional transport code. Source terms are calculated using the TRITON/NEWT module of the SCALE package (SCALE 6.1). Cross section tables used in the MCNP analysis are the default provided in the MCNP5 1.30 distribution and draw on mcplib04 for gamma analysis and isotope dependent data from actia, rmccs, or t16_2003 data for neutron evaluations.

5.3.20.1 SLOWPOKE Fuel Source Term

Source terms are calculated to bound the irradiation history of the SLOWPOKE fuel rods. Fuel rod characteristics are summarized in Table 5.3.20-1. Inputs for irradiation and material parameters required by TRITON are given in Table 5.3.20-2. Key parameters differing between the input and analysis are reduced enrichment, increased fuel mass, and increased irradiation time. All parameters revised to produce bounding source terms. 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. Overall, the conservative irradiation time and fuel core power depletion resulted in a core average ^{235}U depletion of 4.5% versus ~2% average reported for the cores to be transported.

TRITON input is shown in Figure 5.3.20-3, with the resulting TRITON material model shown in Figure 5.3.20-2. Neutron and gamma source terms for a cool time of 14 years from discharge are

presented in Table 5.3.20-3 and Table 5.3.20-4, respectively. The calculated heat load at this cool time is 0.0027 W/rod or 2.17 W/cask (800 rods).

The 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 the 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. A critical core was modeled by adjusting fuel rod pitch (actual pitch not available; core average source changes by less than 1% over evaluated range of pitch). By setting the system to critical (k_{eff}=1) at beginning of life assures that the neutron spectrum is representative of that in the actual core. k_{eff} decreased during the modeled burnup from 1.0 to 0.99. This minor decrease is not expected to significantly effect neutron spectrum or source produced by the calculation. As a full core was modeled, fuel source was extracted at three radial locations (inner, middle, outer ring) to determine which location produces maximum source. The maximum gamma source (controlling for shielding) was obtained from the middle ring location. The middle ring source was then applied to all fuel rods. While the 2-D analysis cannot capture axial distribution of source in a rod, loading of rods in 5x5 arrays four high will assure that the source is relatively uniformly spread through the axial extent of the cask. Any postulated localized peaking in source will be further reduced after penetrating through the radial shield of the NAC-LWT cask. No dose peaking is expected on the cask surface as a result of axial burnup profile of the individual SLOWPOKE rods.

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

5.3.20.2 SLOWPOKE Fuel Shielding Model

MCNP three-dimensional shielding analysis allows detailed modeling of the fuel, basket, and cask shield configurations. Some fuel rod detail is homogenized in the model to simplify model input and improve computational efficiency. The basket and cask body details are explicitly modeled, including the axial extents described by the License Drawings.

The geometric description of a MCNP model is based on the combinatorial geometry system embedded in the code. In this system, bodies such as cylinders and rectangular parallelepipeds, and their logical intersections and unions, are used to describe the extent of material zones.

Source and Canister Models

Options for loading include fuel rod arrays of 4x4 and 5x5 rods. Only the 5x5 array is modeled as it contains maximum fuel/source inventory. These arrays are stacked four high within a canister that also contains a handle. The canister is made of aluminum. Dimensions for the tube array and canister are shown in Table 5.3.20-7. The source region is modeled as a smear within

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. Reference fuel design characteristics, i.e., those documented in available references, are summarized in Table 5.3.23-1. Depletion and shielding evaluation code input characteristics, i.e., input data applied in TRITON depletion/irradiation and MCNP shielding model parameters, are given in Table 5.3.23-2. Key parameters differing between the reference information and code input 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 the 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.

No axial burnup profile for the SLOWPOKE core is available in open literature. Burnup profile impacts gamma and neutron source shape. The primary impact of a burnup profile is neutron source shape because burnup impacts fuel neutron source significantly faster than gamma source. SLOWPOKE HEU cores do not produce a significant neutron source. SLOWPOKE cores apply beryllium reflectors which will reduce axial shape effects. Radial core burnup studies demonstrate a slightly higher power in the periphery rather than a typical drop off. This type of effect from the axial reflectors would produce a slight flattening of the axial dose profile. Furthermore, as the core is ~22 cm in diameter versus a ~100 cm diameter, cask surface geometry effects/dispersion will assure that any minor axial profile on the core will not result in any significant cask surface dose changes.

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

A 0.1374 cm gap between the lead outer diameter and the cask outer shell is applied under normal conditions of operations. A lead gap slump, based on the normal condition gap, 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, with heaters controlling lead conditions, assuring minimal contraction gaps. The modeled top end drop gap is 2.71 inches which is conservative versus the gap calculated in Chapter 2. The radial gap modeled is 0.788 inch. . 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.

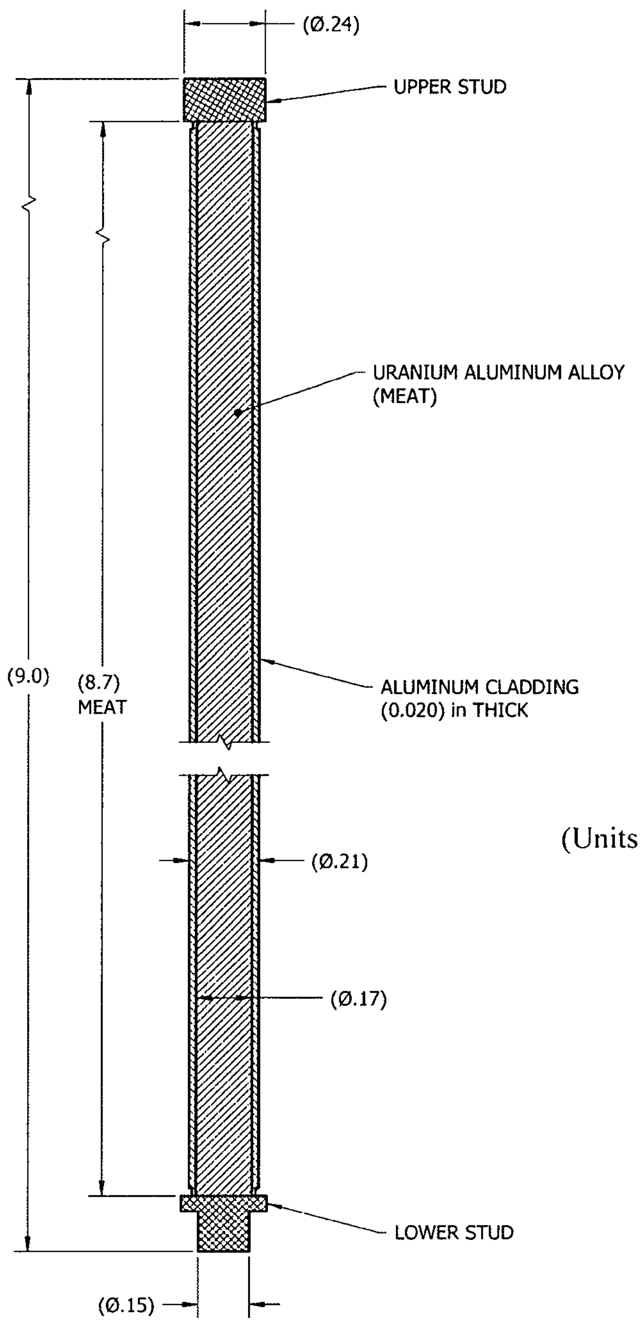
Calculated normal condition radial surface dose rates are below 200 mrem/hr. The Transportation Index (TI) is 15.2 (dose at 1 meter). As the transport index is over 10, an exclusive use designation for the NAC-LWT is used.

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)

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