NAC INTERNATIONAL

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January 14, 2013

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

Attn: Document Control Desk

Subject: Submission of NAC Responses to the NRC's Request for Additional Information to NAC Amendment Request for Certificate of Compliance (CoC) No. 9225 for the NAC-LWT Cask to Incorporate AECL NRU/NRX fuel as Authorized Content

Docket No. 71-9225 TAC No. L24697

Reference:

1. Model No. NAC-LWT Package, U.S. Nuclear Regulatory Commission (NRC) Certificate of Compliance (CoC) No. 9225, Revision 57, August 9, 2012
2. Safety Analysis Report (SAR) for the NAC Legal Weight Truck Cask, Revision 41, NAC International, April 2010
3. ED20120125, Submission of a Request for an Amendment of Certificate of Compliance (CoC) No. 9225 for the NAC-LWT Cask to Incorporate AECL NRU/NRX fuel as Authorized Content, October 26, 2012
4. NRC Letter, Request for Additional Information for Review of the Certificate of Compliance No. 9225, for the Model No. NAC-LWT Package, December 20, 2012

NAC International (NAC) hereby submits responses to Reference 4 in Enclosure 1 to this letter. In response to RAI 3-1, NAC is providing proprietary data input files on CD media. These files were used to perform thermal analyses for normal conditions of transport. This information is to be withheld from public disclosure via 10 CFR 2.390. An affidavit, executed by Mr. George Carver, is enclosed with this letter.

This submittal package includes one hard copy of this transmittal letter and Revision LWT-13A changed pages to the Reference 2 SAR and Reference 3 amendment request. Enclosure 2 contains a brief summary of the changes to the SAR for Revision LWT-13A. Consistent with NAC administrative practice, this proposed SAR revision is numbered to uniquely identify the applicable changed pages. Revision bars mark the SAR text changes on the Revision LWT-13A pages. Enclosure 3 to this transmittal letter lists all drawing changes in detail, if any. The included List of Effective Pages identifies the current revision level of all pages in the Reference 2 SAR.

Enclosure 4 to this transmittal letter includes the requested changes to Reference 1 and Enclosure 5 contains the LWT-13A changed pages.

ED20130001

NMSS01

In order to better facilitate the review process, NAC is providing the Revision LWT-13A change pages with appropriate backing pages. Consequently, a number of LWT-12D and Revision 41 pages are included. In accordance with NAC's administrative practices, upon final acceptance of this application, the LWT-12D and LWT-13A changed pages will be reformatted and incorporated into the next revision of the NAC-LWT SAR.

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

Approval of the amendment to Reference 1 is requested by March 1, 2013, to support obtaining an updated US DOT Competent Authority Certificate, Canadian Foreign Validation and the US Department for Energy/Savannah River Site shipping schedules.

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

Sincerely,

Anthony L. Patko
Director, Licensing
Engineering

Enclosures:

Enclosure 1 – RAI Responses, NRU/NRX Amendment
Enclosure 2 – List of Changes, NAC-LWT SAR, Revision LWT-13A, NRU/NRX Amendment
Enclosure 3 – List of Drawing Changes, NAC-LWT SAR, Revision LWT-13A, NRU/NRX Amendment
Enclosure 4 – Proposed Changes for Revision 57 of Certificate of Compliance No. 9225 for the NAC-LWT Cask NRU/NRX Amendment
Enclosure 5 – Change Pages for Revision 41 of Safety Analysis Report (SAR) for the NAC Legal Weight Truck Cask NRU/NRX Amendment

NAC INTERNATIONAL
AFFIDAVIT PURSUANT TO 10 CFR 2.390

George Carver (Affiant), Vice President, Engineering, of 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 International Legal Weight Truck (LWT) Transport Cask.

   - NAC International Proprietary Calculations
     - ANSYS Data Disc (Thermal) Supporting RAI 3-1 Response (Data Disks 1 thru 1)

   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. George Carver, 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 14 day of January, 2013.

George Carver
Vice President, Engineering
NAC International

Subscribed and sworn before me this 14 day of January, 2013.

Notary Public

SARAH PREHMUS
MY COMMISSION EXPIRES
NOTARY
PUBLIC
MAY 6, 2016
GWINNETT COUNTY, GEORGIA

January 2013

NAC-LWT

Legal Weight Truck Cask System

NRU/NRX RAI Response Package

Docket No. 71-9225

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
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Enclosure 1

RAI Responses

No. 9225 for NAC-LWT Cask

NAC-LWT SAR, Revision LWT-13A

NRU/NRX Amendment

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

December 20, 2012

FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9225, REVISION FOR THE MODEL NO. NAC-LWT PACKAGE TO INCORPORATE NRU/NRX FUEL

(TAC NO. L24697 DOCKET NO. 71-9225)

January 2013

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NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

2-1 Evaluate the NRU and NRX fuel to justify that the geometric form of the contents would not be substantially altered as required by 10 CFR 71.55(d)(2) for the side and end drops evaluated for normal conditions of transport.

While the application does provide criticality analysis for broken fuel rods, it does not address whether the geometric form of the fuel would be substantially altered after the evaluation of the tests for normal conditions of transport.

The information is required to ensure compliance with 10 CFR 71.55(d)(2).

NAC International Response to Structural Evaluation RAI 2-1:

The design of the NRU/NRX fuel basket is fundamentally the same as a damaged fuel can. Specifically, there are 18 screened fuel tubes that share a common basket lid. This prevents any loaded fuel rods from leaving the tube in which they reside.

While NRU/NRX fuel is classified as "undamaged", the package contents will be handled fundamentally the same as a damaged fuel can (e.g. removal of the fuel rods is performed after removal of the basket assembly). The fuel rods will not leave their loaded location during normal and accident conditions of transport due to the screened fuel tube and installed basket lid.

In addition, this physical restraint was the basis for the criticality analyses performed for broken NRU/NRX fuel rods, which is considered the credible fuel configuration if the contents were to be damaged. Thus, there are no fuel-specific functions for the NRU/NRX payload.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

2-2 Clarify the NRU/NRX basket design criteria.

Section 2.1.2.2 (Noncontainment Structures) in the consolidated safety analysis report (SAR) dated June 18, 2010, follows similar structural criteria in Section 2.1.2.1 for containment structures (consistent with Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels"). The Model No. NAC-LWT fuel basket allowable stress limits are listed in Table 2.1.2-2 in the SAR.

To be consistent with NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," the NAC LWT NRU/NRX basket should follow the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV), Section 3, Division 1, Subsection NG for design, fabrication, examination, etc. Subsection NG is a "design by analysis code" that uses the stress intensity limits of NG-3200. It appears that some analysis criteria used (e.g. localized buckling) and the Table 2.1.2-2 are inconsistent with the stress criteria in Subsection NG ASME B&PV Code. For example:

- allowable yield stress is permitted (if greater than S_m), and
- SAR Table 2.1.2-2's ($.5S_u$) accident pure shear criteria vs. NG pure shear criteria ($.42S_u$).

Clarify the application to clearly tabulate the design criteria and ensure that it is consistent with ASME B&PV Code Section III, Division 1, Subsection NG.

The information is needed to ensure compliance with 10 CFR 71.31.

NAC International Response to Structural Evaluation RAI 2-2:

The stress criteria of ASME B&PV, Section III, Div. 1, Subsection NG, paragraph NG-3200 were used to evaluate the stresses calculated for the NRU/NRX basket as documented in Sections 2.6.12.13 for "Normal Conditions of Transport" and Section 2.7.7.15 for "Accident Conditions of Transport".

For Normal Conditions of Transport:	**For Accident Conditions of Transport:**
$P_m < S_m$	$P_m < 0.7S_u$
$P_m+P_b < 1.5S_m$	$P_m+P_b < 1.0S_u$
Shear stress $< 0.6\ S_m$	Shear stress $< 0.42\ S_y$
Bearing stress $< S_y$	Bearing stress is not evaluated
$P_m+P_b+Q < 3.0S_m$	Primary + Secondary stress is not evaluated

The SAR Sections containing the NRU/NRX basket evaluations do not reference Table 2.1.2-2 for any of the stress allowables used in the structural evaluations. Even though the allowable stress limits listed in Table 2.1.2-2 were not used for the NRU/NRX evaluations, Table 2.1.2-2 remains in the NAC/LWT SAR since it has been used as the licensing basis for previously approved NAC-LWT transport license amendments.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

2-3 Clarify the fuel caddy top end drop discussion.

Section 2.6.12.13.1.1 in the application discusses the fuel caddy top end drop analysis, however, the second sentence in the fuel caddy discussion on page 2.6.12-102 states that "the only loading that the caddy will experience is its own weight of 5 lbs during **the bottom end drop**." This statement appears to be incorrect.

The information is needed to ensure compliance with 10 CFR 71.71.

NAC International Response to Structural Evaluation RAI 2-3:

The 2nd paragraph in Section 2.6.12.13.1.1 on page 2.6.12-102 has been corrected to read "The only loading that the caddy will experience is its own weight of 5 lbs during **the top end drop**."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

2-4 Correct/justify the statement in SAR Section 2.7.7.1 that nine fuel basket designs are analyzed for accident conditions, since this application adds a tenth fuel basket design that is analyzed for accident conditions.

The information is needed to ensure compliance with 10 CFR 71.33(b).

NAC International Response to Structural Evaluation RAI 2-4:

The second sentence of the first paragraph of Section 2.7.7.1 has been corrected to read "**Ten** fuel basket designs are analyzed for accident condition loads."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

3-1 Provide the supporting thermal ANSYS input files that support the results for the normal conditions of transport thermal analysis given within the SAR.

Staff reviewed "Chapter 3: Thermal Evaluation" of the SAR. Staff viewed the thermal results generated from a set of ANSYS files.

The information is needed to ensure compliance with 10 CFR 71.71.

NAC International Response to Thermal Evaluation RAI 3-1:

The supporting NAC proprietary thermal ANSYS input files that support the results for the normal conditions of transport thermal analysis are provided on CD media in a separate sealed envelope. An affidavit pursuant to 10 CFR 2.390 is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-1 Provide the conditions under which each analysis in Table 6.7.2-6 in the application is performed.

Include details for fuel type, enrichment, moderation inside the tubes, moderation inside the basket (outside the tubes), condition/presence of cask outer shell, lead shield, and neutron shield.

The information is needed to ensure compliance with 10 CFR 71.35, 10 CFR 71.43, 10 CFR 71.51, and 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-1:

Newly created Table 6.7.2-7, located on page 6.7.2-29 contains the requested information. The List of Tables has been updated on page 6-xvi and Section 6.7.2.10 is modified to add a reference to Table 6.7.2-7.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-2 Explain why the k_{eff} for an infinite array of packages after the tests for normal conditions of transport (0.07690) is so much lower than the k_{eff} for a single package.

The explanation in Section 6.7.2.8 says: "*Normal conditions are based on an infinite array of packages, square array / touching casks, with a dry cask interior with optimum moderator between casks. Both full density moderator and void were evaluated between casks, and the maximum reactivity is achieved by the array having a dry exterior....*" Does the single cask have a dry cask interior? It is not clear whether the volume inside the tubes is dry or just between the fuel tubes. Justify that the configuration chosen is the most conservative.

The information is needed to ensure compliance with 10 CFR 71.35 and 10 CFR 71.59.

NAC International Response to Criticality Evaluation RAI 6-2:

The single cask was evaluated using maximum reactivity, moderated to the maximum extent per 71.55 (b)(2). With a moderated interior cask reactivity is high as demonstrated in the SAR sections preceding 6.7.2.8. For normal condition array of casks, NUREG-1617 in Section 6.5.5.1 states that "water inleakage need not be assumed." As the analysis followed the guidance of NUREG-1617, no further analysis is required. Per this section, and 10 CFR 71.59, the cask exterior is also dry but in this calculation both dry and wet exterior were evaluated.

The cask cavity is considered to be dry in the evaluation. This includes the space inside the tube and between the tubes. The SAR, Section 6.7.2.8, is modified to clarify this statement.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-3 Discuss the use of the caddy (Drawing No. 315-40-175, Rev. 1) for NRU fuel and ensure this discussion is consistent across the application. Clarify which contents are required to be in the fuel caddy and which contents may be placed in the fuel caddy.

The contents listing on page 1-4 states, *"Loose rods may be placed into an aluminum caddy prior to placement into the basket. 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. NRX fuel rods shall be loaded into a fuel rod caddy for handling."*

Section 1.1 states: *"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."*

Section 2.6.12.2 in the application states, *"Each fuel tube is capable of holding a single NRU/NRX fuel assembly, or up to 7 NRX rods or up to 12 NRU rods, positioned inside an aluminum caddy."* Based on this description, the aluminum caddy (which has been evaluated for normal conditions of transport and hypothetical accident conditions side and end drops), is ONLY required for transport content of 1-7 NRX rods or 1-12 NRU rods. This is consistent with the statement made in Section 2.7.7.15, *"Each fuel tube is capable of holding a single NRU/NRX fuel assembly, or equivalent number of fuel rods, positioned inside of the aluminum caddy."*

However, the criticality section states, *"Based on the results of the previous sections, any full loading of 18 undamaged NRU or NRX assemblies is allowed in the NAC-LWT. Undamaged fuel assemblies can include cropped fuel, loose fuel rods or damaged fuel clad, provided the rod is structurally sound. NRX fuel must be placed into a caddy. Maximum reactivates are summarized in Table 6.7.2-6."*

The information is needed to ensure compliance with 10 CFR 71.31 and 10 CFR 71.33.

<u>NAC International Response to Criticality Evaluation RAI 6-3:</u>

Page 1-4 has been modified to make the description match Section 1.1 by:

1. Revising the 4th sentence in the 3rd bullet to "Assemblies/rods may be placed into ....." and revising the last sentence in the 3rd bullet to "NRX fuel assemblies/rods shall be loaded ....."
2. Section 2.6.12.13 (correct section for quoted text in this RAI), Page 2.6.12-95, 1st full paragraph, 3rd sentence is revised as follows:

   "Each fuel tube is capable of holding a single NRU/NRX fuel assembly or equivalent fuel rod quantity. NRU/NRX fuel assemblies/rods may be placed inside of an aluminum caddy. NRX

fuel assemblies/rods shall be placed into a caddy. The total weight inside each fuel tube is limited to 20 pounds, which includes the weight of the aluminum caddy and NRU/NRX fuel assembly/rods."

3. Section 2.5.12.13, Page 2.6.12.102, 1st full paragraph of "Fuel Caddy", 1st sentence is revised to:

   "The fuel assembly/rods may be contained in an aluminum caddy."

4. Section 6.7.2.10 (criticality), Page 6.7.2-6, 3rd sentence is modified to:

   "... is structurally sound. NRU fuel may be placed into a caddy while NRX must be placed into a caddy."

5. Section 6.7.2.1, Page 6.7.2-1, 2nd and 3rd sentences from the end of the paragraph, are revised to clarify that NRX fuel must be placed into a caddy by replacing "are" with "must" when referring to the use of a caddy with NRX fuel. Wording is revised to eliminate the reference to "handling prior to loading" as placement of the fuel into the caddy could be done within the basket. The use of NRU fuel and caddy is made optional. The revised text is as follows:

   "NRX fuel assemblies/rods must be placed into a caddy. NRU fuel assemblies/rods may be placed into a caddy."

6. Section 6.7.2.1, Page 6.7.2-1, 2nd sentence has an editorial change. In the second sentence assemblies are referred to as intact. As the assembly is cropped, this statement may be incorrectly interpreted. Therefore, "intact" is removed.

7. The proposed CoC, Section 5.(b)(2)(xx) on Page 27 of 31, has been revised to delineate that placement of NRX fuel within a caddy is required while the placement of NRU fuel is optional by adding the following sentences:

   "NRX fuel shall be placed in a caddy. Placement of NRU fuel in a caddy is optional."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-4 Clarify the differences between the models for normal conditions of transport and hypothetical accident conditions. Include any differences in materials used.

Section 6.7.2.2 states *"The models are analyzed separately under normal conditions and hypothetical accident conditions to ensure that all possible configurations are subcritical."*

The information is needed to ensure compliance with 10 CFR 71.35, 10 CFR 71.43, 10 CFR 71.51, and 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-4:

The response to RAI 6-5 revised the text on page 6.7.2-2, in the middle of the page in Section 6.7.2.2 to clarify that the differences between normal and accident conditions for the cask are the removal of the neutron shield, the neutron shield shell, and the impact limiters for the accident condition models. There are no other differences in materials used.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-5 Clarify the modeling assumptions with respect to the neutron shield.

Section 6.7.2.3 states the payload was most reactive when assuming loss of the neutron shield. Is this an assumption made for all normal conditions of transport and hypothetical accident condition criticality calculations shown in Table 6.7.2-6?

Provide justification for this assumption. The staff finds that there are competing reactivity effects, since there is boron present in the neutron shield, this may reduce the reactivity, however, the neutron shield also provides neutron reflection, which may increase the reactivity.

The information is needed to ensure compliance with 10 CFR 71.35, 10 CFR 71.43, 10 CFR 71.51, and 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-5:

In Section 6.7.2.2, the subsection entitled "Description of Calculational Models" is revised to add the text below. The text clarifies that boron is not included in the neutron shield and that normal condition models contain a neutron shield and accident condition models do not. Note that as boron is a thermal neutron absorber and that neutrons at thermal energies in the shield region must travel through steel and lead cask shields and a flooded cask cavity prior to reaching fissile material. The absence or presence of boron in the neutron shield is therefore not expected to significantly influence the results of the analysis.

"Normal condition analyses model the cask with the liquid neutron shield in place, while accident conditions remove the neutron shield and the neutron shield shell from the model. Accident conditions also remove the aluminum honeycomb impact limiters from the model. The NAC-LWT neutron shield contains soluble boron. Modeled neutron shield material, as listed in Table 6.7.2-4 does not include boron (removing a neutron absorber). For fully water-reflected single cask models, this produces similar reactivities (see Table 6.7.2-6) as radial cask model neutronic differences are limited to reflection by the ethyl glycol / water mixture and thin neutron shield shell versus reflection by water."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-6 Clarify the version of MCNP5 used.

Section 6.7.2.2 states that the criticality evaluations were performed using the MCNP5 code package. The benchmarking results in Section 6.5.4 provide results for MCNP5 version 1.30 whereas the benchmarking results in Section 6.5.5 provide results for MCNP5 version 1.60.

The information is needed to ensure compliance with 10 CFR 71.3 and 10 CFR 71.35.

NAC International Response to Criticality Evaluation RAI 6-6:

Section 6.7.2.2, Page 6.7.2-2, first sentence at top of page is revised to state that Version 1.60 of MCNP is used in the NRU/NRX criticality evaluations. The revised sentence is as follows (change underlined): "Each model uses the MCNP5 (Version 1.60) code package with the ENDF/B-VI cross-section set."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-7 Provide clarifying information on what is the most reactive moderation configuration.

Section 6.7.2.3 says that the payload is most reactive with flooded cask cavity and exterior, then Section 6.7.2.4 says (1st paragraph) *"The most reactive configuration for both fuel types is a system where the tube is fully flooded at maximum density water while the cask cavity is dry."* Then later in Section 6.7.2.4, the application says: *"Most reactive condition switches to a wet cask cavity for NRX fuel when considering a radial out fuel rod pitch."* In Figures 6.7.2-16 and 6.7.2-17 for the broken rods, it appears that the most reactive configuration for both fuel types is with the moderator density of the cask interior at some value between 30-90%. The staff does not understand the varying statements.

The information is needed to ensure compliance with 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-7:

Section 6.7.2.3 lists the configuration determined to produce the maximum reactivity. This is an iterative process arrived at by the discussion in Sections 6.7.2.4 through 6.7.2.8. The most reactive moderator condition is dependent on the applied fuel configuration.

Section 6.7.2.4 starts analysis with undamaged fuel rods when the rods are at an "as-is/in-core" configuration. At this configuration, results shown in Figures 6.7.2-11 and 6.7.2-12 demonstrated that cask loaded with either fuel type significantly increases in system reactivity when raising fuel tube interior moderator up to full density (indicating an under-moderated rod array). Fuel tube exterior moderator density showed decreasing reactivity with increasing density. The tube exterior moderator effect is significantly less pronounced than that of the tube interior. This analysis supports the 1st paragraph statement.

As the fuel rod geometry and caddy location are not fixed, a location study was performed, as illustrated in Figures 6.7.2-12 and 6.7.2-13. Consistent with the under-moderated lattice statement made in the preceding paragraph, the result of Figures 6.7.2-11 and 6.7.2-12 confirms that rods apart (providing space for additional moderator in the system) increased system reactivity. When fuel rods move to the perimeter of the fuel tube, the optimum moderation between tubes conclusion may be affected. Rather than serving as an absorber, the water in this condition may serve as the required moderator. Rods interacting at the perimeter fuel tube locations now require moderator between tubes to slow neutrons to fission energies (in the nominal configuration there is moderator along the outer surface of the rods as seen in the in-core VISED slices shown in Figures 6.7.2-5 and 6.7.2-6). Due to the potential of this effect, both wet and dry cask cavity (outside the fuel tube) systems were evaluated. For NRU fuel, this resulted in the wet cask being very similar in reactivity to that of the dry cask cavity and for the NRX fuel, the wet cavity being bounding. This supports the statement later in Section 6.7.2.4, quoted above.

The configurations discussed in Section 6.7.2.4 are low reactivity configurations with maximum reactivity less than 0.8. To determine if a more reactive configuration could be achieved, Section 6.7.2.5 evaluates a hypothetical condition of the fuel rods breaking. Fuel rods are aluminum alloy and not expected to significantly alter in configuration as a result of any condition of operation. The broken rods scenario significantly affects moderator studies previously performed as the resulting reduced active fissile material height has less axial leakage but also is significantly under-moderated (within the tube). The trade-off documented in Figures 6.7.2-14 and 6.7-15 demonstrates that leakage dominates and that a smaller height fuel region with more rods is more reactive. Since the reconfiguration of the fuel region will impact the conclusion drawn for the outside tube moderator density study (and may impact the tube inner bounding density), additional optimum moderator density studies were run. Figures 6.7.2-16 and 6.7.2-17 document the results of these additional studies. Maximum density inside the fuel tube was found to still be bounding (as expected). For NRX fuel, a peak was found for tube outside (cask cavity outside the fuel tube) moderator density of 0.55 g/cm^3 with no statistically significant change between 0.35 and 0.80 g/cm^3. Maximum reactivity for this configuration is ~0.85. The more reactive, bounding, NRU HEU fuel showed a statistically flat plot between 0.50 and 0.9982 g/cm^3. As there was statistically significant variation between the various data points above 0.50 g/cm^3, full density moderator was chosen for component tolerance and for final maximum reactivity reporting.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-8 Clarify what is meant by *"radial maximum shift of the outer rods"* in the section 6.7.2.4, second paragraph. Additionally, clarify the legend of Figures 6.7.2-12 and 6.7.2-13.

The staff does not understand this explanation of the most reactive fuel pitch and the legends are not clear in terms of what they are supposed to represent.

The information is needed to ensure compliance with 10 CFR 71.35, 10 CFR 71.43, 10 CFR 71.51, and 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-8:

SAR Section 6.7.2.4 is revised to include additional description/explanation of the modeled geometry. The added text links Figure 6.7.2-5 and Figure 6.7.2-6, which contain illustrations of the rod geometries. The illustrations are updated to point out inner/outer rod locations.

Text added to Section 6.7.2.4 is as follows:

"The results from the models presented in Figures 6.7.2-5 and 6.7.2-6 are shown in Figures 6.7.2-12 and 6.7.2-13, respectively. The model figures are annotated to indicate outer ring fuel locations for NRU and NRX fuel and inner rod locations for the NRU fuel. In the context of the result figures "nominal" refers to the "as-built/in-core" location of the rods. Maximum outer rod location indicates shift of the outer rods away from the tube center to the maximum permitted by tube. Shift along the "x-axis" from 1 to 11 moves the referenced rod type (inner/outer) radially out from the center of the tube, with "1" indicating a close in (minimum radial location) shift and "11" indicating the maximum permitted shift (each shift direction is limited by either adjacent rods or tube/caddy).

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-9 Provide the dimensions of the water reflector.

Section 6.7.2.7 states that *"the most reactive case is reevaluated by removing the lead and outer shells (including neutron shield), and reflecting the system by water at full density on the X, Y, and Z faces."*

The information is needed to ensure compliance with 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-9:

Twenty centimeters of water is the water reflector. The sentence in Section 6.7.2.7 is revised (change underlined) to "...the most reactive case is reevaluated by removing the lead and outer shells (including neutron shield), and reflecting the system by 20 cm of water at full density on the X, Y, and Z faces."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-10 Clarify the calculated k_{eff} in Section 6.7.2.7. Explain what the analysis assumptions are for this value with respect to moderation and pitch and fuel.

Section 6.7.2.7 states that *"Using the maximum reactivity model from Section 6.7.2.5, the calculated k_{eff} + 2σ is 0.85218."* The value of 0.85218 is not reproduced in the summary table of maximum reactivity, Table 6.7.2-6.

The information is needed to ensure compliance with 10 CFR 71.55.

NAC International Response to Criticality Evaluation RAI 6-10:

The value of 0.85218 is not included in the summary table as it is significantly below that of the normal condition cask including lead and outer steel shell. Greater neutron reflection is produced by retaining the lead and outer stainless steel cask shell. The analysis is consistent with the requirements of 10 CFR 71.55 (b)(3).

To clarify the steps taken between accident, normal, and containment reflected conditions, Section 6.7.2.5 is revised by adding a paragraph to the end of the section to provide a link between the accident condition cask model used in the fuel configuration/moderator/tolerance studies (maximum k_{eff} including 2 sigma of 0.92560 and the reported value for the normal condition model (k_{eff} + 2σ of 0.92525).

Section 6.7.2.7 is revised to provide the link to the normal condition model by revising text (change underlined) to state:

"While no operating condition results in a removal of the cask outer shell and lead gamma shield, the most reactive normal condition case (i.e., the k_{eff} of 0.92525 case containing HEU NRU fuel with broken fuel rod sections, fully moderated cask interior, described in Section 6.7.2.5) is reevaluated by removing the lead and outer shells (including neutron shield), ..."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-11 Clarify statements in the SAR regarding the upper subcritical limit (USL) trend with respect to enrichment.

Section 6.5.5.3 that states: *"The range of applicability (area of applicability) of this limit may be extended to lower enrichment as the correlation shows an increase in USL as a function of reduced enrichment. The range may be extended up to fully enriched (100% ^{235}U) as the USL is only very weakly correlated to enrichment and that an extrapolated USL based on the enrichment correlation results in a higher predicted USL value than the EALCF derived 0.9270."* This statement appears contradictory. It is not clear how extrapolating the USL to 100% enrichment gives a higher predicted USL value, if the USL is negatively correlated to enrichment (also shown in Figure 6.5.5-1).

The information is needed to ensure compliance with 10 CFR 71.31 and 10 CFR 71.35.

NAC International Response to Criticality Evaluation RAI 6-11:

The paragraph contains two distinct arguments. One statement for going below the 17% minimum AOA and one for going above the 93.2 maximum. Each statement is discussed further in the context of the USL correlation in Table 6.5.5-5 and the low coefficient of determination (R^2) documented in Table 6.5.5-4.

"The range of applicability (area of applicability) of this limit may be extended to lower enrichment as the correlation shows an increase in USL as a function of reduced enrichment."

The energy dependent correlation is documented as 0.9300 – 2.1904E-05x, which indicates that the USL will increase as "x" is reduced.

"The range may be extended up to fully enriched (100% ^{235}U) as the USL is only very weakly correlated to enrichment and that an extrapolated USL based on the enrichment correlation results in a higher predicted USL value than the EALCF derived 0.9270."

The coefficient of determination, R^2, for the enrichment correlation is 0.0069 per Table 6.5.5-4. An R^2 near 0 indicates no significant (very weak) correlation of independent and dependent values. As a range of 64% (17% to 93%) is covered by the correlation and no significant dependence occurred, extrapolation to 100% enrichment is considered reasonable. At 100%, the above listed correlation extrapolates to 0.92781 (below the USL low value of 0.928 in Table 6.5.5-5). As stated in the SAR text, this value is still higher than the 0.9270 value obtained from the EALCF correlation (per Table 6.5.5-5).

As stated in the preceding paragraph, applying the 100% enriched extrapolated value (from the enrichment correlation), while producing a lower USL than that of the AOA enrichment range, produces a higher USL than that obtained from EALCF correlation.

The SAR as written represents the data obtained from the benchmark cases.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

6-12 Explain the differences in the benchmarking results between MCNP5 version 1.30 and MCNP5 version 1.60 that causes the USL to increase.

In switching from MCNP5 version 1.30 to MCNP5 version 1.60 the USL increases by about 1% (up from 0.9171 to 0.9270). Please explain the differences in the benchmarking evaluation that causes the difference in USL.

The information is needed to ensure compliance with 10 CFR 71.31 and 10 CFR 71.35.

NAC International Response to Criticality Evaluation RAI 6-12:

Validation of MCNP 5 version 1.30 used experiments covering a larger energy range. As seen in Figure 6.5.4-2 and Table 6.5.4-5, the version 1.30 correlation includes data points with energies (average lethargy of neutrons causing fission – EALCF) up to 1.2 eV. This energy is significantly higher than that which results in maximum reactivity LWT configurations and only limited data points are available at high energies. The validation for version 1.60 was therefore based on a reduced energy data set. Maximum energy was set to 0.4 eV (Figure 6.5.5-2 and Table 6.5.5-5) for version 1.60. This modification removed 7 state points from the sample set (the 7 state points tripled the energy range covered by the remaining 47 points). While neither set of data showed a significant linear fit correlation to EALCF, the correlations, and therefore USL, changed between the two versions (beyond minor changes associated with code/cross sections).

Enclosure 2

List of Changes

NAC-LWT SAR, Revision LWT-13A

NRU/NRX Amendment

Initial Application

List of Changes, NAC-LWT SAR, Revision LWT-13A

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

Chapter 1

- Page 1-4, third bullet near top of page was modified to match Section 1.1.

Chapter 2

- Page 2.6.12-95, revised text beginning with third sentence of first paragraph in Section 2.6.12.13.
- Page 2.6.12-96, bottom of page, revised equation to read "MS = $(1.0S_m/\ \sigma_c)$-1 = 9.0" instead of "FS = $(1.0S_m/\ \sigma_c)$-1 = 9.0"
- Page 2.6.12-98, top of page, revised equation to read "MS = $(0.6S_m/\ \tau)$-1 = large" instead of "FS = $(0.6S_m/\ \tau)$-1 = large"
- Page 2.6.12-101, top of page, revised equation to capitalize the letter "m" in "MS = $(1.0S_m/\ \sigma_c)$ -1 = 0.84"
- Page 2.6.12-102, revised the first sentence under heading "Fuel Caddy" to state "The fuel assembly/**rods** may be contained in an aluminum caddy." Revised the second paragraph to state "The only loading that the caddy will experience is its own weight of 5 lbs during the **top end drop.**"
- Page 2.6.12-102, bottom of page, revised equation to read "MS = $(1.0S_m/\ \sigma_c)$-1 = large" instead of "FS = $(1.0S_m/\ \sigma_c)$-1 = large"
- Page 2.7.7-1, revised second sentence of the paragraph to state "**Ten** fuel basket designs are analyzed for accident condition loads:"
- Page 2.7.7-75, bottom of page, revised text in equation and note to read "Margin of Safety" and "Note: all margins of safety…", instead of "Factor of Safety" and "Note: all factors of safety…".
- Page 2.7.7-78, bottom of page, revised equation to read "MS = $(0.7S_u/\ \sigma_c)$-1 = large" instead of "FS – $(0.7S_u/\ \sigma_c)$-1 = large"
- Page 2.7.7-80, center of page, revised equation to read "MS_{m+b} = $(1.0S_u/P_m+P_b)$ -1 = 0.09" instead of "FS_{m+b} = $(1.0S_u/P_m+P_b)$ -1 = 0.09"

Chapter 3

- No changes.

Chapter 4

- No changes.

List of Changes, NAC-LWT SAR, Revision LWT-13A (cont'd)

Chapter 5

- No changes.

Chapter 6

- Page 6-xvi, updated TOC to include new Table 6.7.2-7.
- Page 6.7.2-1, Section 6.7.2.1, first sentence, as an editorial change, removed the word "intact" to prevent incorrect interpretation. Added second to last sentence to end of paragraph, which states "NRU fuel assemblies/rods may be placed into a caddy.
- Page 6.7.2-2, Section 6.7.2.2, first sentence, added "(Version 1.60)" after "MCNP5."
- Page 6.7.2-2, Section 6.7.2.2, second paragraph under heading "Description of Calculational Models", added several sentences of text to the middle of the paragraph.
- Page 6.7.2-3 is changed by text flow.
- Page 6.7.2-4, Section 6.7.2.4, first paragraph, added several sentences of text to the middle of the paragraph.
- Page 6.7.2-5, Section 6.7.2.5, added last paragraph of section near the middle of the page.
- Page 6.7.2-6, Section 6.7.2.7, added text to the first partial paragraph on the page.
- Page 6.7.2-6, Section 6.7.2.8, added text to the middle of the first paragraph.
- Page 6.7.2-6, section 6.7.2.10, modified and added text to the end of the paragraph.
- Page 6.7.2-7, removed text "Figure 6.7.2-2 NRX Fuel Assembly" from the page.
- Pages 6.7.2-8 through 6.7.2-9, made editorial change to bring figure titles for Figures 6.7.2-2 and 6.7.2-4 to correct pages above corresponding figures from incorrect locations at bottom of previous pages.
- Page 6.7.2-10, Figure 6.7.2-5, added arrows and callouts to figure, and text to note "b".
- Page 6.7.2-11, Figure 6.7.2-6, added arrows and callout to figure.
- Page 6.7.2-29, added Table 6.7.2-7 to the section.

Chapter 7

- No changes.

Chapter 8

- No changes.

Chapter 9

- No changes.

Enclosure 3

List of Drawing Changes

NAC-LWT SAR, Revision LWT-13A

NRU/NRX Amendment

List of Drawing Changes, NAC-LWT SAR, Revision LWT-13A

No additional drawing changes, since LWT-12D, are being requested.

Enclosure 4

Proposed Changes for Revision 57 of Certificate of Compliance

No. 9225 for NAC-LWT Cask

NAC-LWT SAR, Revision LWT-13A

NRU/NRX Amendment

Drawings (new)

CoC Page 4 of 31:

LWT 315-40-170, Rev. 1
"LWT Transport Cask Assy., AECL NRU/NRX Components"

LWT 315-40-172, Rev. 0 (Sheets 1-2)
"Lid Assembly, NRU/NRX"

LWT 315-40-173, Rev. 0 (Sheets 1-2)
"Basket Weldment, NRU/NRX"

LWT 315-40-174, Rev. 0
"Basket Spacer, NRU/NRX"

LWT 315-40-175, Rev. 1
"Caddy Assembly, NRU/NRX"

CoC Sections (new)

CoC Page 19 of 31:

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

(xix) Undamaged NRU or NRX fuel assemblies as specified below:

Parameter	NRU (HEU)	NRU (LEU)	NRX
Maximum Cask Heat Load (W)	640.0		
Maximum Per Tube Heat Load (W)	35.6		
Payload Limit (lb/tube)	20.0		
Maximum ^{235}U per rod (g)	43.24	43.68	79.05
Maximum U per rod (g)	48	230	87
Minimum cool time (yr)	19	3	18
Maximum burnup (MWd or wt% ^{235}U Depletion	364.0 87.4	363.0 83.6	375.0 85.1

CoC Page 27 of 31:

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

(xx) For the NRU/NRX fuel described in Item 5.(b)(1)(xix):

Up to 18 undamaged NRU or NRX fuel assemblies (or the equivalent number of loose rods) may be loaded per NRU/NRX fuel basket in accordance with NAC Drawing Nos. 315-40-172, 315-40-173, 315-40-174 and 315-40-175. NRX fuel shall be placed in a caddy. Placement of NRU

fuel in a caddy is optional. Cask configuration to be in accordance with NAC Drawing No. 315-40-170.

CoC Sections (revised)

CoC Page 29 of 31:

5(c) Criticality Safety Index (CSI)

For NRU/NRX fuel described in 5.(b)(1)(xix) and limited in 5.(b)(2)(xx)	100.0

CoC Page 31 of 31

REFERENCES

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

NAC International, Inc., supplements dated February 3, March 2, May 24, October 26, 2012 and January 18, 2013.

Enclosure 5

SAR Page Changes and LOEP

No. 9225 for NAC-LWT Cask

NAC-LWT SAR, Revision LWT-13A

NRU/NRX Amendment

January 2013
Revision LWT-13A

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Docket No. 71-9225

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
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LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

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 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. • up to 4,000 lbs of solid, irradiated and contaminated hardware, which may include fissile material less than a Type A quantity and meeting the exemptions of 10 CFR 71.15, paragraphs (a), (b) and (c). Total allowed mass includes the weight of spacers, shoring and dunnage.
Impact Limiters	Aluminum honeycomb energy absorbers located at the ends of the cask.
Intact LWR Fuel (Assembly or Rod)	Spent nuclear fuel that is not Damaged LWR Fuel, as defined herein. To be classified as intact, fuel must meet the criteria for both intact cladding and structural integrity. An intact fuel assembly can be handled using normal handling methods, and any missing fuel rods have been replaced by solid filler rods that displace a volume equal to, or greater than, that of the original fuel rod.
Damaged LWR Fuel (Assembly or Rod)	Spent nuclear fuel that includes any of the following conditions that result in either compromise of cladding confinement integrity or recognition of fuel assembly geometry. 1. The fuel contains known or suspected cladding defects greater than a pinhole leak or a hairline crack that have the potential for release of significant amounts of fuel particles. 2. The fuel assembly: i. is damaged in such a manner as to impair its structural integrity;

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

2.6.12.13 NRU/NRX Fuel Basket

The NRU/NRX fuel basket assembly consists of a top basket weldment, lid assembly and a basket spacer assembly. The basket weldment is 122.25 inches long and consists of 18 fuel tubes with a 2.5-inch outside diameter with a 0.12-inch wall thickness. Each fuel tube is capable of holding a single NRU/NRX fuel assembly or equivalent fuel rod quantity. NRU/NRX fuel assemblies/rods may be placed inside of an aluminum caddy. NRX fuel assemblies/rods shall be placed into a caddy. The total weight inside each fuel tube is limited to 20 pounds, which includes the weight of the aluminum caddy and NRU/NRX fuel assembly/rods. The 6 center tubes are supported by a center tube assembly which consists of 1.5-inch Schedule 80 pipe with 6 equally spaced scalloped center locators. The lower end of the center tube assembly is welded to the bottom support disk and the top end is bolted to the lid assembly. The 12 outer tubes are supported by 7 scalloped, circular support disks. There is a top support disk, 5 center support disks and a bottom support disk. These support disks have an outer diameter of 13.27 inches and are ½ inch thick. The 12 outer tubes are welded to these support disks on the outside of the tubes.

The spacer assembly is 51.8 inches long and consists of a center tube with an outside diameter of 8.0 inches and a wall thickness of 0.12 inches. This center tube is welded to 3 circular support disks. The spacer assembly rests on a base tube that consists of a 10-inch Schedule 80 pipe. The base tube is welded to the base disk. The total weight of the NRU/NRX basket assembly plus the spacer assembly bears directly on the bottom forging of the cask through the base tube.

The basket assembly, basket spacer assembly and lid assembly are fabricated from SA 240, Type 304 stainless steel. The lid assembly attachment bolts are made from SA564, Type 630 (17-4PH).

2.6.12.13.1 NRU/NRX End Drop Analysis for Normal Conditions of Transport

This section includes the evaluation of the NRU/NRX basket and spacer assembly components for lateral (side) and longitudinal (end) 1-foot drop. The acceleration load of 25 g's is conservatively assumed for normal condition of transport end drop and side drop respectively.

Basket/Lid Assembly

Basket assembly consists of 18 cylindrical tube assemblies with a center tube assembly and 7 outer support disks. The center tube assembly is not attached to the 6 tubes in the middle of the basket but the outer 12 tubes are welded to the outer support disks. All 18 tube assemblies are welded to the bottom support disk. The basket lid assembly is attached to the top support disk and the center tube assembly with 7 lid bolts with a 3/8-16 UNC thread. One bolt attaches the lid to the center tube assembly and 6 bolts attach the lid to the top support disk. The bottom support

disk is attached to the center tube assembly by two 1/4 inch fillet welds around the entire circumference and the outer tubes by a 1/16 inch fillet weld spanning 60 degrees.

2.6.12.13.1.1 Top End Drop

The acceleration for a 1 foot top end drop is 25 g.

Fuel Tube Assembly

The compressive stress on the fuel tube assembly is calculated as:

$\sigma_c = Wg/18(A_{tube}) = 1.86$ ksi for 25 g

where; W = Weight of (basket assembly + 18 fuel assemblies + spacer assembly)
= 695 lbs + 30 + 18(20lbs) + 115 lbs = 1,200 lbs

Note: This is conservative since the weight of the top disk (5 lbs) is not carried by the fuel tubes in a top end drop.

and

$A_{tube} = \pi/4(OD^2-ID^2) = 0.897\ in^2$

OD = 2.5 in
ID = 2.26 in

Margin of safety:

$MS = (1.0S_m/\ \sigma_c)-1 = 9.0$

where S_m = 18.6 ksi for SA240, Type 304 at 400 °F

Fuel tube buckling analysis

The column length to radius ratio is;

$L/r_m = 20.5/1.19 = 17.2$

where;

L= distance between disks = 20.5 in
r_m = mean radius of fuel tube = ¼(OD+ID) = 1.19 in

This means that the tube will not buckle as a classical Euler column, instead the buckling mode would be localized buckling (bellows or diamond pattern). For this mode of buckling, the critical compressive stress is given by Blake;

$$S_{cr} = E\frac{0.605 - m^2 x 10^{-7}}{m(1+0.004\phi)} = 264.1 \text{ ksi}$$

where;

$E = 26.5x10^6$ psi for SA240, Type 304 at 400 °F
$S_y = 20.7$ ksi for SA240, Type 304 at 400 °F
$\phi = E/S_y = 1280$
$m = r_m/t = 9.92$
$r_m = 1/2(OD/2+ID/2) = 1.19$ in
$t = 1/2(OD-ID) = 0.12$ in

Margin of safety:

$MS = (S_{cr}/\sigma_c)-1 = \text{large}$

Note: All margins of safety greater than 10 are reported as large.

Support Disk to Tube Welds

The welds for the middle 5 support disks only carry the disk self-weight in a top end drop. The middle support disks are attached to the outer 12 tubes with a 1/16-inch fillet weld on each side of the support disk. Due to lack of access, these welds only extend 60 degrees in the circumferential direction around each tube.

The primary stress in these attachment welds is shear.

$\tau = Wg/A_{eff} = 0.09$ ksi for 25 g

where;

W = middle disk weight = 5 lbs
$A_{eff} = 12[2(60/360)(\pi D_{tube})(t_{eff})] = 1.39 \text{ in}^2$
$D_{tube} = 2.5$ inch
$t_{eff} = 0.7071(1/16) = 0.0442$ in

Margin of safety:

$$MS = (0.6S_m/\tau)-1 = \text{large}$$

where S_m = 18.6 ksi for SA240, Type 304 at 400 °F

Center Tube to Bottom Disk Weld

The fuel tube assembly consists of a cylindrical tube attached to a tube cap on the bottom of the fuel tube. This cap is then welded to the bottom disk to form part of the center tube assembly. This attachment weld is a full circumference double sided 1/8-inch fillet weld. This weld does not support the full weight of the center tube assembly since the center tube assembly is also bolted to the basket lid assembly at the top of the basket. For the weld calculation it is conservatively assumed that this weld will carry the full weight of the center tube assembly.

The primary stress in these attachment welds is shear.

$$\tau = Wg/A_{eff} = 1.32 \text{ ksi for 25 g}$$

where;

W = center tube assembly weight = 56 lbs
$A_{eff} = 2(\pi D_{center\ tube})(t_{eff}) = 1.06 \text{ in}^2$
D_{tube} = 1.9 inch
$t_{eff} = 0.7071(1/8) = 0.0884$ in

Margin of safety:

$$MS = (0.35 \times 0.6S_m/\tau)-1 = 1.96$$

where S_m = 18.6 ksi for SA240, Type 304 at 400 °F

0.35 = weld quality factor for visual inspection, ASME Section III Subsection NG, Table 3352-1.

Basket Tube End Cap to Bottom Disk Weld

The basket tube end cap is welded to the base disk with a 1/4-inch fillet weld around the circumference. However, the end cap fits inside of a hole in the base disk so this weld would not

$A_{collar\ tube} = \pi/4(OD^2-ID^2) = 2.97\ in^2$

for

OD = 8.0 in
ID = 7.76 in

Margin of safety:

$MS = (1.0S_m/\ \sigma_c) -1 = 0.84$

where S_m = 18.6 ksi for SA240, Type 304 at 400 °F

Bearing Stress between the Basket Lid Assembly and the LWT Cask Lid

$\sigma_b = Wg/A_{collar+cover\ plate} = 0.781$ ksi for 25 g

where;

W = 1,200 lbs

$A_{collar+cover\ plate} = \pi/4(OD^2) - A^2 = 38.43\ in^2$

OD = 8.0 in
A = Width of opening in collar cover plate = 3.44 in

Margin of safety:

$MS = (1.0S_y/\ \sigma_b) -1 = large$

where S_y = 20.7 ksi for SA240, Type 304 at 400 °F

Bottom Support Disk

This disk will only support the weight of the spacer assembly (115 lbs) for the top end drop whereas it will support the total weight of the basket assembly, the basket lid assembly and the 18 fuel assemblies (1,085 lbs) in the bottom end drop case. Therefore, the bottom end drop will be controlling for the bottom support disk.

Spacer Assembly

For the top end drop, the spacer assembly is only subjected to its own weight. However, in the bottom end drop the spacer assembly will have to carry the weight of the basket assembly and the fuel assemblies. Therefore, the bottom end drop will be controlling for the spacer assembly.

Fuel Caddy

The fuel assembly/rods may be contained in an aluminum caddy. The caddy consists of a cylinder with an end cap on the bottom. The cylinder has a 2.0-inch outer diameter and a 0.065-inch wall thickness and is constructed from 6061-T6 aluminum.

The only loading that the caddy will experience is its own weight of 5 lbs during the top end drop.

The compressive stress on the fuel caddy is calculated as:

$\sigma_c = Wg/A_{net} = 0.374$ ksi for 25 g

where; W = Weight of fuel caddy = 5 lbs

$A_{net} = \pi/4(OD^2-ID^2)[360-2\theta)/360] = 0.334\ in^2$

OD = 2.0 in
ID = 2.0-2(0.065) = 1.87 in

and

$\Theta = 2\ Sin^{-1}\ (L/2r) = 53.5°$

where; L= 0.9 in (lateral width of cut-out in caddy wall) and r = OD/2

Margin of safety:

$MS = (1.0S_m/\ \sigma_c)-1 = large$

where $S_m = 5.6$ ksi for 6061-T6 aluminum at 400 °F

2.7.7 Fuel Basket Accident Analysis

2.7.7.1 Discussion

Aluminum and stainless steel fuel baskets support NAC-LWT cask contents and retain them in a subcritical and safe geometry. Ten fuel basket designs are analyzed for accident condition loads: the PWR basket (Section 2.7.7.2); the BWR basket (Section 2.7.7.4); the metallic fuel basket (Section 2.7.7.5); the MTR basket (Section 2.7.7.6); the failed metallic fuel basket (Section 2.10.13); the TRIGA fuel basket (Section 2.7.7.9); the DIDO basket (Section 2.7.7.10); the GA IFM basket (Section 2.7.7.11); the TPBAR basket (Section 2.7.7.12); and the NRU/NRX basket (Section 2.7.7.15). Side and end impact orientations are the two limiting accident cases. In the side drop orientation, the basket is supported in bearing on the inner shell, and all structural loads are transmitted to the cask structure. Analysis shows that the structural load occurring during the end drop will not cause the basket assemblies or the analyzed spacers to buckle.

2.7.7.2 PWR Basket Construction

The PWR basket is cylindrical in shape, and constructed from 6061-T6 aluminum. A central hollow, square cavity supports the cask contents during transport. An aluminum spacer assembly is welded to the bottom of the PWR basket. It supports the fuel basket and contents longitudinally and limits their movement within the cask cavity. Additional spacers may be bolted to the cask lid as required by the length of the contents. A complete description of the basket and its construction is provided in Section 2.6.12. For the shipment of up to 25 individual PWR rods, a spacer canister will be utilized to position the PWR rods within the PWR or BWR basket. The PWR rods and canister are bounded by the PWR basket analyses of Section 2.7.7.3.

2.7.7.3 PWR Basket Analysis

The NAC-LWT cask maximum inner shell diameter is 13.405 inches at 70°F. The basket body outside diameter is 13.25 inches at 70°F. Except when the cask is empty, the cask cavity temperature is always higher than 70°F. The 6061-T6 aluminum alloy expands approximately 1.5 times more per degree Fahrenheit than does stainless steel. During the -40°F ambient, high heat load, normal transport condition, the average inner cavity wall temperature is 151°F. Accounting for the thermal response of the basket, the diameter of the basket body is 13.26 inches. The maximum as-designed gap between the basket and the cavity, when the basket is centered in the cavity, is 0.094 inch (both inner baskets are considered to be at 70°F). The basket is assumed to be in bearing contact with the inner shell during a side drop accident. All

loads from the contents are transmitted through the basket to the inner shell and the cask structure.

2.7.7.3.1 Bearing Stress Calculation – Side Drop

The bearing stress is calculated using Case 6 (Roark, page 320), which models the cylindrical basket in a circular groove. The maximum compressive stress is calculated using:

$$S_{c_{max}} = 0.798 \left[\frac{\dfrac{P(D_1 - D_2)}{D_1 D_2}}{\dfrac{1 - \nu_1^2}{E_1} + \dfrac{1 - \nu_2^2}{E_2}} \right]^{0.5}$$

$$= 2242 \text{ psi}$$

where the material properties at 250°F are:

Stainless Steel

D_1 = 13.405

E_1 = 27.3×10^6 psi

ν_1 = 0.275

Aluminum (6061-T6)

D_2 = 13.25 inches

E_2 = 9.4×10^6 psi

ν_2 = 0.334

contents + basket weight = 4,000 lbs

P_{1g} = 4,000 lb/178 in = 22.5 lb/in

$P_{49.7g}$ = (22.5 lb/in)(49.7 g) = 1,118 lbs/in

The side drop g load, 49.7 g, is determined in Section 2.6.7.4.

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

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

Margin of safety:

$$MS = (0.7S_u/\sigma_c) - 1 = 8.89$$

where S_u = 64 ksi for SA240, Type 304 at 400 °F

Fuel tube buckling analysis

The column length to radius ratio is;

$L/r_m = 20.5/1.19 = 17.2$

where;

L= distance between disks = 20.5 in
r_m = mean radius of fuel tube = ¼(OD+ID)

This means that the tube will not buckle as a classical Euler column, instead the buckling mode would be localized buckling (bellows or diamond pattern). For this mode of buckling, the critical compressive shape is given by;

$$S_{cr} = E\frac{0.605 - m^2 x 10^{-7}}{m(1 + 0.004\phi)} = 264.1 \text{ ksi}$$

where;

$E = 26.5 \times 10^6$ psi for SA240, Type 304 at 400 °F
S_y = 20.7 ksi for SA240, Type 304 at 400 °F
$\phi = E/S_y = 1{,}280$
$m = r_m/t = 9.92$
$r_m = 1/2(OD/2+ID/2) = 1.19$ in
$t = 1/2(OD-ID) = 0.12$ in

Margin of safety:

$$MS = (S_{cr}/\sigma_c) - 1 = \text{large}$$

Note: All margins of safety greater than 10 are reported as large.

Support Disk to Tube Welds

The welds for the support disks are evaluated for normal conditions of transport in Section 2.6.12.13.

The primary stress in these attachment welds is shear.

$$\tau = Wg/A_{eff} = 0.22 \text{ ksi for 61 g}$$

where;

W = middle disk weight = 5 lbs
$A_{eff} = 12[2(60/360)(\pi D_{tube})(t_{eff})] = 1.39\ in^2$
$D_{tube} = 2.5$ inch
$t_{eff} = 0.7071(1/16) = 0.0442$ in

Margin of safety:

$MS = (0.42S_u/\tau) - 1$ = large

where S_u = 64 ksi for SA240, Type 304 at 400 °F

Center Tube to Bottom Disk Weld

The center tube assembly attachment weld is evaluated in Section 2.6.12.13 for normal conditions of transport.

The primary stress in these attachment welds is shear.

$\tau = Wg/A_{eff} = 3.22$ ksi for 61 g

where;

W = center tube assembly weight = 56 lbs
$A_{eff} = 2(\pi D_{center\ tube})(t_{eff}) = 1.06\ in^2$
$D_{tube} = 1.9$ inch
$t_{eff} = 0.7071(1/8) = 0.0884$ in

Margin of safety:

$MS = (0.35 \times 0.42S_u/\tau) - 1 = 1.92$

where S_u = 64 ksi for SA240, Type 304 at 400 °F

0.35 = weld quality factor for visual inspection, ASME Section III Subsection NG, Table 3352-1.

Basket Tube End Cap to Bottom Disk Weld

The basket tube end cap is welded to the base disk with a 3/16 fillet weld around the circumference. However, the end cap fits inside of a hole in the base disk so this weld would not be loaded significantly by the top end drop. The reaction load for the top end drop would be carried primarily by bearing between the end cap and the base disk.

Basket Lid assembly

The basket lid assembly is evaluated in Section 2.6.12.13. A complete discussion and illustration of the FEA model used to evaluate the basket lid assembly is given in that section.

The linearized membrane stress and membrane plus bending stress for a 1 g load from section 2.6.12.13 are 281 psi and 676 psi, respectively. Since this is a linear model, these stresses can be scaled. Therefore for a 61 g load;

$P_m = 17.1$ ksi
$P_m + P_b = 41.2$ ksi

Margin of safety:

$MS_m = (0.7S_u/P_m) - 1 = 1.62$
$MS_{m+b} = (1.0S_u/P_m+P_b) - 1 = 0.55$

where $S_u = 64.0$ ksi for SA240, Type 304 at 400 °F

Lid Collar Tube

The lid collar tube is attached to the top of the basket lid. The lid collar tube is also subjected to compressive stress in the top end drop:

$\sigma_c = Wg/A_{collar\ tube} = 24.6$ ksi for 61 g

where;

W = Weight of (basket assembly + basket lid assembly + 18 fuel assemblies + spacer assembly)
= 695 lbs + 30 lbs + 18(20 lbs) = 115 lbs = 1,200 lbs

$A_{collar\ tube} = \pi/4(OD^2-ID^2) = 2.97\ in^2$

for

OD = 8.0 in
ID = 7.76 in

Margin of safety:

$MS = (0.7S_u/\sigma_c)-1 = 0.82$

where $S_u = 64$ ksi for SA240, Type 304 at 400 °F

Bottom Support Disk

This disk will only support the weight of the spacer assembly (115 lbs) for the top end drop whereas it will support the total weight of the basket assembly, the basket lid assembly and the 18 fuel assemblies (1,085 lbs) in the bottom end drop case. Therefore, the bottom end drop will be controlling for the bottom support disk.

Spacer Assembly

For the top end drop the spacer assembly is only subjected to its own weight. However, in the bottom end drop the spacer assembly will have to carry the weight of the basket assembly and fuel assemblies. Therefore, the bottom end drop will be controlling for the spacer assembly.

Fuel Caddy

The fuel assembly may be contained in an aluminum caddy. The caddy consists of a cylinder with an end cap on the bottom. The cylinder has a 2.0-inch outer diameter and a 0.065-inch wall thickness and is constructed from 6061-T6 aluminum.

The only loading that the caddy will experience is its own weight of 5 lbs during the bottom end drop.

The compressive stress on the fuel caddy is calculated as:

$\sigma_c = Wg/A_{net} = 1.1$ ksi for 61 g

where; W = Weight of fuel caddy = 5 lbs

$A_{net} = \pi/4(OD^2-ID^2)[360-2\theta)/360] = 0.334\ in^2$

$OD = 2.0$ in

$ID = 2.0-2(0.065) = 1.87$ in

and

$\Theta = 2\ Sin^{-1}\ (L/2r) = 53.5°$

where; $L = 0.9$ in (lateral width of cut-out in caddy wall) and $r = OD/2$

Margin of safety:

$MS = (0.7S_u/\ \sigma_c)-1 = large$

where; $S_u = 16.8$ ksi for 6061-T6 aluminum at 400 °F

Bottom End Drop

The acceleration for a 30 foot bottom end drop is 61 g.

Fuel Tube Assembly

The compressive stress on the fuel tube assembly is less than the top end drop since only the weight of the basket assembly and the basket lid assembly is supported in this case.

where; W = Weight of basket assembly + Weight of basket lid assembly = 695 lbs + 30 lbs
= 725 lbs

Since this is less than the weight of 1,200 lbs supported by the tubes in the top end drop, the top end drop is controlling.

Support Disk to Basket Tube Welds

The load on these welds is the same for the bottom end drop as the top end drop therefore the shear stress would be the same.

Center Tube to Bottom Disk Weld

The center tube assembly consists of a cylindrical tube attached to the bottom support disk. This weld was evaluated in Section 2.6.12.13 for normal conditions of transport.

The primary stress in these attachment welds is shear.

$$\tau = Wg/18A_{eff} = 5.73 \text{ ksi for } 61 \text{ g}$$

where;

W = basket + lid + fuel assemblies – bottom disk
= 695 lbs + 30 lbs +18(20 lbs) – 15 lbs = 1,070 lbs

$A_{eff} = (\pi D_{bottom\ disk\ opening})(t_{eff}) = 0.845 \text{ in}^2$

$D_{bottom\ disk\ opening} = 1.52$ inch

$t_{eff} = 0.7071(3/16) = 0.1326$ in

Margin of safety:

$$MS = (0.35 \times 0.42S_u/\tau) - 1 = 3.69$$

where S_u = 64 ksi for SA240, Type 304 at 400 °F

0.35 = weld quality factor for visual inspection, ASME Section III Subsection NG, Table 3352-1.

Basket Lid Assembly

The basket lid assembly will not experience any significant loading in the bottom end drop.

Bottom Support Disk

The bottom support disk was evaluated in Section 2.6.12.13 with a FEA model. A complete discussion of the FEA model is discussed and illustrated in Section 2.6.12.13.

The membrane stress and membrane plus bending stress for a 1 g load are 351 psi and 961 psi, respectively. Since this is a linear model, these stresses can be scaled.

Therefore for a 61 g load;

$P_m = 21.4$ ksi
$P_m + P_b = 58.6$ ksi

Margin of safety:

$MS_m = (0.7S_u/P_m) - 1 = 1.09$
$MS_{m+b} = (1.0S_u/P_m+P_b) - 1 = 0.09$

where $S_m = 64.0$ ksi for SA240, Type 304 at 400 °F

Spacer Assembly

The spacer assembly will be subjected to its own weight plus the weight of the basket, lid and fuel assemblies.

Spacer Tube

The spacer tube will experience a compressive stress from the bottom end drop. The compressive stress is given by;

$\sigma_c = Wg/A_{spacer\ tube} = 24.2$ ksi for 61 g

where:

W = Weight of (basket assembly + basket lid assembly + 18 fuel assemblies + spacer assembly- base disk)

= 695 lbs + 20 lbs + 18(20 lbs) = 115 lbs – 20 lbs = 1,180 lbs
$A_{spacer\ tube} = \pi/4(OD^2-ID^2) = 2.97\ in^2$

for

OD = 8.0 in
ID = 7.76 in

List of Tables (continued)

List of Tables (continued)

6.7.2 NRU and NRX Fuel Assemblies

This section includes input, analysis method, results, and criticality benchmark evaluations for the NAC-LWT cask containing a payload of up to 18 NRU or NRX fuel assemblies. NRU assemblies are built with either highly enriched uranium (HEU) or low enriched uranium (LEU) rods and NRX assemblies contain all highly enriched uranium rods. The uranium fuel meat is composed of an aluminum matrix material.

6.7.2.1 Package Fuel Loading

Up to eighteen NRU or NRX fuel assemblies may be loaded into the NAC-LWT. NRU and NRX rods may be loaded as either loose rods or fuel assemblies. NRU and NRX fuel rods are aluminum clad uranium-aluminum alloy with aluminum end plugs. The NRX fuel assemblies are analyzed at 94 wt% ^{235}U. The NRU fuel is analyzed at 94 and 21 wt% ^{235}U. The NRU assemblies are made up of 12 rods while the NRX assemblies contain 7 rods. NRU and NRX fuel rods contain fins attached to the rod clad. NRU assemblies also contain five spacer disks assuring the rods retain their in-core configurations. Both NRU and NRX assemblies are encased in an aluminum flow tube during in-core operations but NRU assemblies will have their flow tube removed before loading. NRU and NRX fuel assemblies may be cropped before loading into the NAC-LWT. NRX fuel assemblies/rods must be placed into a caddy. NRU fuel assemblies/rods may be placed into a caddy. The caddy has been structurally evaluated to retain its shape through all transport conditions.

Up to 18 NRU or NRX undamaged fuel assemblies may be loaded into the NAC-LWT. Undamaged fuel may include loose fuel rods as the assemblies will be cropped and may have the flow tube removed before loading in the NAC-LWT. Undamaged fuel includes rods with clad damage, provided structural integrity is maintained. Although the aluminum based fuel rods are expected to survive all transport conditions and are not subject to fuel debris formation, such as oxide pellets would be, a damaged fuel composition of fractured rod segments is evaluated.

NRU/NRX fuel rod and assembly characteristics are summarized in Table 6.7.2-1. A sketch of a NRU fuel assembly is shown in Figure 6.7.2-1 and a NRX fuel assembly is shown in Figure 6.7.2-2.

6.7.2.2 Criticality Model Specifications

This section describes the models that are used in the criticality analyses for the NAC-LWT cask containing up to 18 NRU or NRX fuel assemblies. The models are analyzed separately under normal conditions and hypothetical accident conditions to ensure that all possible configurations are subcritical.

Each model uses the MCNP5 (Version 1.60) 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

NRU and NRX fuel are modeled in the NAC-LWT. Fuel parameters in Table 6.7.2-2 are employed for the evaluations of HEU and LEU types and are based on the data presented in Table 6.7.2-1. As the fuel will be cropped before loading the end fittings are not modeled in the evaluation.

Evaluations are performed for fuel at in-core conditions, loose fuel rods, and fractured rod sections (broken rods). The NAC-LWT has potential for significant neutron interaction when placed into an array configuration, while assuming a loss of the neutron shield. To eliminate this interaction all models, except the 10 CFR 71.59 normal condition array, employ a single cask, fully water reflected boundary. Normal condition analyses model the cask with the liquid neutron shield in place, while accident conditions remove the neutron shield and the neutron shield shell from the model. Accident conditions also remove the aluminum honeycomb impact limiters from the model. The NAC-LWT neutron shield contains soluble boron. Modeled neutron shield material, as listed in Table 6.7.2-4, does not include boron (removing a neutron absorber). For fully water-reflected single cask models, this produces similar reactivities (see Table 6.7.2-6) as radial cask model neutronic differences are limited to reflection by the ethyl glycol / water mixture and thin neutron shield shell versus reflection by water. The basket and cask models constructed for the NRU/NRX assemblies evaluations are based on the dimensions listed in Table 6.7.2-3. The caddy in the NRX model restricts component movement.

NRU and NRX rod sketches, including the radial fins attached to the clad, are illustrated in Figure 6.7.2-3 and Figure 6.7.2-4. The end plug contains a small section inserted into the fuel region. This region was modeled as fuel meat instead of an end plug. The minor quantity of additional fuel will not affect the conclusions of this analysis.

Figure 6.7.2-5 is a VISED cross-section of three basic configurations of NRU fuel evaluated. For the NRU fuel, these configurations are (a) fuel rods at a pitch identical to that of the rods during in-core configuration. The second configuration (b) represents the fuel with a maximum, most reactive pitch. The third configuration (c) represents a hypothetical condition where the rods are modeled as rod segments. Figure 6.7.2-6 is a VISED cross-section of three basic configurations of NRX fuel evaluated. For the NRX fuel, these configurations are (a) fuel rods at a pitch identical to that of the rods during in-core configuration; this includes the flow tube and fuel caddy. The second configuration (b) represents the fuel with a maximum, most reactive pitch. To achieve this pitch the flow tube is removed. The third configuration (c) represents a

hypothetical condition where the rods are modeled as rod segments. NRX models include the caddy.

The NRU/NRX fuel is placed into an 18-tube basket in the NAC-LWT. A bottom spacer is used to shift the NRU/NRX basket up in the NAC-LWT. A cross-section of the NRU/NRX basket loaded in the NAC-LWT is shown in Figure 6.7.2-7.

A radial sketch of the basket cross-section in the NAC-LWT is shown in Figure 6.7.2-8.

This model neglects the impact limiters and models the cask under accident conditions with the neutron shield voided.

Package Regional Densities

The composition densities (gm/cc) and nuclide number densities (atm/b-cm) evaluated in subsequent criticality analyses are shown in Table 6.7.2-4. Displayed are the NRU HEU, NRU LEU, and NRX HEU material densities for the fuel assemblies.

6.7.2.3 Criticality Calculations

This section presents the criticality analyses for the NAC-LWT cask with NRU and NRX fuel assemblies. Criticality analyses are performed to satisfy the criticality safety requirements of 10 CFR Parts 71.55 and 71.59, as well as IAEA TS-R-1. All criticality evaluations performed use a single cask model. An analysis of the NAC-LWT with each of the basket loadings shows that NRX and NRU fuel remain below the USL even when they are considered damaged (rod sections). The payload is most reactive under the following model characteristics.

Maximum OD basket tubes and caddy (NRX only)
Minimum basket tube and caddy (NRX only) thickness
Flooded cask cavity and exterior
Loss of neutron shield

A single cask containment water reflected evaluation is also performed to comply with 10 CFR 71.55(b)(3). The analyses demonstrate that, including all calculation and mechanical uncertainties, the NAC-LWT remains subcritical under normal and accident conditions.

6.7.2.4 NRU/NRX HEU Assembly (Undamaged Configuration)

An undamaged NRU or NRX fuel assembly is placed in each of the 18 tubes in the NRU/NRX basket assembly. Optimum moderator studies for the package are shown in Figure 6.7.2-10 for NRU fuel and Figure 6.7.2-11 for NRX fuel. The studies show that the tube moderator significantly influences system reactivity with cask interior density having a smaller effect on reactivity. The most reactive configuration for both fuel types is a system where the tube is fully flooded at maximum density water while the cask cavity is dry.

The next stage of analysis varies the fuel rod pitch. NRU fuel contains five axial rod spacer disks (disks containing openings for each rod) which prevent rod movement. For conservatism the disks are not credited in the prevention of rod movement. NRX fuel rods are located in a flow tube which fits tightly around the array and in conjunction with the fins attached to the clad to prevent rod movement. For conservatism, the flow tube is removed from the model. Results of the rod spacing studies are shown in Figure 6.7.2-12 for NRU fuel and Figure 6.7.2-13 for NRX fuel. The results from the models presented in Figures 6.7.2-5 and 6.7.2-6 are shown in Figures 6.7.2-12 and 6.7.2-13, respectively. The model figures are annotated to indicate outer ring fuel locations for NRU and NRX fuel and inner rod locations for the NRU fuel. In the context of the result figures, "nominal" refers to the "as-built/in-core" location of the rods. Maximum outer rod location indicates shift of the outer rods away from the tube center to the maximum permitted by the tube. Shift along the "x-axis" from 1 to 11 moves the referenced rod type (inner/outer) radially out from the center of the tube, with "1" indicating a close in (minimum radial location) shift and "11" indicating the maximum permitted shift (each shift direction is limited by either adjacent rods or tube/caddy). Both models indicate that moving fuel toward tube/caddy ID is most reactive. For NRU fuel, the interior rods are spaced approximately at the midpoint between the center of the tube and outer rods are most reactive. Changes in location of the interior rods have only a minor effect on system reactivity near this midpoint. As the NRX fuel is located in a caddy, the location of the caddy in the tube affects system reactivity. As indicated in the plots, a radial shift of the caddy toward the center of the cask is most reactive. The radial maximum shift of the outer rods is most reactive; therefore, no rod shifts require analysis. Note that both wet and dry cask cavities are evaluated in this section. Most reactive condition switches to a wet cask cavity for NRX fuel when considering a radial out fuel rod pitch. There is no statistical difference in reactivity for wet or dry cask cavity conditions for the NRU fuel type.

6.7.2.5 NRU/NRX Fuel in Hypothetical Damaged Condition (Rod Sections)

NRU/NRX fuel sections (broken rods) were evaluated in the NRU/NRX basket tubes. For the NRX rods, the caddy was assumed to retain the rod sections, as the space between the caddy and tube is smaller than a rod and the caddy runs essentially the full length of the basket tube.

Fuel mass is conserved by reducing active fuel height as the number or rod sections increases. Although reducing active fuel height reduces the H/U ratio, it also compacts the fuel region and produces a lower neutron leakage configuration.

Plots of system reactivity for NRU and NRX assemblies, based on a function of number of rod segments, are shown in Figure 6.7.2-14 and Figure 6.7.2.15, respectively. NRU studies were done at fully flooded (1 g/cm^3) interior cask cavity and tube moderator density; as Figure 6.7.2-

demonstrates a near flat reactivity curve as a function of cask cavity moderator density above 50% density. NRX moderator density studies (run with the maximum 15 rod segments feasible within the caddy) indicate small reactivity difference, wet or dry. Therefore, both were evaluated. The radial, in caddy, shift was also applied.

For both fuel types, maximum number of rod sections produced the maximum reactivity. These reactivities are well above those of the initial studies using full active fuel length rods. The k_{eff} of the NRU assembly is well above that of the NRX assembly, primarily due to the NRX fuel being restrained by the caddy. As the NRU fuel is more reactive than the NRX fuel, the effect of manufacturing tolerances on the fuel tube is only evaluated for the NRU elements.

The single cask maximum calculated $k_{eff} + 2\sigma$ is 0.92560, and is the result of a minimum tube wall thickness and maximum tube OD. There is no statistically significant effect of tube OD, as the effects of increased space for the rod shift are offset by increased separation of tube center to center and increased steel tube mass. Small tube wall thickness allows for increased moderator space and rod shift space and reduces the steel which absorbs neutrons while not affecting the tube to tube pitch.

The broken rod model is significantly higher in reactivity than the model of full length rods and, therefore, establishes the CSI. The CSI for accident conditions is 100. As a single cask model is applied, the cask accident model represents both 10 CFR 71.55 and 71.59 configurations.

To confirm that 10 CFR 71.55 required a single normal condition cask (i.e., with neutron shield and impact limiters) is not neutronically significantly different than that of the accident condition cask model, the "rod section" NRU HEU case that resulted in a $k_{eff} + 2\sigma$ of 0.92560 was evaluated with neutron shield and impact limiter on the cask. The result of a $k_{eff} + 2\sigma$ of 0.92525 demonstrates that the two configurations are not statistically different in the context of criticality analysis.

6.7.2.6 NRU LEU Fuel

All previous evaluations were at 94 wt% ^{235}U fuel assemblies. NRU assemblies were also made with an initial enrichment of 19.75 wt% ^{235}U. The enrichment was conservatively increased to 21 wt% ^{235}U and the most reactive configuration of rod sections was re-evaluated. The LEU NRU fuel results in a $k_{eff} + 2\sigma$ of 0.89508, compared to 0.92560 for the HEU fuel.

6.7.2.7 Single Cask Evaluation to Conform to 10 CFR 71.55 Requirements

The 10 CFR 71.55(b)(3) requires an evaluation of the NAC-LWT with the containment system fully reflected by water. The containment for the NAC-LWT is the cask inner shell. While no operating condition results in a removal of the cask outer shell and lead gamma shield, the most

reactive normal condition case (i.e. the k_{eff} of 0.92525 case containing HEU NRU fuel with broken fuel rod sections, fully moderated cask interior, described in Section 6.7.2.5) is reevaluated by removing the lead and outer shells (including neutron shield), and reflecting the system by 20 cm of water at full density on the X, Y, and Z faces. Using the maximum reactivity model from Section 6.7.2.5, the calculated $k_{eff} + 2\sigma$ is 0.85218.

6.7.2.8 Normal Condition Cask Array Evaluation to Conform to 10 CFR 71.59 Requirements

The 10 CFR 71.59 requires the evaluation of 5xN normal condition packages. Normal conditions are based on an infinite array of packages, square array / touching casks, with a dry cask interior with optimum moderator between casks. "Dry cask interior" applies dry conditions within the fuel tubes, between tubes, and any other void space in the cask cavity. Per NUREG-1617 Section 6.5.5.1, water inleakage need not be assumed during normal condition array analysis. Both full density moderator and void were evaluated between casks, and the maximum reactivity is achieved by the array having a dry exterior, resulting in a $k_{eff}+2\sigma$ of 0.07691. The resulting normal condition CSI for the infinite array is 0.

6.7.2.9 Code Bias and Upper Safety Limit (USL)

Critical benchmarks and USL are discussed in detail in Section 6.5.5. The following evaluates the applicability of the USL to the NRU/NRX fuel assemblies.

The EALCF of the most reactive case is 0.123 eV, and is within the area of applicability of the research reactor benchmark. At the thermal energy range of an EALCF at less 0.378 eV, the USL correlation derived in Section 6.5.5 provides a USL of 0.9270.

The LEU NRU assemblies are analyzed at an enrichment of 21 wt% ^{235}U, within the enrichment range of applicability for the USL. All evaluated LEU NRU fuel is below the EALCF USL, which is lower than the USL based on ^{235}U enrichment.

6.7.2.10 Allowable Cask Loading

Based on the results of the previous sections, any full loading of 18 undamaged NRU or NRX assemblies is allowed in the NAC-LWT. Undamaged fuel assemblies can include cropped fuel, loose fuel rods, or damaged fuel clad, provided the rod is structurally sound. NRU fuel may be placed into a caddy while NRX fuel must be placed into a caddy. Maximum reactivates are summarized in Table 6.7.2-6. Conditions at which the maximum reactivity cases occur are summarized in Table 6.7.2-7.

Figure 6.7.2-1 NRU Fuel Assembly

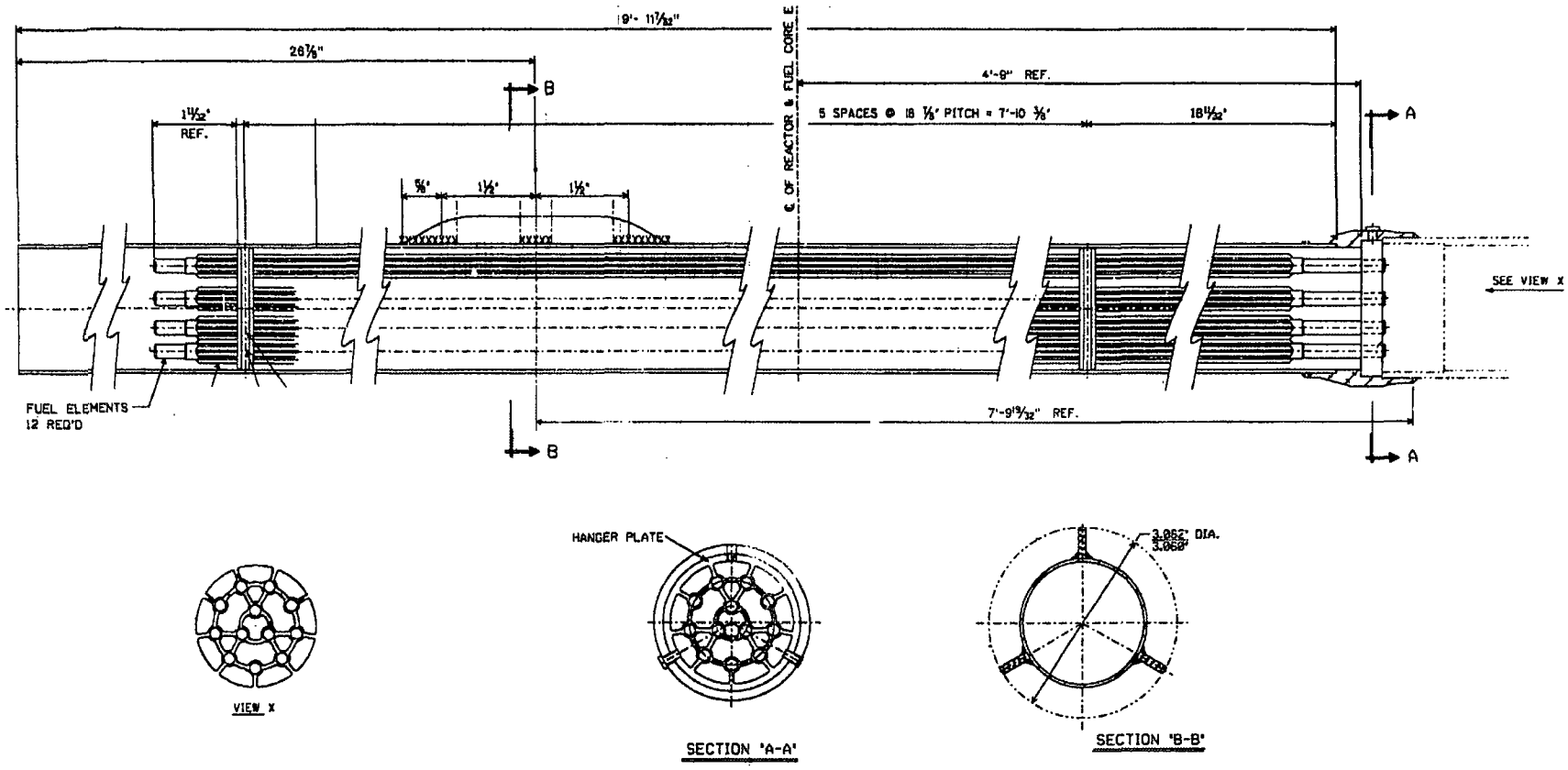


Figure 6.7.2-2 NRX Fuel Assembly

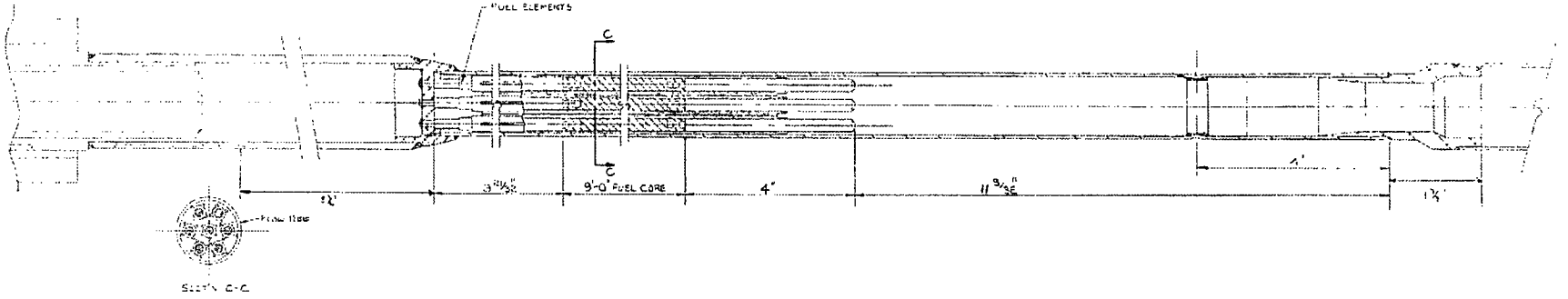


Figure 6.7.2-3 NRU Fuel Rod

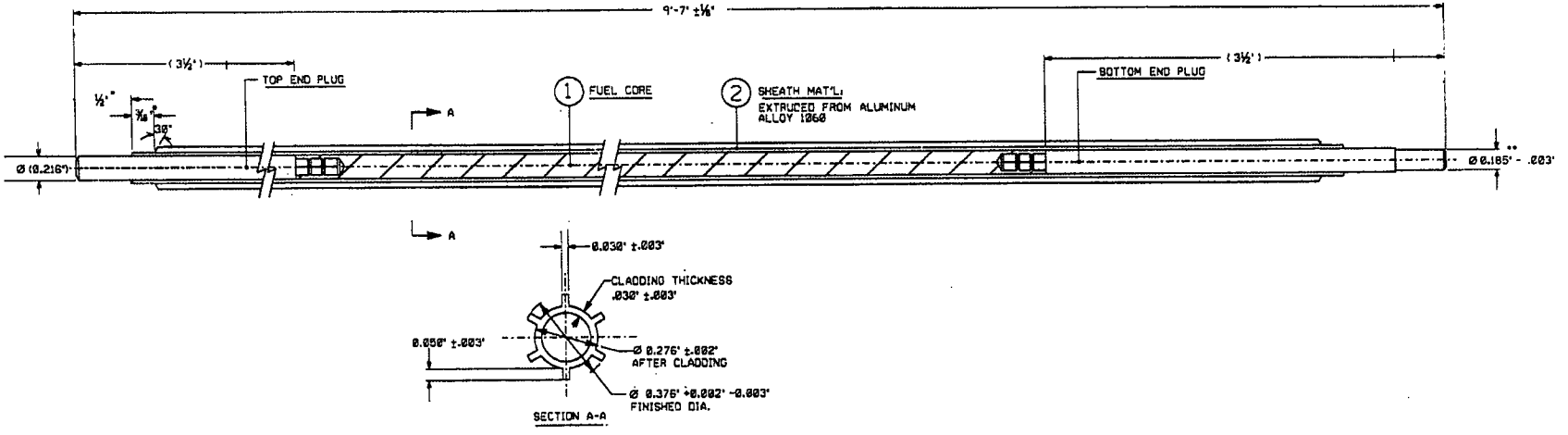


Figure 6.7.2-4 NRX Fuel Rod

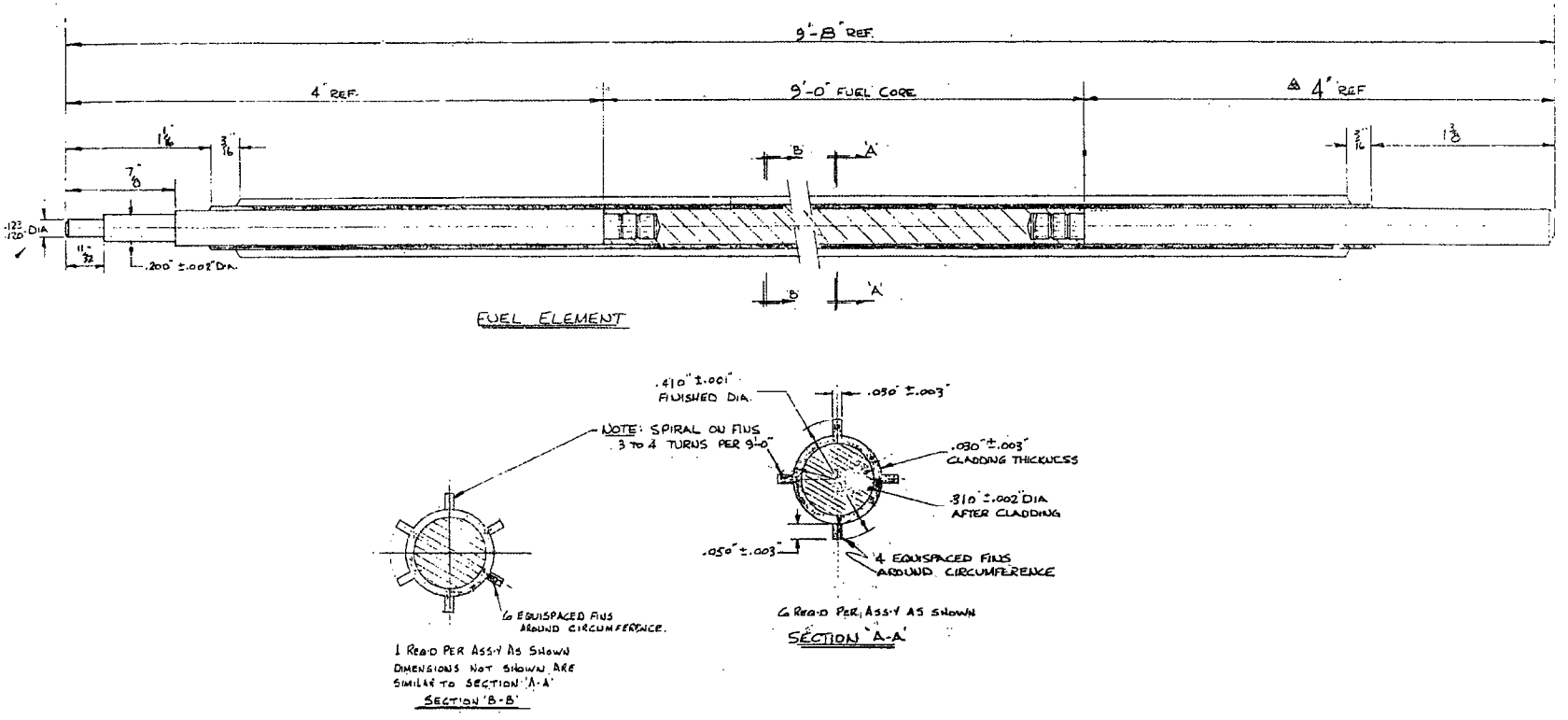
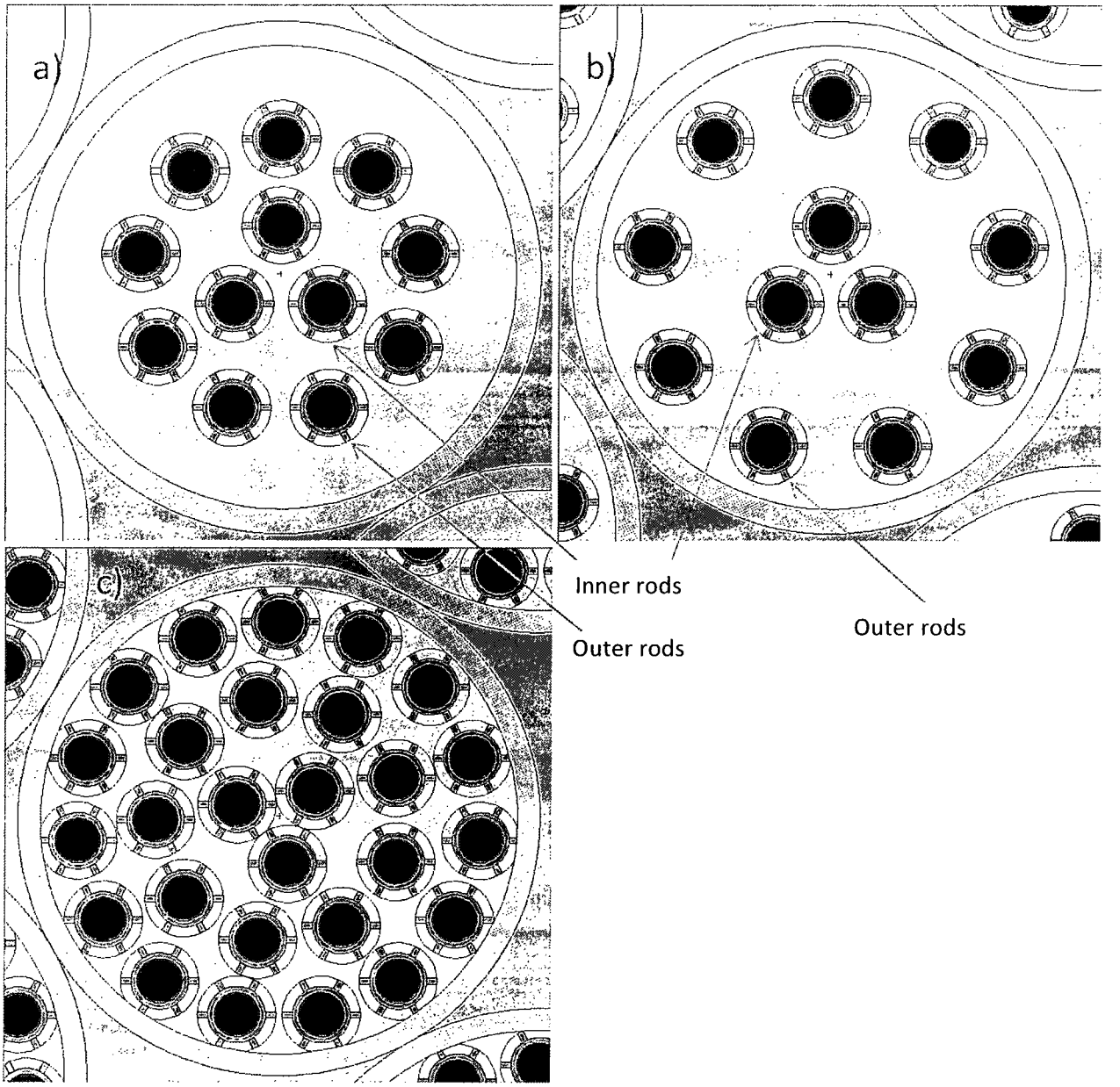


Figure 6.7.2-5 MCNP NRU Fuel in Fuel Tube Cross-Section (No Flow Tube)

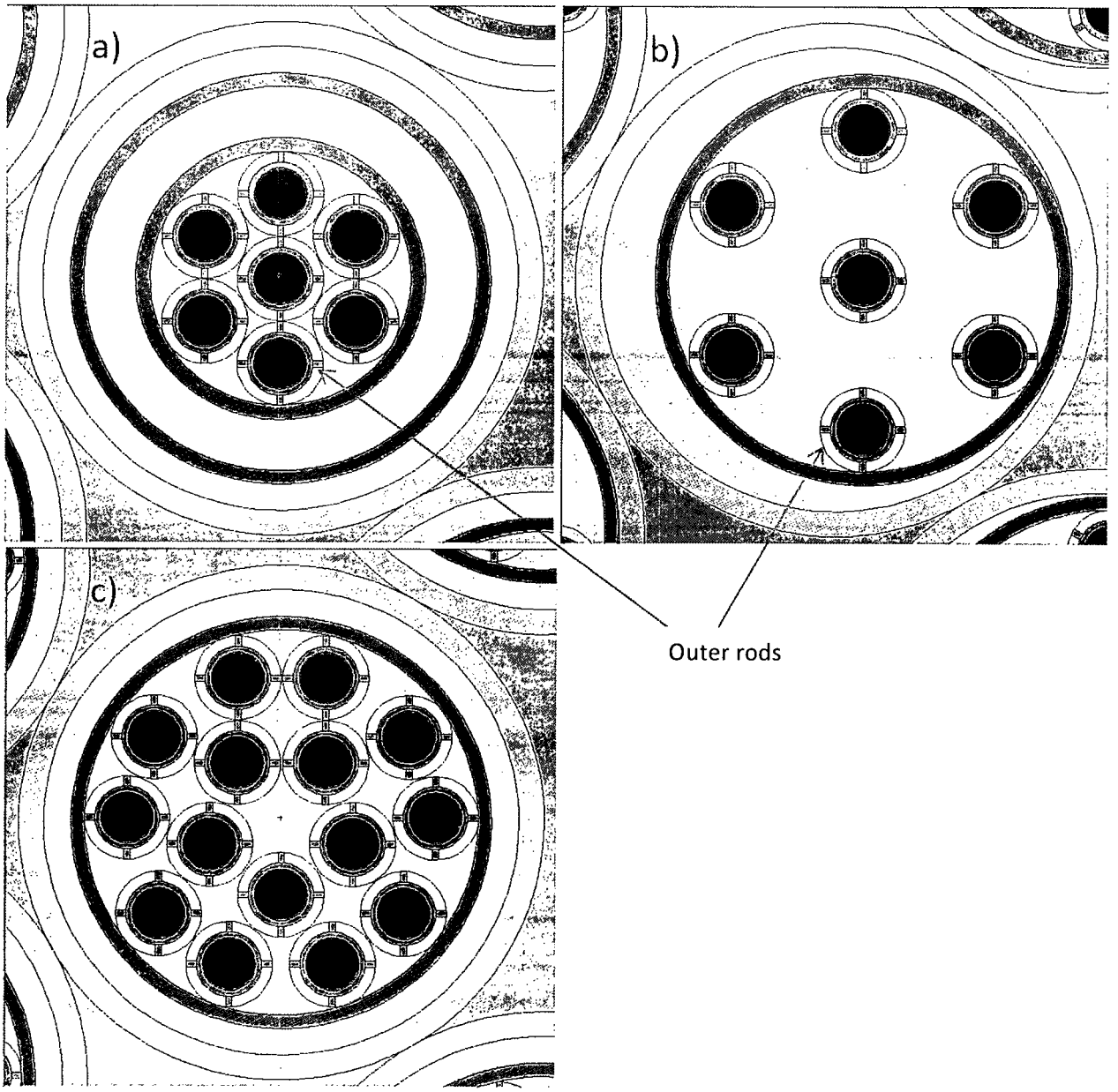


a) Fuel rods in the in-core configuration

b) Maximum reactivity rod pitch (space equivalent to flow tube thickness is retained at inner perimeter of fuel tube but fuel tube is not modeled)

c) Broken Rods – Maximum reactivity is achieved by maximum number of rod sections. Rod height reduced to conserve fuel mass

Figure 6.7.2-6 MCNP NRX Fuel in Fuel Tube Cross-Section (Caddy)



a) Fuel rods in the in-core configuration – with flow tube

b) Maximum reactivity rod pitch – Conservatively removed flow tube

c) Broken Rods – Maximum reactivity is achieved by maximum number of rod sections. Rod height reduced to conserve fuel mass

Figure 6.7.2-7 Sketch of NAC-LWT Cask Cross-Section with NRU/NRX Basket

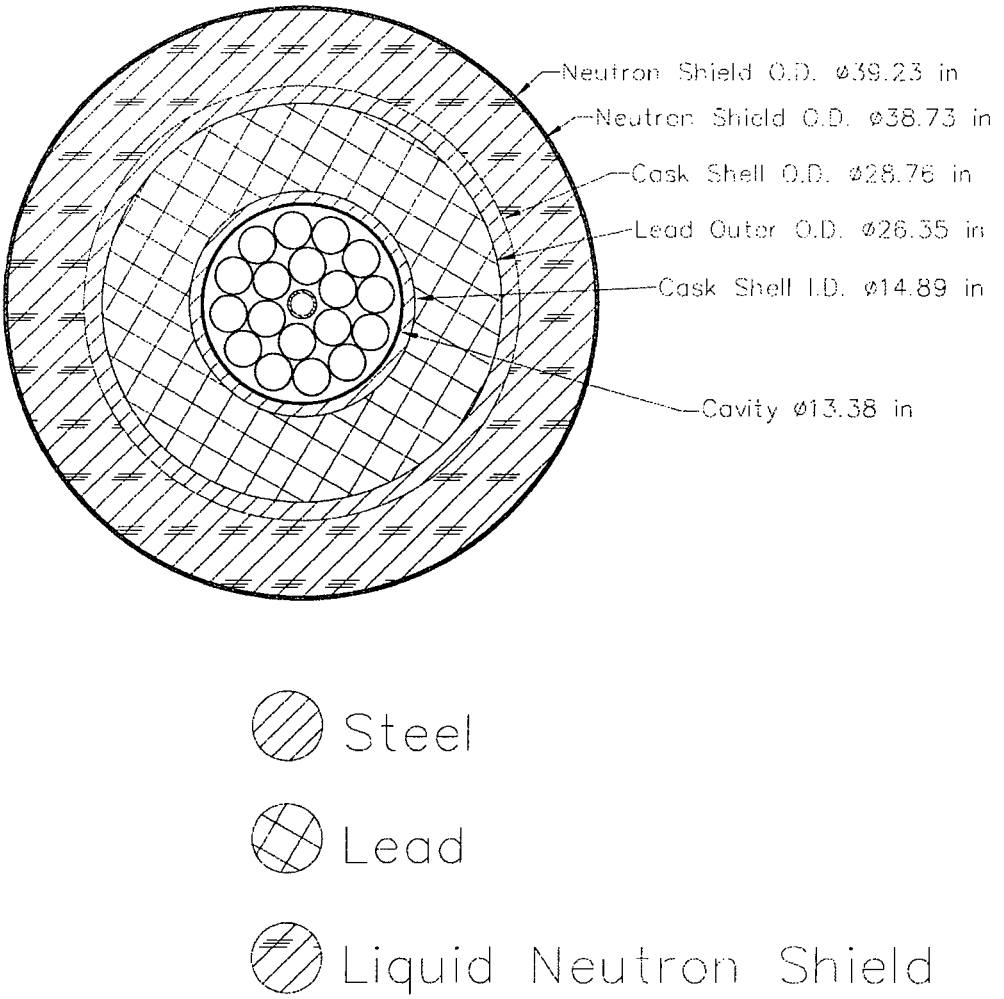


Table 6.7.2-7 Cask Fuel Conditions for Maximum System Reactivity

	10 CFR 71.55		10 CFR 71.59	
Condition	Normal	Accident	Normal	Accident
Fuel Type	NRU HEU	NRU HEU	NRU HEU	NRU HEU
Fuel Enrichment	94wt% ^{235}U	94wt% ^{235}U	94wt% ^{235}U	94wt% ^{235}U
Fuel Condition	Broken Rods	Broken Rods	Broken Rods	Broken Rods
Cask/Array	Single Cask	Single Cask	Infinite Array	Single Cask
Neutron Reflection	20 cm Water	20 cm Water	N/A[2]	20 cm Water
Neutron Shield	Yes	No	Yes[1]	No
Cask Lead / Outer Steel Shell	Yes	Yes	Yes[1]	Yes
Fuel Tube Interior Moderator	0.9982 g/cm^3	0.9982 g/cm^3	0.9982 g/cm^3	0.9982 g/cm^3
Fuel Tube Exterior Moderator	0.9982 g/cm^3	0.9982 g/cm^3	0.9982 g/cm^3	0.9982 g/cm^3
Cask Exterior Moderator	0.9982 g/cm^3	0.9982 g/cm^3	N/A[2]	0.9982 g/cm^3

Notes:

1.) Section 6.7.2.7 demonstrates that removing cask material outside the containment boundary (cask inner shell) reduces system reactivity.
2.) MCNP reflective boundary condition is applied to the cask surface.