NAC INTERNATIONAL

Atlanta Corporate Headquarters
3930 East Jones Bridge Road, Suite 200
Norcross, GA 30092
Phone 770-447-1144
Fax 770-447-1797
www.nacintl.com

September 22, 2009

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

Attn.: Document Control Desk

Subject: Submittal of NAC International Responses to NRC Request for Additional Information for Review of Amendment No. 6 to Certificate of Compliance No. 1025 for the NAC-MPC Storage System Requesting Approval to Incorporate La Crosse Boiling Water Reactor (LACBWR) Spent Fuel Assemblies as Approved Contents

Docket No. 72-1025, TAC No.L24303

References:

1. Certificate of Compliance No. 1025, Amendment 5, for the NAC-MPC System, U.S. Nuclear Regulatory Commission (NRC), July 24, 2007
2. Final Safety Analysis Report (FSAR) for the NAC-MPC System, Revision 7, NAC International, January 2008
3. Request for an Amendment of Certificate of Compliance (CoC) No. 1025 for the NAC International Multi-Purpose Canister (NAC-MPC) System to Incorporate Dairyland Power Cooperative La Crosse Boiling Water Reactor Spent Fuel as Approved Contents, NAC International, January 16, 2009, as Supplemented April 1 and 30, 2009
4. Request for Additional Information for Review of Amendment No. 6 to Certificate of Compliance No. 1025 for the NAC-MPC Storage System Requesting Approval to Incorporate La Crosse Boiling Water Reactor (LACBWR) Spent Fuel Assemblies as Approved Contents, NRC, June 25, 2009

NAC International (NAC) herewith submits its response to Reference 4. NAC has prepared the responses to the RAI questions relative to Reference 3, along with the corresponding MPC-LACBWR Safety Analysis Report (SAR) amendment changed pages.

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 MPC-LACBWR Revision 09C pages, and the included List of Effective Pages identifies the current revision level of all pages in the Reference 2 FSAR. Attachment 1 to this transmittal letter contains a complete listing of all changes requested in response to Reference 4.

In order to better facilitate the review process, NAC is providing the Revision 09C changed pages with corresponding backing pages. Consequently, the backing pages are identified as MPC-LACBWR Revision 08A, 09A or 09B as appropriate and have revision bars identical to those in the specific Revision.

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All changes requested by the responses to the RAI questions are included on Revision 09C changed pages and are marked with revision bars in the margin. In accordance with NAC's administrative practices, upon final acceptance of this application, the MPC-LACBWR Revision 08A, 09A, 09B and 09C changed pages will be reformatted and incorporated into the next revision of the NAC-MPC FSAR.

This submittal consists of eight copies of the following documents:

1. This transmittal letter and its attachment (List of Changes)
2. Reference 4 RAI questions with the NAC responses presented in standard NAC RAI response format,
3. SAR amendment MPC-LACBWR Revision 09C changed pages
4. Corrected proposed Technical Specifications, Appendix B, Page B3-8.

During the RAI response effort, NAC personnel identified that the design basis earthquake horizontal acceleration at the top of the ISFSI pad [Reference 3, Chapter 12, Section B 3.4 Site Specific Parameters and Analyses, Item 3. a)] was not stated for the MPC-LACBWR application. This has been corrected by inserting a MPC-LACBWR specific item (new Item 3.b) and the revised Page 12.B-34 has been included in the MPC-LACBWR amendment changed pages. The corresponding Technical Specification Page B3-8 is also enclosed. This change was necessary in order to synchronize the Technical Specifications with the rest of Reference 3.

Approval of this amendment to Reference 1 and the issuance of the draft CoC/Safety Evaluation Report are requested by November 30, 2009, to support Dairyland Power Cooperative's fuel loading schedule planned for 2010. Applying the Direct Final Rulemaking process, the estimated/ desired Direct Final Rule effective date is June 1, 2010.

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

Attachment 1: List of Changes, MPC-LACBWR Amendment, Revision 09C
Enclosures:
1. NAC Response to US NRC RAI dated June 25, 2009
2. NAC-MPC LACBWR Changed Pages (Revision 09C)
3. Proposed NAC-MPC CoC/Technical Specifications – Page B3-8

Attachment 1

List of Changes

MPC-LACBWR Amendment, Revision 09C

SAR Changed Pages Resulting from RAI Responses

List of Changes, MPC-LACBWR Amendment, Revision 09C

Note: The List of Effective Pages and the Chapter Table of Contents/List of Figures/List of Tables were revised to incorporate the following changes.

Appendix 1.A

- Page 1.A-2, Table 1.A-1, definition of LACBWR Damaged Fuel Assembly, item 5 – changed "intact fuel bundle" to "INTACT FUEL ASSEMBLY" and added "individual fuel rods"
- Page 1.A-3, Table 1.A-1, definition of Fuel Debris – added "(may be placed into a handling container)"
- Page 1.A-5, Table 1.A-1 – added definition for Handling Container

Appendix 2.A

- Page 2.A.1-3, Section 2.A.1.3, 2nd paragraph, 1st sentence – changed "fuel debris handling canister" to "fuel debris handling container"
- Page 2.A.1-5, Table 2.A.1-2 – changed the units of measurement for Fuel Rod Pitch, Clad Thickness and Displacement Rod Diameter from "(cm)" to "(in)"

Chapter 3

- Page 3.1-8 – added new Section 3.1.1.3 titled "Description of the MPC-LACBWR Storage System" in response to RAI 3.1

Appendix 3.A

- Page 3.A.4-16, Figure 3.A.4.4.1-2 – revised by adding Note 1 in response to RAI 3.3

Chapter 4

- Page 4.1-1, Section 4.0, 2nd paragraph, 1st sentence – revised to clarify fuel assembly heat load

List of Changes, MPC-LACBWR Amendment, Revision 09C (cont'd)

Appendix 4.A

- Page 4.A.1-1, Section 4.A.1, 2nd paragraph – added 2nd sentence in response to RAI 4.2
- Page 4.A.3-1, Section 4.A.3, 2nd paragraph – deleted 2nd sentence and added new sentences 2-6 in response to RAI 4.2
- Page 4.A.3-4, Table 4.A.3-3 – revised throughout in response to RAI 4.2
- Page 4.A.3-5, Section 4.A.3.1, 3rd paragraph – added sentences 2-9 in response to RAI 4.2
- Page 4.A.3-11, Figure 4.A.3-1 – added "not in the DFC" to the title of figure
- Page 4.A.3-15, Section 4.A.3.1.2, 3rd paragraph – revised 2nd sentence & added 3rd & 4th sentences in response to RAI 4.2
- Page 4.A.3-29, Tables 4.A.3-4 & 4.A.3-5 – revised Maximum Temperature column of both tables using the compacted fuel properties
- Page 4.A.3-30, Table 4.A.3-6 – revised the Maximum Temperature column using the compacted fuel properties
- Page 4.A.3-31, Section 4.A.3.5, 1st paragraph, 2nd sentence – changed "release of 30 percent of the fission gas" to "release of 70 percent of the fission gas"; 2nd paragraph, last sentence – changed "releasing 30 percent of their total fission gas" to "releasing 70 percent of their total fission gas"
- Page 4.A.3-34, Canister Pressure – revised two equations at bottom of page
- Page 4.A.3-35 – revised equation

Appendix 5.A

- Page 5.A.1-2, 1st, 2nd, 3rd & 4th full paragraphs – revised throughout to address bounding payload for MPC-LACBWR spent fuel assemblies; last paragraph, 1st sentence – added "Release 1.30" to show specific version of MCNP used
- Page 5.A.1-5 – added new Figure 5.A.1-1 titled "MPC-LACBWR Loading Pattern"
- Page 5.A.3-2 – added new paragraph to justify that the homogenized LACBWR analysis is conservative (RAI 5.3)
- Page 5.A.4-1, Section 5.A.4.1, 2nd paragraph – added new sentences 3rd & 4th sentences to provide information on cross-sections used in the analysis

Appendix 6.A

- Page 6.A.1-1, Section 6.A.1, 2nd paragraph – replaced last sentence with two new sentences for clarification of optimum moderator density

List of Changes, MPC-LACBWR Amendment, Revision 09C (cont'd)

Appendix 6.A (cont'd)

- Page 6.A.1-2, last line of last paragraph before table – changed "Section 6.A.4.4" to "Section 6.A.3.4" to reflect section renumbering
- Page 6.A.1-3 – added new paragraphs 1-4 to provide additional discussion on the maximum reactivity configuration
- Page 6.A.3-2, Section 6.A.3.1.1, last paragraph – added new last sentence in response to RAI 6.6
- Page 6.A.3-19, Table 6.A.3-9, last column, next to last line – changed "9.29E-02" to "9.30E-03" to correct editorial error
- Page 6.A.4-1, Section 6.A.4.1, 1st paragraph – changed "Section 6.A.4.4" to "Section 6.A.3.4" to reflect renumbering; added two new last sentences in response to RAI 6.7
- Pages 6.A.4-2 & 6.A.4-3, section titled "Fabrication Tolerances" – added new 3rd sentence
- Page 6.A.4-3, section titled "Component Shift" – added new last sentence; "Damaged Fuel Evaluations" section number changed from "6.A.4.3" to "6.A.4.2.2"
- Page 6.A.4-4, 1st partial paragraph – added new last sentence in response to RAI 6.2 b.; section titled "Fuel/Water Mixture" – added new paragraph in response to RAI 6.2 c.
- Page 6.A.4-5 – "Storage Cask Evaluations" section number changed from "6.A.4.3.1" to "6.A.4.2.3"; "Criticality Results" section number changed from "6.A.4.4" to "6.A.4.3"; "Undamaged Fuel Maximum Reactivity Basket Confirmation" section number changed from "6.A.4.4.1" to "6.A.4.3.1"
- Page 6.A.4-7 – "Damaged Fuel Can (DFC) Evaluations" section number changed from "6.A.4.4.2" to "6.A.4.3.2"
- Page 6.A.4-8, 1st partial paragraph – new next-to-last sentence added for clarity
- Page 6.A.4-9 – "Storage Cask Evaluations" section number changed from "6.A.4.4.3" to "6.A.4.3.3"; "Allowed Fuel Specifications and Comparison to USL Range of Applicability" section number changed from "6.A.4.4.4" to "6.A.4.3.4"
- Page 6.A.4-21, Table 6.A.4-8 – added rows to model a larger number of missing fuel rods
- Page 6.A.4-22, Table 6.A.4-9 – added rows to model a larger number of missing fuel rods
- Page 6.A.4-23, Table 6.A.4-10 – corrected two line items under Mix. Height column; line 2 from "0.23" to "0.25" and line 8 from "0.9" to "0.8"
- Pages 6.A.5-4 & 6.A.5-5, Section 6.A.5.3, 3rd paragraph – added new 2nd sentence to show correlation between case number listed and the experiment identification

Appendix 8.A

- Page 8.A.1-6, items 44 d & 44 e. – revised hydrostatic pressure test acceptance criteria

List of Changes, MPC-LACBWR Amendment, Revision 09C (cont'd)

Appendix 9.A

- Page 9.A.2-1, Section 9.A.2.1, 2nd sentence – changed "of 10 minutes" to "10-minute hold period"; added two new sentences to the end of the paragraph to clarify hydrostatic pressure test acceptance criteria
- Page 9.A.2-5, Section 9.A.2.7.2, 1st paragraph – added new last sentence to clarify neutron absorber test methods

Appendix 10.A

- Page 10.A.3-4, Table 10.A.3-1, steps 7 & 8 – revised column showing average dose rate (mrem/hr)

Appendix 11.A

- Page 11.A.1-3, 1st full paragraph, 1st sentence – changed "475°F" to "475°K" [the value should have been listed as Kelvin (K)]; 2nd sentence – changed "10.1 psig" to "11.4 psig"
- Page 11.A.2-1, Section 11.A.2.1.1, 1st paragraph, 2nd sentence – changed "30%" to "70%"; 3rd sentence – changed "680°F" to "400°F"
- Page 11.A.2-2 – revised 1st equation and all numbers in the last paragraph of that section; Section 11.A.2.1.2 – revised the accident pressure to 32.1 psig, along with the maximum stresses and the minimum margin of safety
- Page 11.A.2-2, Section 11.A.2.1.2, 1st paragraph, 2nd sentence – changed "20 psig" to "32.1 psig"; 2nd paragraph, 1st sentence – changed "20 psig" to "32.1 psig"; 2nd sentence – changed "20 psig" to "32.1 psig" & "11.11 ksi and 44.99 ksi" to "15.19 ksi and 61.88 ksi"; 3rd sentence – changed "46.23 ksi and 66.04 ksi" to "45.65 ksi and 65.2 ksi"; last sentence – changed margin of safety from "0.47" to "+0.05"
- Page 11.A.2-26, subsection titled "Support Disk and Weldment Stress Analysis," 1st paragraph – added two new last sentences to provide justification for employing a different mesh in the evaluation of the basket
- Page 11.A.3-1, Section 11.A.3 – Sections 11.A.3.1 and 11.A.3.2 were deleted as requested by RAI 3.5; Section 11.A.3 is revised accordingly
- Page 11.A.4-1, Section 11.A.4, subsection titled "End Drop Evaluation," 3rd paragraph – revised throughout in response to RAI 8.4 and RAI 8.5
- Page 11.A.6-1, Section 11.A.6 – added new reference [A4]

Appendix 12.A

- Page 12.A-5, Definitions – added definition for HANDLING CONTAINER
- Page 12.A-6, Definitions, definition of LACBWR DAMAGED FUEL ASSEMBLY, item 5 – changed "intact fuel bundle" to "INTACT FUEL ASSEMBLY" and added "individual fuel rods"
- Page 12.A-9, Definitions, definition of STRUCTURAL DAMAGE – added "Yankee and CY" to beginning last sentence

Additional changes to address the seismic acceleration issue for the MPC-LACBWR system are contained in the following pages:

Appendix 12.B

- Page 12.B-34, Section B 3.4, Item 3 – a) is revised by adding "For Yankee-MPC and CY-MPC,"; b) is new to address MPC-LACBWR; former b) is now c)

Proposed Technical Specifications, CoC No. 1025 Amendment 6

- Page B3-8, Section B 3.4, Item 3 – a) is revised by adding "For Yankee-MPC and CY-MPC,"; b) is new to address MPC-LACBWR; former b) is now c)

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

JUNE 25, 2009

FOR REVIEW OF AMENDMENT NO. 6 TO CERTIFICATE OF COMPLIANCE NO. 1025 FOR THE NAC-MPC STORAGE SYSTEM REQUESTING APPROVAL TO INCORPORATE LA CROSSE BOILING WATER REACTOR (MPC-LACBWR) SPENT FUEL ASSEMBLIES AS APPROVED CONTENTS

(TAC NO. L24303, DOCKET NO. 72-1025)

SEPTEMBER 22, 2009

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NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

3.0 Structural Design

3.1 Section 3.1.1 Discussion. Revise the SAR text by providing also a description on LACBWR to establish links to Appendix 3.A, as appropriate, to mirror the evaluations performed on CY-MPC and Yankee-MPC.

The subject description is incomplete. It is required for compliance with 10 CFR 72.244.

NAC International Response

Section 3.1.1 currently contains Section 3.1.1.1 for the Yankee-MPC design and Section 3.1.1.2 for the CY-MPC design. In these two sections, an overview of the design is presented. A similar description of the MPC-LACBWR system is contained in Section 3.A.1.1. Section 3.1.1 is revised to add Section 3.1.1.3; this new section provides direction to reference Section 3.A.1.1 for the description of the MPC-LACBWR system.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

3.A Structural Design for the MPC-LACBWR Storage System

3.2 3.A.4.4.1.1 MPC-LACBWR Canister Analysis. Verify that the finite analysis model depicted in SAR Figure 3.A.4.4.1-1 is used for the canister structural evaluation.

The element discretization scheme at the closure lid-to-shell interface region is seen markedly different from that displayed in Figure 11.A.2.12-2. It's unclear whether a relevant finite element model and its analysis are properly reported in the SAR for the canister structural analysis. This information is required for compliance with 10 CFR 72.236(b).

NAC International Response

The finite element model shown in Figure 3.A.4.4.1-1 was employed in the structural analysis of the canister. The canister basket model shown in Figure 11.A.2.12-2 was used in the structural analysis of the basket. Both canister models employ the same dimensions and material properties. The canister was included with the model of the basket to allow the canister shell displacements to be included as a boundary condition in the structural analysis of the basket. Since the canister was included only for the purpose of accurately simulating the action of the canister against the basket support disk, it was not necessary to employ the same detail in the canister finite element model that was used for the canister structural analysis. The element discretization in the lid and weld does not affect the restraint imposed on the top of the canister shell by the lid.

Revised SAR amendment text is provided in Section 11.A.2.12.2.2, page 11.A.2-26, to include the justification for employing a different mesh in the evaluation of the basket.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

3.A Structural Design for the MPC-LACBWR Storage System

3.3 3.A.4.4.1.1 MPC-LACBWR Canister Analysis. Clarify SAR-Figure 3.A.4.4.1-2 , Detail A, by providing appropriate section annotations indicating the stress reporting summary for the upper left corner for the inner ring.

Details are lacking for a sufficient description of the modeling approach to evaluating stress performance of the canister closure redundant sealing configuration. This information is required for compliance with 10 CFR 72.236 (b).

NAC International Response

Detail A of Figure 3.A.4.4.1-2 defines the stress section locations used in the evaluation of the canister lid and closure ring welds to the canister shell. Sections 9, 10, and 12 capture the local structural cross-sections for the ½-inch thick weld between the closure lid and the canister shell, as well as the canister shell adjacent to the weld. Sections 11 and 13 represent the structural cross-sections through the weld at the closure ring and the canister shell, as well as through the canister shell local to the closure ring weld. The closure ring to the canister shell weld represents the more bounding load path than the inner closure ring to the closure lid weld. The inner closure ring weld is insulated from external loading on the canister shell by the closure ring; in comparison, the weld for the outer closure ring is integral to the canister shell. Therefore, both the closure lid weld and the closure ring weld that attach to the canister shell are evaluated with the local canister through wall stress sections to perform a comprehensive structural evaluation for the canister closure.

The following comparison of stresses from the finite element analysis results for the bounding condition of the tip-over accident provides an example confirming the above discussion.

NAC International Response to RAI 3.3 (cont'd)

	Inner Weld Stress (ksi)	**Outer Weld Stress (ksi)**
P_m	23.7	28.6
P_{m+b}	26.7	32.2

A note has been added to Figure 3.A.4.4.1-2 of the SAR amendment, Revision 09C, indicating that Section 13 of the stress section locations is bounding.

Revised SAR amendment Figure 3.A.4.4.1-2 is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

11.A Accident Analysis for the MPC-LACBWR Storage System

3.4 11.A.2.12.2.1 Analysis of the MPC-LACBWR Concrete Cask. Clarify the basis for selecting 10,000 psi as the bounding modulus of elasticity in the cask tip-over analysis for the LACBWR ISFSI site. Clarify the basis for the soil depth of 6 ft for the site-specific application.

It's unclear how the soil modulus of elasticity is selected for the ISFSI site, which could be far away from the reactor site and render the soil boring data at the reactor premises inapplicable to the tip-over analysis. This information is required for compliance with 10 CFR 72.236 (b).

NAC International Response

The acceptability of the soil at the ISFSI pad (not at the reactor containment building site itself) will be evaluated by the 72.212 evaluation process. This evaluation will take into account the requirement of the 10,000 psi soil modulus, as well as the minimum soil depth of 5 feet associated with the soil modulus.

Note that the MPC-LACBWR soil depth and modulus are defined in the Technical Specifications, Appendix B, Section B 3.4, Site Specific Parameters and Analyses, Item 6. It is noted that the LACBWR ISFSI site-specific soil data identifies the soil modulus in the range of 9,000 psi and that the soil depth is larger that the 5-foot depth considered in the analysis. Therefore, the tip-over analysis presented in the SAR amendment is conservative.

Based on the content of this response, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

11.A Accident Analysis for the MPC-LACBWR Storage System

3.5 11.A.3.1 Basket Support Disk Evaluation for the 30-ft Side Drop. Remove the subject evaluation from the SAR.

The side drop evaluation for a 10 CFR 71.73(c) requirement is not relevant to the amendment that is on request. This is required for compliance with 10 CFR 72.244.

NAC International Response

Section 11.A.3.1 is removed from the MPC-LACBWR amendment request (Revision 08A). Section 11.A.3 is revised to state that analyses have been completed to verify that the MPC-LACBWR amendment design satisfies 10 CFR 71 loading conditions in response to 10 CFR 72.236 (m) requirements.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL

11.A Accident Analysis for the MPC-LACBWR Storage System

3.6 11.A.3.2 Basket Support Disk Evaluation for the 30-ft Top End Drop. Remove the subject evaluation from the SAR.

The top end drop evaluation for a 10 CFR 71.73(c) requirement is not relevant to the amendment that is on request. This is required for compliance with 10 CFR 72.244.

NAC International Response

Similar to the response to RAI 3.5, Section 11.A.3.2 is removed from the MPC-LACBWR amendment request (Revision 08A). Section 11.A.3 is revised to state that analyses have been completed to verify that the MPC-LACBWR amendment design satisfies 10 CFR 71 loading conditions in response to 10 CFR 72.236 (m) requirements.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL

4.1 Correct the maximum heat load per assembly of the Yankee-MPC from the stated value of 0.102 kW in Section 4.0, 2nd paragraph, to reflect the value derived from 36 assemblies with a total canister heat load of 12.5 kW and as shown in Table 3.1 "Yankee-MPC Design Basis Fuel Characteristics."

This information is required for compliance with 10 CFR 72.236 (f).

NAC International Response

The stated fuel assembly value of 0.102 kW is associated with the reconfigured fuel assembly as indicated in the 2nd paragraph of Section 4.0 of the NAC-MPC FSAR, Revision 7. The maximum fuel assembly heat load of 0.347 kW/assembly is shown in the NAC-MPC FSAR, Revision 7, in Section 4.4.1.6 on page 4.4-30 and in Section 4.4.3.2 on page 4.4-33. To improve the clarity of Section 4.0, the 2nd paragraph on page 4.1-1 has been revised to show (12.5 kW/36 assemblies = 0.347 kW per fuel assembly), along with the 0.102 kW for the reconfigured fuel assembly.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL

4.2 Evaluate and explain in the SAR the thermal effect of compaction of all 32 damaged fuel cans' fuel to 50% of their original length. Since damaged fuel has no quantified structural integrity, its ultimate failure and reconfiguration must be presumed and evaluated.

This information is required for compliance with 10 CFR 72.236 (f).

NAC International Response

The purpose of the 32 damaged fuel cans (DFC) is to contain 32 Allis Chalmers (AC) assemblies in each canister. As indicated, these damaged fuel assemblies are described as having no quantified structural integrity. As a result, segments of the fuel rods could be repositioned during handling operations.

To determine the length over which the fuel assembly is distributed, the physical volume (V) of the fuel assembly is first computed using fuel weight data and material densities. One hundred percent (100%) of the fuel volume (V) corresponds to a length (L) determined by V/A, where A is the cross-section of the fuel can. A condition of 50% compaction is achieved when the modeled length (L_m) of the compacted fuel region is computed by L/(0.5). The 66.2W per assembly is distributed over the modeled length (L_m) for each of the 32 DFCs. The center of the compacted fuel region is positioned at the midplane of the basket and represents a system bounding thermal configuration since the maximum temperature of the undamaged fuel occurs at the basket axial midplane. The conduction properties of the compacted region were conservatively taken to be the thermal properties of the undamaged fuel, since the undamaged fuel has lower conductance due to lack of contact of the fuel segments. Outside the compacted region, the volume inside the DFC was considered to be helium gas or water without any consideration of radiation across the DFC region. These thermal analyses showed that the maximum clad temperature was increased by 6°F, resulting in the normal condition fuel clad temperature of 449°F (as compared to the previous value of 443°F). The margin of the maximum fuel clad temperature (to an allowable of 806°F) remains large.

NAC International Response to RAI 4.2 (cont'd)

Although the influence of the compacted damaged fuel in the DFCs is shown to have no impact on system safety, Section 4.A.1 and Section 4.A.3 of the SAR amendment are revised to use the compacted DFC as the bounding configuration when defining the maximum temperatures of the system for normal, off-normal and accident conditions.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL

4.3 Explain why the calculated maximum temperature for the neutron shield of 347°F exceeds the temperature limit of 300°F in Table 4.A.3-5 "MPC-LACBWR Maximum Component Temperatures for the Transfer Condition – Helium and Vacuum in Canister." Also correct the SAR to show a neutron shield maximum temperature of less than the temperature limit of 300°F.

Since the heat load of the LACBWR is approximately a factor of 4 less than the CY-MPC, it is not clear why the neutron shield temperature would be exceeded for the LACBWR and not for the CY-MPC as shown in Tables 4.5.3-5 & -6 for transfer operations.

This information is required for compliance with 10 CFR 72.236 (f).

NAC International Response

The maximum temperatures of the MPC-LACBWR system in the transfer cask were revised in a supplemental submittal (MPC-LACBWR Revision 09A dated March 2009), which included a revised Table 4.A.3-5 with data confirming that the neutron shield temperature is significantly lower than the 300°F allowable temperature.

In response to RAI 4.2, the transfer cask temperatures in Section 4.A.3.1.2 and the temperatures in Tables 4.A.3-4, 4.A.3-5 and Table 4.A.3-6 have also been revised using the compacted fuel properties described in RAI 4.2.

Based on the above, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL

4.4 Justify the maximum internal pressure of 20 psig used in Section 11.A.2.1.2 "MPC-LACBWR Canister Maximum Stress Due to Internal Pressure."

When using the ideal gas law and the methodology from Section 4.A.3.5 "Maximum MPC-LACBWR Internal Pressure for Normal Conditions" to determine the maximum canister internal pressure under accident conditions, a value of 30 psig is obtained rather than the utilized value of 20 psig. The increase is due to 100% of the spent fuel tubes failing with 30% of the fission gas escaping into the canister and the accident temperature increasing to 680°F.

Pressure = (256.3 moles)(0.0821 atm-liter/mole-K)(633K)/(4371 liters) = 3.04 atm or 30 psig.

This value of 30 psig also exceeds the internal pressure loading used in the stress calculations in Chapter 3, as discussed in the second paragraph of this section.

This information is required to demonstrate compliance with 10 CFR 72.236 (l).

NAC International Response

The gas temperature listed in the SAR, MPC-LACBWR Revision 08A, was quoted incorrectly. The gas temperature applied in the pressure calculation, which employed the correct value, is 400°F, versus a 387°F temperature calculated in the thermal analysis for the maximum temperature condition. This temperature results in the 20 psig pressure quoted in the SAR. Note that as a response to RAI 8.2, the gas release fraction was raised resulting in increased pressure (32.1 psig). Section 11.A.2.1.1 is revised accordingly.

While reviewing all pressure SAR sections to ensure that a similar error was not made in the normal and off-normal condition descriptions, it was noted that the off-normal pressure analysis temperature was listed with the unit °F. Value should have been listed as Kelvin (K) and is corrected in the SAR on page 11.A.1-3.

NAC International Response to RAI 4.4 (cont'd)

The stress evaluation of the canister is revised for the accident pressure of 32.1 psig and the maximum stresses and minimum margin of safety are updated in Section 11.A.2.1.2, page 11.A.2-2.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.1 Provide additional information describing the loading pattern used to perform the shielding analyses.

Justify the fuel loading pattern chosen for the shielding analysis is conservative and bounds all possible loading patterns. The staff notes from Figure B2-3 of the technical specifications that there are multiple possible loading patterns.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

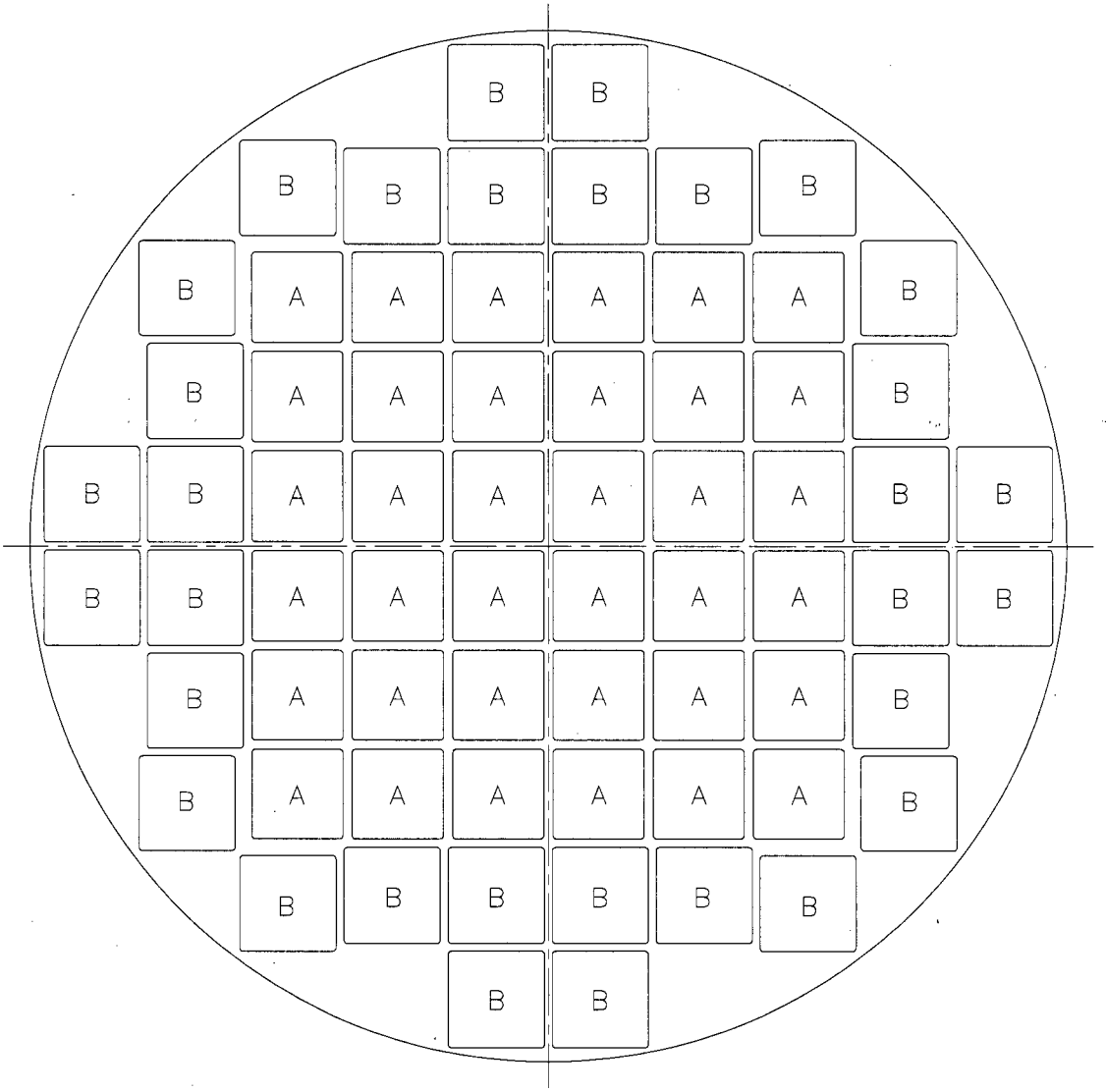
NAC International Response

Shielding evaluations are based on Allis Chalmers and Exxon fuel assembly groupings, but do not differentiate in the enrichment levels of the Allis Chalmers fuel. A bounding fuel enrichment is applied in the shielding analysis as discussed in Section 5.A.2 (specifying a minimum enrichment). The Allis Chalmers enrichment grouping and restrictions listed in proposed Technical Specification Figure B2-3 are the result of the criticality evaluations/limitations on the system. From a shielding perspective, Figure B2-3 may be simplified to Figure 5.1-A as shown on the following page. This figure is added to SAR Section 5.A.1 as Figure 5.A.1-1 to form the basis of the bounding payload discussion in paragraphs 6 and 7 of Section 5.A.1. These paragraphs describe the configurations evaluated and justify that the requested configuration in Figure B2-3 is acceptable.

The text in Section 5.A.1 is revised to provide the additional information on loading patterns requested. Revised SAR amendment text is provided.

NAC International Response to RAI 5.1 (cont'd)

Figure 5.1-A MPC-LACBWR Shielding Evaluation Loading Pattern



Slot A: Undamaged Exxon fuel

Slot B: Undamaged or damaged Exxon or Allis Chalmers fuel

Fuel Type	Max. Burnup [MWd/MTU]	Min. Initial Enrichment [wt % ^{235}U]	Min. Cool Time [years]
Allis Chalmers	22,000	3.6	28
Exxon Nuclear Company	21,000	3.6	23

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.2 Provide additional information about the use of continuous energy cross sections with grouped energy source term.

You use a source term with grouped energy structure for photons and neutrons and input this into a code that has continuous energy cross sections. Did you run MCNP with grouped cross sections? If not, provide information explaining what impact (if any) there is on calculated dose using a discrete energy source term propagated through the cask using continuous energy cross sections.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

NAC International Response

NAC calculations were performed using the continuous MCNP cross-section sets. As a response to RAI 5.5, information on the cross-sections used in the analysis is added to SAR Section 5.A.4.1.

Using continuous energy transport in MCNP will always provide more accurate results versus using multigroup cross-sections with MCNP, regardless of the method of the generation of the source term. Generating group cross-sections and tracking particle transport in a group structure clearly requires a reduction in the information available on the particles being transported. Providing source terms to MCNP in a group structure is a standard method of inputting sources, even if those sources are calculated with other continuous energy calculations (in the LACBWR case, grouped sources are calculated with SCALE SAS2H). The continuous energy cross-sections for MCNP undergo rigorous testing and validation, while there are no multigroup cross-section libraries that undergo the same level of testing and validation.

Based on the content of this response, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.3 Provide additional information regarding the use of homogenized fuel source term when calculating neutron dose.

The staff understands that the fuel source term is homogenized. This may be non-conservative when calculating the neutron dose rate for the flooded condition because homogenized fuel and moderator can show an artificial increase in resonance absorption in U-238. How did you account for this in your neutron dose rate calculations for the flooded condition?

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

NAC International Response

It is understood that homogenizing the fuel assembly affects neutron absorption behavior in the fuel. To address the NRC's concern, NAC revised the wet, flooded calculation to include the detailed basket/fuel configuration used in the criticality analysis. The model contains the full fuel rod lattice, while simplifying the absorber configuration to limit the number of tube configurations (modeled configuration reduces the number of exterior neutron absorber panels). This model accounts for subcritical multiplication within the MCNP run rather than applying the multiplication factor to the neutron source term magnitude, as documented in Section 5.A.2.2 of the SAR. The model also considers a single source term in all 68 fuel assemblies to simplify the analysis. The results of the heterogeneous to homogeneous fuel model comparison are documented below and clearly demonstrate the conservative nature of the NAC calculation. The results demonstrate that modeling a heterogeneous fuel rod array with MCNP-calculated subcritical multiplication produces dose rates more than 80% lower than the homogeneous model with a neutron source subcritical multiplication factor.

		Dose Rate (mrem/hr)	
Source	**Component**	**Homogeneous**	**Heterogeneous**
Neutron	Maximum	22.6	3.8
	Average	7.8	1.3
N-Gamma	Maximum	0.29	0.06
	Average	0.11	0.02

NAC International Response to RAI 5.3 (cont'd)

Note that homogenizing the fuel region is identical in method to currently certified NAC-MPC, NAC-UMS® and MAGNASTOR® calculations and represents a typical industry approach. Large capacity cask dose rates are controlled by thick shields. Analysis effort is, therefore, driven by optimizing and concentrating on particle transport in the shield, not the fuel region. Interior fuel material is effectively shielded by exterior material, and doses are primarily produced by exterior facing source materials. As neutrons tend to leak from these regions, the phenomena in question is not considered significant to the transfer cask dose solution.

The analysis approach of homogenizing the fuel region is consistent with the other NAC-MPC evaluations, and the approach used in the MPC-LACBWR amendment is demonstrated to be conservative in this response. In addition, the SAR amendment text in Section 5.A.3.1 is modified to include the justification that the homogenized MPC-LACBWR analysis is conservative.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.4 Provide additional information on the confirmation of boron content in NS-4-FR.

Provide additional information discussing how you determine that the NS-4-FR material has the appropriate boron content and uniformity. Discuss any testing and acceptance criteria.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

NAC International Response

NAC orders raw materials used in the borated neutron shield material (NS-4-FR) formulation as commercial grade materials. Prior to assembling the kits for use in mixing and installation of NS-4-FR into the actual component, materials are tested to confirm that the NS-4-FR minimum requirements are met, i.e., density, hydrogen and boron carbide content. This is accomplished by assembling Test Kits for each combination of lots used in the preparation of actual Production Kits. Each Test Kit is then mixed in accordance with approved procedures. Samples are extracted from the Test Kit mix. One sample is used to determine the dry density, and a second is sent to the laboratory for hydrogen and boron content testing.

NAC applies this procedure to all NS-4-FR applications to ensure that the NS-4-FR material is appropriately installed and contains a uniform material composition. This procedure was tested in the year 2000 by pouring a mock-up neutron shield characteristic of NAC transfer casks. The mock-up was then sectioned with samples from various locations tested by an independent laboratory. This testing confirmed that the procedures used in the NS-4-FR pour result in a uniform hydrogen and boron distribution (hydrogen content being the key characteristic of the neutron shield).

The NS-4-FR in the proposed MPC-LACBWR storage system has been deployed at a number of NAC-UMS® and NAC-MPC ISFSI installations, including Yankee Rowe, Connecticut Yankee, Maine Yankee, Duke's McGuire and Catawba Nuclear Stations, and APS' Palo Verde Nuclear

NAC International Response to RAI 5.4 (cont'd)

Generating Station. There have been no dose observations indicating peaks resulting from postulated nonuniform boron distributions in the neutron shield. The transfer cask used in the Yankee Rowe loading operations is the same hardware that will be used for the MPC-LACBWR cask loadings.

Note that the NAC-LACBWR system's use of neutron shielding is limited to the transfer cask radial shield and that the LACBWR fuel is low mass (0.12 MTU per assembly), low burnup (~20 GWd/MTU) and has a long cool time (>20 years). Therefore, the LACBWR fuel does not have a significant neutron source (~10^6 n/sec versus ~10^8 for a typical LWR current generation assembly). Absorption of the low neutron flux within the shield is, therefore, not expected to produce a significant secondary gamma source. Furthermore, the transfer cask contains a 1.25-inch exterior shell that will attenuate/absorb typical secondary gamma rays. Calculated n-gamma dose rates with the minor boron content of the neutron shield (0.6 wt % B_4C typical) are less than 5 mrem/hr for this system. A hypothetical removal of boron from the shield is not expected to significantly increase dose rates.

Based on the content of this response, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.5 Provide additional information on the code used to perform the shielding calculations.

Provide the specific version of MCNP used to perform the shielding calculations as well as the cross section set used.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

NAC International Response

MCNP is referenced as Reference A1. Reference A1 lists MCNP5, with a manual release of April 24, 2003. This corresponds to MCNP5 Release 1.30. Section 5.A.1 of the SAR is modified to include this information directly.

Default neutron and photon cross-sections in this release of MCNP are being used in the photon and neutron shielding analysis. Default cross-sections are invoked by omitting the .XXX extension in the sample input material cards shown in Section 5.A.6. Default neutron data libraries are various releases of the ENDF/B-V and -VI libraries (depending on element or isotope, specified data are available in various ENDF releases), and the MCNPLIB04 photoatomic data set. As a response to this RAI, cross-section information is added to Section 5.A.4.1 of the MPC-LACBWR SAR.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING

5.6 [Editorial] The units for the Fuel Rod Pitch and Displacement Rod Diameter in Table 2.A.1-1 are stated to be centimeters; however, the dimensions given appear to be in inches.

NAC International Response

The correct units on fuel rod pitch, clad thickness, and displacement rod diameter in Table 2.A.1-1 are inches. The table is revised to correct this editorial error. Correcting the units to inches makes the values consistent with the criticality fuel data listed as inches in Table 6.A.2-1 and the centimeter values in Table 5.A.2-1.

Revised SAR amendment text (Table 2.A.1-1) is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.1 Provide additional information on the assumptions used to calculate the values of k_{eff} on Page 6.A.1-2.

Provide the assumptions used in the criticality models that were used to calculate the values of k_{eff} on Page 6.A.1-2. Justify that these conditions are conservative. Include specifically information such as:

a. Loading pattern, including damaged fuel assemblies;
b. Provide additional and specific details of the "combined tolerance model";
c. Reflector assumptions (outside of the cask);
d. Clad gap flooding; and
e. Details on the flooding (i.e., which areas of the cask/assemblies were flooded with what density water).

The staff understands that sensitivity studies were performed to find the most reactive conditions for these parameters; however, the conditions used to determine the k_{eff} values on Page 6.A.1-2 are not explicitly listed.

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

Section 6.A.1 is modified to provide additional discussion on the maximum reactivity configuration. Details on this evaluation are provided in Section 6.A.4, which include the step-by-step discussion moving from nominal/centered components for undamaged fuel to damaged fuel. The outline of this analysis is presented in Section 6.A.4.1. Each of the requested items is discussed in the following section with similar text added to SAR Section 6.A.1.

Note: Section 6.A.4 was renumbered beginning with existing Section 6.A.4.3, "Damaged Fuel Evaluations," as this section should be a subsection under Section 6.A.4.2, "Fuel Loading

NAC International Response to RAI 6.1 (cont'd)

Optimization." The title of the first subsection, 6.A.4.2.1, is "Undamaged Fuel"; therefore, the "Damaged Fuel" subsection should be numbered 6.A.4.2.2. The subsequent section numbers are also revised accordingly to correct this oversight.

a. Figure 6.A.4-4 (duplicated in Section 6.A.1 as Figure 6.A.1-1), Section 6.A.4.4.2 (renumbered 6.A.4.3.2), provides the allowed loading locations for the damaged fuel. This configuration was required to assure a final $k_{eff}+2\sigma$ below the USL.

b. The combined tolerance model is discussed in Section 6.A.4.4.1 (renumbered 6.A.4.3.1) under the subheading "Fabrication Tolerances – Centered Components" in that it lists the combination of parameters resulting in a maximum reactivity configuration. This configuration is repeated in the same section under the subheading "Combined Basket Tolerance and Shift." As noted in Section 6.A.4.4.2 (renumbered 6.A.4.3.2) under the subheading "Maximum Reactivity Configuration Confirmation Studies," for the preferential flooded maximum reactivity, configuration tolerances on basket components do not affect DFC system reactivity significantly. The tolerances tend to affect neutron absorber distance and quantity between tubes. For the dry canister system, there is no moderator between tubes and, therefore, these items lose their significance.

c. Storage casks are based on infinite array calculations with transfer cask analysis based on a single cask model. An additional MCNP macro body is placed at a 20-cm distance from the transfer cask surface to allow exterior moderator density to be evaluated. The storage cask model includes a reflective boundary condition to produce an infinite cask array. This discussion is added to the model description. The table containing a summary of DFC reactivities, Table 6.A.4-15, indicates array or single cask information for each variation.

d. As discussed in Section 6.A.4.4.1 (renumbered 6.A.4.3.1) under "Fuel Type and Enrichment Variation Study," wet and dry pellet to clad configurations were evaluated, with the wet gap being bounding. Note that the maximum reactivity configuration is for damaged fuel cans with no clad in the preferentially flooded DFCs and a dry TSC (canister cavity). As there is no clad modeled in the DFC, there is no gap to flood.

NAC International Response to RAI 6.1 (cont'd)

e. Maximum reactivity detailed in Section 6.A.4.4.2 (renumbered 6.A.4.3.2), "Optimum Moderator," describes a preferentially flooded DFC (0.9982 g/cc) with dry canister (modeled at 0.0001 g/cc water) to be the most reactive flood condition.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.2 Provide additional information on the analyses performed for the damaged fuel.

Provide additional information regarding the assumptions used when modeling a damaged fuel assembly and justify that they are conservative relative to an actual damaged fuel assembly. The staff understands that evaluations were performed to find the maximum reactivity; however, the conditions used to determine the k_{eff} values on Page 6.A.1-2 are not explicitly stated. Address the following concerns related to damaged fuel evaluations:

a. What were the results of the analyses for the unclad rods? For the models that produce the results in the Table on Page 6.A.1-2, state if the damaged fuel assemblies have cladding.
b. The studies for missing rod geometries go up to the removal of 20 rods. Are there any fuel assemblies with more than 20 missing fuel rods? If so, justify that any fuel assemblies with more than 20 missing fuel rods are bounded by the analysis. In addition, justify that the pattern in which the rods were removed is conservative.
c. Provide additional information on the fuel/water mixture studies. Justify the conditions used to simulate fuel rubble are conservative or realistic. Was the density of the mixture varied with height? What was the maximum density used? Justify that it is conservative or realistic.
d. Clarify what is meant by "DFC fuel material is increased by 5% over that of an undamaged assembly" (Section 6.A.3.2.1(e)). What was increased? Density? Active fuel length?

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

The NAC responses to the listed issues are as follows:

Note: Section 6.A.4 was renumbered beginning with existing Section 6.A.4.3, "Damaged Fuel Evaluations," as this section should be a subsection under Section 6.A.4.2, "Fuel Loading

NAC International Response to RAI 6.2 (cont'd)

Optimization." The title of the first subsection, 6.A.4.2.1, is "Undamaged Fuel"; therefore, the "Damaged Fuel" subsection should be numbered 6.A.4.2.2. The subsequent section numbers are also revised accordingly to correct this oversight.

a) As a response to RAI 6.1, the text in Section 6.A.1 was modified to explicitly state that the maximum reactivity is based on unclad pellet stacks. Differences in reactivity between the clad and the unclad configuration may be seen in Section 6.A.4.4.2 (renumbered 6.A.4.3.2). Table 6.A.4-8 shows the results for clad rods at the as-manufactured pitch, and Table 6.A.4-9 shows the results for fuel rods with no clad.

b) There are no fuel assemblies with 20 rods, or more, removed in the inventory to be loaded. The analysis is designed to demonstrate reactivity effects of the potential collapse of fuel rods due to clad damage. This type of event is not expected to occur in transfer conditions, but may be postulated to result from a transport cask accident. Table 6.A.4-8 demonstrates that system reactivity for the canisters with DFCs, with fuel clad surrounding the pellet, does not change as a function of the number of fuel rods removed. The system reactivity remains statistically the same as the system with all rods in place, i.e., system reactivity is controlled by the higher reactivity center portion of the TSC that contains undamaged fuel. As a significant amount of fissile material was removed from the system without changing reactivity, an engineering judgment was made that further analysis would not result in an reactivity increase. To confirm this judgment, additional data with more missing fuel rods were added to the table. Only when half the fuel was removed from the DFCs was a drop in reactivity indicated.

   The missing rod pattern was chosen to increase potential reactivity by not removing adjacent fuel rods, i.e., not generating large moderator volumes that would result in exceeding the H/U peak and result in excess water neutron absorption; and by initially removing fuel rods near the center of the assembly, i.e., away from the neutron absorbers and in the area where increases in the moderator ratio are most likely to increase system reactivity. As the analysis demonstrated that rod removal did not affect system reactivity in either a positive or negative manner, additional studies were not considered to further the assurance that maximum reactivity system configuration was achieved. Similar text to this response is added to the SAR amendment Section 6.A.4.3 (renumbered 6.A.4.2.2), subsection titled "Clad Rods – Grid Undamaged," as a response to this RAI.

NAC International Response to RAI 6.2 (cont'd)

While not invoked by this RAI, the studies in Table 6.A.4-9 were also increased to model a larger number of missing fuel rods. This data also demonstrate that rod removal from the LACBWR assemblies does not result in a reactivity increase. While not included in the SAR, the calculation supporting the system also included missing rod studies at the maximum reactivity preferentially flooded configuration. The results of this analysis are included herein for information.

# Missing Rods	k_{eff}	σ	$k_{eff}+2\sigma$	$\Delta k/\sigma$
0	0.92680	0.00167	0.93014	--
1	0.92228	0.00150	0.92528	-2.2
4	0.92398	0.00164	0.92726	-1.2
8	0.91967	0.00148	0.92263	-3.4
12	0.90724	0.00162	0.91048	-8.4
16	0.90587	0.00156	0.90899	-9.3
20	0.89483	0.00166	0.89815	-13.6
32	0.85834	0.00143	0.86120	-31.4
50	0.80161	0.00147	0.80455	-56.4

c) The fuel mixture studies discussed in Section 6.A.4.3 (renumbered 6.A.4.2.2) and Section 6.A.4.4.2 (renumbered 6.A.4.3.2), Table 6.A.4-10, model fuel material at various material fractions, expressed as mixture height (i.e., fuel/water mixture height in DFC). At the low end fraction, the material is modeled as a solid mass at preirradiation density (10.522 g/cc or 96% theoretical), while at the high end, the fuel water mixture fills the DFC cavity. Per Table 6.A.4-10, a compacted mixture at full density occupies approximately 25% of the DFC. Therefore, the full height mixture will result in a fuel density approximately 25% of the initial case (i.e., 2.6 g/cc). This study is, therefore, an optimum moderator density study for the fuel/water mixture. Water within the mixture is full density in all cases. As shown in Table 6.A.4-10, and expected based on low enriched fuel, a compacted fuel is undermoderated. Increasing the mixture height, i.e., adding water to the matrix, increases reactivity. For the LACBWR assembly, a statistical peak is reached at approximately 80% of the canister cavity volume, with no significant change after this point. Note that this model removes any stainless steel from the DFC model. Damaged clad and fuel assembly structure

NAC International Response to RAI 6.2 (cont'd)

occupy space conservatively assumed to be available to the fuel clad mixture. As a response to this RAI, additional cases with lower than full density water in the fuel/water mixture were run and are presented as follows. As expected, there is no increase in reactivity with lower moderator density fractions in the fuel/water mixture.

Water Density in Fuel/Water Mixture	**0.9982 g/cc**	**0.9 g/cc**	**Δ to 0.9982 g/cc**	**0.8 g/cc**	**Δ to 0.9982 g/cc**
Mixture Fraction of DFC Cavity	k_{eff}	k_{eff}	$\Delta k/\sigma$	k_{eff}	$\Delta k/\sigma$
0.25	0.85118	0.85181	0.6	0.85090	0.6
0.3	0.85096	0.85165	0.6	0.85341	-0.6
0.4	0.85511	0.85515	0.0	0.85493	-2.5
0.5	0.86074	0.85563	-4.7	0.86023	-0.6
0.6	0.86393	0.86045	-3.0	0.86167	-4.0
0.7	0.86783	0.85952	-7.3	0.86306	-5.3
0.8	0.86864	0.86012	-7.5	0.86315	-3.9
0.9	0.86780	0.86122	-5.9	0.86369	-2.7
1	0.86659	0.85893	-7.1	0.86343	-3.4

A description of the DFC content model similar to that in this response is added to the SAR amendment Section 6.A.4.3 (renumbered 6.A.4.2.2), subsection titled "Fuel/Water Mixture."

A realistic configuration of fuel rubble, resulting from either loading of debris or collapse of a fuel rod would place all material at the bottom of the fuel tube. As light water reactor fuel (UO_2) at less than 5 wt % does not form a critical system without interspersed moderator, and LWR fuel assemblies are designed with a negative temperature coefficient (i.e., undermoderated), the configurations evaluated are by their very nature nonrealistic and conservative. Uranium oxide has a density of approximately 10 to 11 times that of water and would, therefore, not resuspend in the water moderator during any postulated condition of operation. Further, there is no probable mechanism that would result in the removal of the steel clad, which represents a significant neutron absorber, from the fuel lattice or mixture. Combining these nonlikely or nonphysically feasible events with a preferential flooding of all DFCs, which were designed to drain with the canister cavity, represents a conservative configuration for analysis, albeit not a realistic one.

NAC International Response to RAI 6.2 (cont'd)

EPRI Report No. 1015050, "Fuel Relocation Effect for Transportation Package," contains a discussion of the individual events, including nonmechanistic events, required to achieve a reactivity increase for damaged fuel. As the MPC-LACBWR criticality analysis has applied unclad rods at the maximum pitch allowed within the DFC, the analysis has already applied conditions beyond those expected during any cask operating condition.

d) Per Section 6.A.4.3 (renumbered 6.A.4.2.2): "For mixture cases, the mass is increased by adjusting the fuel volume in the mixture. For pellet stack cases, the active fuel height is increased." In the pellet stack cases, placing fuel in the active fuel region of the nominal assembly would displace moderator, thereby reducing reactivity. In the mixture cases, increasing the fuel volume fraction, i.e., mass, is the only option, as the study already starts with "full" density material and evaluates out to optimum moderator ratio.

Revised SAR amendment text is provided as noted.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.3 Provide additional information on off-normal and accident conditions.

There is no discussion of the consideration of the effect of accident conditions on the criticality analyses. Provide additional information demonstrating that the criticality analysis bounds any and all off-normal and accident conditions.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

Structural evaluations of the fuel, basket and canister demonstrate that no off-normal or accident condition induces geometry changes in the system outside the toleranced dimensions evaluated in SAR Appendix 6.A. Variations in moderator densities, partial flooding, and preferential flooding configurations bound any off-normal and accident moderator condition in the system.

This discussion is added to Appendix 6.A, Section 6.A.1, second paragraph. Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.4 Provide additional information regarding the composition of Boral.

The composition of Boral in Table 6.A.3-9 is inconsistent with that used in the MCNP input deck as well as that used for the other NAC-MPC applications. Provide the specifications of the Boral and justify what was used in the criticality analyses is conservative.

The staff needs this information to ensure that the applicant has met the requirements of 10 CFR 72.124(a) and 72.236(c).

NAC International Response

The BORAL composition in atom densities was recalculated based on the mass fraction and density provided in the MCNP input file and was also directly extracted from the MCNP output (which provides both atom densities of the material and atom fractions). Both approaches resulted in identical results within the significant digits of the calculation. RAI Figure 6.4-A shows the calculation sequence determining the material fractions followed by the conversion of the atom densities. The review of the calculations indicates that an editorial error occurred in Table 6.A.3-9 in that the carbon density exponent should have been E-03, not E-02. For consistent rounding, the corrected value is now shown as 9.30E-03 rather than 9.29E-02 in the previous version. There is no effect on MCNP calculations, as the atom densities were only generated for inclusion in the SAR table and correct weight fractions were input into MCNP.

Revised SAR amendment text (Table 6.A.3-9) is provided.

NAC International Response to RAI 6.4 (cont'd)

Figure 6.4-A BORAL Composition Calculation

Borated Absorber Material Density Calculation

10B Content	0.02	g/cm2
Core thickness (modelled)	0.127	cm
10B in Boron	17.68%	weight fraction
Boron in B4C	0.7829	weight fraction
10B density	1.575E-01	g/cm3
Boron density	8.909E-01	g/cm3
B4C density	1.138E+00	g/cm3
B4C	4.344E-01	weight fraction
Core density	2.6198	g/cm3
75% density	1.9649	g/cm3

	Weight Fraction	Partial Density (gm/cm3)	Atom Density (atom/b-cm)
B10	6.011E-02	1.181E-01	7.103E-03
B11	2.799E-01	5.500E-01	3.009E-02
C	9.430E-02	1.853E-01	9.298E-03
AL	5.656E-01	1.111E+00	2.481E-02
Total	4.344E-01	8.534E-01	7.129E-02

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.5 Provide additional information about the "component shift" study.

The staff is not clear about the results of the "component shift" study. Section 6.A.4.4.1, page 6.A.4-5 states that moving the fuel assembly and tube towards the basket center increases system reactivity. However, Figure 6.A.3-2 indicates that the criticality analysis was performed with the assemblies shifted away from the basket center as compared to drawing 630045-878. Clarify the results of the "component shift" study and justify that the criticality analysis is conservative.

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

Fuel tube and assembly location is controlled by the structural disks. The structural disk drawing is included in the SAR as drawing 6300458-893. The criticality analysis relevant details of this drawing (i.e, disk opening size and tolerance, and opening location and tolerance) are duplicated in Figure 6.A.3-2, "Support Disk Sketch." Drawing 630045-878 contains the reference (not manufacturing) dimension for the top weldment disk. Tubes do not extend through the top weldment and its dimensions do not serve criticality control functions.

Criticality evaluations in Section 6.A.4.4 (renumbered 6.A.4.3) applied the dimensions in Sketch 6.A.3-2 to investigate the effect of assembly and tube location within the basket at nominal and toleranced dimension sets. The fuel loading optimization section (6.A.4.2) discusses both component shift and fabrication tolerances applied in the analysis. As stated in this section, component shift applies to shifting fuel within the tube and the tube within the basket opening. For damaged fuel, this is extended to movement of the DFC.

NAC International Response to RAI 6.5 (cont'd)

As stated in the RAI request, the result of this analysis, as discussed in Section 6.A.4.4 (renumbered 6.A.4.3), determined that for the undamaged fuel system (no preferential flooding of DFCs), maximum reactivity occurs when shifting the tube and assembly toward the basket center. The preferential flooded, maximum reactivity configuration results in Section 6.A.4.4.2 (renumbered 6.A.4.3.2), under subsection "Maximum Reactivity Configuration Confirmation Studies," demonstrate that these effects are not significant for a dry TSC with flooded DFCs. As the dry TSC environment will have negligible interaction with neutrons moving from one DFC to the next, this behavior was expected.

The results of this analysis demonstrate that the configurations evaluated bound all shift and tolerance scenarios of the system simultaneously, while statistically not all tolerances and shifts are expected to occur simultaneously. The analysis results are, therefore, conservative.

SAR amendment text changes to the quoted SAR sections are made as a result of RAI 6-1 and RAI 6-2. Changes to SAR text from this RAI are limited to the addition of references to the disk sketch (Figure 6.A.3-2) for determining tolerance and component location studies in Section 6.A.4.2.1. The SAR already includes text in Section 6.A.4.2 and Section 6.A.4.3 (renumbered 6.A.4.2.2) describing the individual analysis basis and the bounding, i.e., conservative, application of the movement and tolerances.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.6 Provide additional information on the effect of partial flooding.

This includes scenarios where the cask is partially filled with water, or part water and part steam, or preferential or uneven flooding. The staff recognizes that partial flooding was considered based on a statement in Section 6.A.1 on page 6.A.1-2 that states that "The full moderator density TSC interior condition bounds any off-normal or accident condition with the exception of the preferentially flooded DFC case." However, the staff does not have enough information about the studies performed on partial flooding to understand what was done and that the maximum reactivity was appropriately determined. Clarify the statement in Section 6.A.3.1.1 on Page 6.A.3-2 that states "Partial flooding evaluations are limited to cases with a void region above the active fuel region and canister cavity moderator from the top of the active fuel region to the canister bottom." Provide additional information describing what was done. Provide figures as necessary illustrating which portions of the fuel/basket were flooded and with what density water. Demonstrate that the criticality analysis is conservative considering the effect of partial flooding.

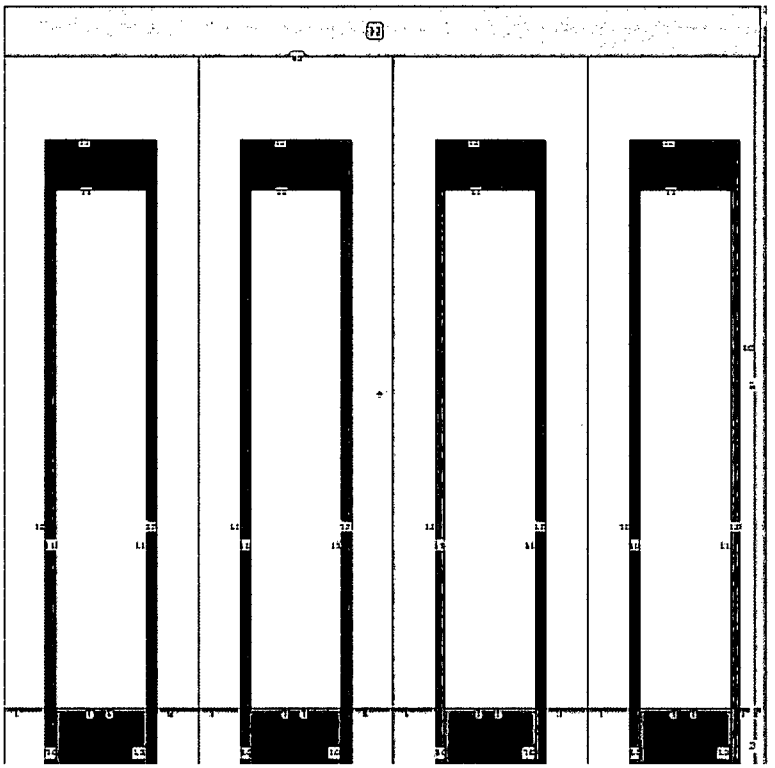
The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirement in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

The optimum moderator studies provided in Sections 6.A.4.1 and 6.A.4.2 demonstrate that maximum reactivity is achieved with full density water in the TSC (canister) for cases without preferential flooding, and void in the TSC when considering preferential flooding of the DFC. Figure 6.4.A-1 demonstrates that for the undamaged fuel configuration, the active fuel region must be covered in maximum density (0.9982 g/cc) moderator to achieve maximum reactivity. Similar Figures 6.4.A-2 and 6.4.A-3 demonstrate that the DFCs must be flooded at maximum density with a dry TSC to achieve maximum system reactivity. Therefore, applying a moderator density change in the active fuel region will result in a reduction in reactivity and would, therefore, not be conservative. In previous RAIs received by NAC on other license applications, the NRC has indicated interest in an evaluation of the relative effects of the steel lid reflection

NAC International Response to RAI 6.6 (cont'd)

versus water above the active fuel region, i.e., a moderator density change above the fuel region. This evaluation represents the only potential effect to be studied, as active fuel region changes were already determined to be bounded by the base optimum moderator analysis. Based on the large LWR active fuel region height (83 inches for the LACBWR fuel), reflection by water or steel above the fuel is not expected to be significant to system reactivity. The partial flood study included in the MPC-LACBWR submittal confirms this engineering judgment. A VISED model image is included in the RAI response for the preferential flood analysis. Yellow in the image is the water region extending to the top of the fuel region. Void is modeled to the canister lid.



Additional text on the condition evaluated is included in SAR Section 6.A.3.1.1. Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.7 Provide additional information about the cross sections used.

State which cross section set was used in the criticality analysis. Verify that these cross sections were also used in determination of the USL, or provide justification that the cross sections would provide similar or conservative results.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

The cross-section sets used are included in the MCNP sample input file (Section 6.A.7) material section as ".XXc" identifiers. Linking these identifiers to the ENDF library revision may be done through the MCNP manual.

The MCNP validation calculation extracted cross-section libraries from the completed critical experiment MCNP outputs and included a tabular listing within the verification calculation. The table is copied into the Excel workbook that constructs the MCNP input file and is used by a look-up function to add validated libraries to the nuclide/elements IDs. This assured that the LACBWR MCNP analysis applies validated cross-section libraries.

Revised SAR amendment text is provided in Section 6.A.4.1, page 6.A.4-1.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY

6.8 Provide additional information about the critical benchmark experiments.

Provide a list of the experiments used in the critical benchmark cases.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

NAC International Response

As stated in the first sentence of Section 6.A.5.3, the fresh fuel critical benchmark experiments are based on Reference [A8], the "International Handbook of Evaluated Criticality Safety Benchmark Experiments." Each experiment case number is included in the MCNP validation statistics summary table provided as Table 6.A.5-3 of the SAR. This case number represents the LEU-COMP-THERM experiment number in the handbook. The correlation of case number to experiment identification statement is added to the third paragraph of SAR amendment Section 6.A.5.3.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CONFINEMENT

7.1 Correct the acceptance criteria for the pressure test as stated in Section 8.A.1.1 "Loading and Closing the TSC," Step 44.e and as stated in Section 9.A.2.1 "Pressure Testing of the TSC" to be in accordance with ASME Code.

The two aforementioned sections state the acceptance criteria to be "no loss in pressure or visible water leakage from the closure lid weld during the 10 minute test period." However, the Code requires that the examination for leakage be done after the 10 minute pressure test duration and then the examination is done while maintaining pressure in the TSC equal to the greater of the design pressure or 3/4ths of the test pressure. In no case is holding the test pressure at a constant value for ten minutes considered as a substitute for examination of leakage.

10 CFR 72.236(l) requires that the spent fuel storage casks must be evaluated, by the appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal and credible accident conditions.

NAC International Response

The hydrostatic pressure test acceptance criteria is revised in SAR Section 8.A.1.1, Steps 44d and 44e, and Section 9.A.2.1 to clarify that the examination for visible water leakage is performed after the test pressure has been held for a minimum of 10 minutes and the test pressure is maintained throughout the examination period. The SAR sections are revised to read as follows:

Section 8.A.1.1, Step 44d

d. Maintain the TSC test pressure for a minimum 10-minute hold period. At the end of the 10-minute hold period, visually examine the closure lid-to-TSC shell weld for water leakage, while maintaining the test pressure. The test pressure shall be maintained until the completion of the visual inspection of the closure lid-to-TSC shell weld is completed.

NAC International Response to RAI 7.1 (cont'd)

Section 8.A.1.1, Step 44e

e. The hydrostatic test is acceptable if there is no visible water leakage from the closure lid-to-TSC shell weld based on visual examination of the weld after a minimum 10-minute hold period, while maintaining the test pressure.

Section 9.A.2.1, Pressure Testing of the TSC (changes are in bold)

Following completion of the closure lid-to-TSC shell weld during the TSC preparation operations after fuel loading, the TSC shall be hydrostatically pressure tested in accordance with ASME Code, Section III, Subsection NB, NB-6000 requirements as described in the operating procedures in Appendix 8.A. The minimum test pressure of 15 psig shall be applied to the drain port connection for a minimum **10-minute hold period**. The minimum test pressure is 125% of the normal design pressure of 12 psig. **There shall be no visible water leakage from the closure lid-to-TSC shell weld based on visual examination of the weld after a minimum 10-minute hold period, while maintaining the test pressure. Test pressure shall be maintained until the completion of the visual weld examination.**

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.1 The acceptance and sampling plan for the neutron absorber in the SAR should be incorporated into the CoC by reference. A statement to the effect that the wet chemistry analysis will be calibrated against neutron absorption methods should be added to the acceptance plan.

Accurate knowledge of the boron content of the absorbers is necessary to assure that the criticality determinations are correct.

This information is required for compliance with 10 CFR 72.236(c).

NAC International Response

The MPC-LACBWR amendment has been structured following the requirement and guidance of 10 CFR 72.244 to minimize change from currently certified dry storage systems in order to reduce review time and expedite the licensing process. The neutron absorber material and basket design configuration proposed for the MPC-LACBWR design is identical to the NAC-MPC systems certified for Yankee Class and Connecticut Yankee fuels (MPC CoC No. 1025), as well as the NAC-UMS® system (UMS® CoC No. 1015) in service at three major operating nuclear power plants.

The neutron absorber material specified for the MPC-LACBWR design is BORAL. This material has been certified for use in numerous dry storage systems over the past 25 years with an imposed performance margin limiting ^{10}B areal density for criticality control to 75% of the design minimum fabrication values. The proposed MPC-LACBWR Technical Specifications in SAR Chapter 12, Appendix 12.B, Section B 3.2.1, define the neutron absorber material as BORAL and the design minimum ^{10}B areal density as 0.02 g/cm^2. As this application does not open the certified NAC-MPC system to the use of alternative neutron absorber materials that have entered the market over the past 10 years, i.e., borated aluminum and metal matrix composite (MMC) materials, and limits the design to BORAL, the neutron absorber material acceptance testing requirements that have been reviewed and approved in existing regulatory dry storage system licenses have been adopted into the current amendment application. Condition 2 of the NAC-MPC CoC No. 1025 incorporates the acceptance tests and maintenance program described in Chapter 9 of the FSAR by reference.

NAC International Response to RAI 8.1 (cont'd)

Historically, chemical analysis has been a standard testing method during the fabrication process of BORAL to evaluate acceptability of the completed product to the specified ^{10}B limits. Limiting ^{10}B credit to 75% of the design minimum fabrication values, in combination with chemical analysis, has provided acceptable assurance that these past licensed dry storage systems have been designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

From the above discussion and previous licensing bases, the sampling plan defined in Section 9.A.2.7.1 provides assurance that the fabricated material is compliant with the specification. No CoC requirements on acceptance testing of BORAL are, therefore, proposed for the MPC-LACBWR system.

Based on NAC discussions with the NRC materials review staff and the request to link wet chemistry testing to neutron attenuation testing, the SAR amendment Section 9.A.2.7.2 is modified as follows (in bold).

9.A.2.7.2 Wet Chemistry Test Performance

An approved facility with chemical analysis capability shall be selected to perform the wet chemistry tests. The tests will verify the presence of boron and provide input to the calculation of the ^{10}B areal density. **Chemical test methods will be calibrated against neutron absorption methods for the BORAL® neutron absorber.**

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.2 Provide assurance that the 30% fission gas release (FGR) used in the pressurization calculations is bounding for the La Crosse fuel.

SAR (Section 11.A.2.1.1) uses a 30% FGR to estimate the internal transportable storage container (TSC) pressure. This is a good bounding gas release for modern day fuel. Some earlier vintages of BWR fuel exhibited significantly higher fission gas release.

This information is required for compliance with 10 CFR 72.236(d).

NAC International Response

The LACBWR fuel represents early generation BWR fuel with a 1 atm backfill, which is not typical of current generation fuel rods. La Crosse Allis Chalmers (AC) fuel rods were sent for PIE examination and a comprehensive report of the results was published in 1976. Two of the rods sent for PIE examination were measured for fission gas release (FGR) percentage. These rods were ~22 GWd/MTU burned and are representative of the maximum burnup material in the LACBWR spent fuel inventory. The rods released less than 1% of fission gas inventory (measured by a Kr-85 released to generated fraction). Licensing documentation provided to NRC as part of the fuel redesign in 1976 justified that Exxon fuel assembly fission gas pressurization would be bounded by the AC fuel design. It is, therefore, conservative to apply a 30% FGR to all La Crosse fuel types.

As the LACBWR assemblies represent first-generation BWR fuel and the PIE FGR results were limited to two rods, the NRC may be concerned that the LACBWR fuel may not be sufficiently characterized. NAC has, therefore, recalculated TSC pressures applying a 70% FGR fraction. The 70% fraction bounds publicly available references on LWR FGR (both experimental and theoretical) for fuel with higher burnup and significantly higher linear heat generation rates (and temperature) than LACBWR fuel and, therefore, represents a very conservative value. Normal and accident condition pressures were recalculated based on the 70% FGR. Maximum accident condition pressure under this condition is 32.1 psig.

NAC International Response to RAI 8.2 (cont'd)

Normal, off-normal and accident pressure calculations in the SAR amendment are revised to reflect the increase in FGR fraction, and SAR amendment text changes are made to pages 4.A.3-31, 4.A.3-34, 4.A.3-35, 11.A.1-3, 11.A.2-1 and 11.A.2-2.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.3 Specify the minor changes that were made to the vertical concrete cask (VCC).

The request letter said that minor changes were made to the VCC but never specified the changes. An evaluation of the importance or significance of these changes cannot be made without knowing their nature.

This information is required for compliance with 10 CFR Part 72.

NAC International Response

The minor differences between the VCC for MPC-LACBWR versus the currently licensed Yankee MPC and CY-MPC designs are limited to the following:

- Height 162 in. (MPC-LACBWR) versus 160 in. (Yankee MPC) and 190.6 in. (CY-MPC)
- Shielding 22 in. concrete and 2.5 in. steel (MPC-LACBWR) versus 21 in. concrete and 3.5 in. steel (both Yankee MPC and CY-MPC)
- Weight 141,200 lbs nominal (MPC LACBWR) versus 155,000 lbs nominal (Yankee MPC) and 186,000 lbs nominal (CY-MPC)

The above data are reflected in Table 1.A.2-3 for MPC-LACBWR and in Table 1.2-3 for Yankee and CY-MPC.

In addition, the following enhancements were implemented for the MPC-LACBWR VCC:

- The base weldment carbon steel plate is covered with a ¼-in-thick stainless steel plate to prevent contact of the stainless steel canister and the carbon steel plate.
- The lid includes an 8-in concrete plug encased in carbon steel extending into the cask cavity from the bottom of the 1.5-in-thick carbon steel lid.
- Radiation shielding is installed in the air inlets to reduce dose rates local to the air inlets at the base of the VCC.

NAC International Response to RAI 8.3 (cont'd)

All changes/enhancements are described in Section 1.A.2.1.2 of the MPC-LACBWR, Revision 08A submittal, and are depicted on the applicable license drawings.

Based on the content of this response, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.4 Provide mechanical and thermal properties of the alloy 348H stainless steel used in the cladding.

The SAR only specifies that cladding as stainless steel. No specific alloy is given although the staff was able to determine it is 348H.

This information is required for compliance with 10 CFR 72.236(a) and 10 CFR 72.236(e).

NAC International Response

The PIE examination and a comprehensive report of the results published in 1976 refer to this clad as stainless steel Type 348H. The mechanical properties of this stainless steel fuel clad, which were obtained from the 1976 Report, are used in the accident evaluations of the fuel assembly in Section 11.A.4. In Section 11.A.4, the following properties are listed:

Elastic modulus:	25.3×10^6 psi
Density:	0.291 lb./in^3
Yield strength:	37 ksi (obtained from the reported margin of safety and maximum stress)

The thermal properties employed in the thermal evaluation correspond to those of stainless steel contained in Table 4.4-2.

Revised SAR amendment text is provided in Section 11.A.4, page 11.A.4-1, to identify the stainless steel comprising the clad and the source of this information as indicated above.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.5 Provide justification or preferably measurement data supporting a maximum fuel assembly bow of 0.25 inch for the Exxon Nuclear Company (ENC) stainless steel clad fuel assemblies.

SAR Section 11.A.4 indicates that a maximum assembly bow of 0.25 inch is expected for the ENC assemblies. While this is reasonable for Zircaloy cladding, it is not clear that it is the maximum for assemblies containing stainless steel clad fuel. Excessive bow would create problems with inserts and removing assemblies from the basket tubes.

This information is required for compliance with 10 CFR 72.236(h).

NAC International Response

The fuel integrity is evaluated for the Exxon Nuclear Company (ENC) stainless steel clad fuel assemblies, since all the Allis Chalmers fuel assemblies are loaded into DFCs. The inside dimension for a standard fuel tube (or DFC) is 5.75 inches, while the fuel assembly dimension is 5.614 inches. Therefore, the maximum bow for fuel placed in the basket tube is 0.136 inch. A bow of 0.25 inch is conservatively used in the model.

Revised SAR amendment text is provided in Section 11.A.4, page 11.A.4.1, to change the largest bow between a straight fuel assembly and the basket fuel tube inner wall to 0.136 inch.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.6 Define a "handling container" and specify its properties.

The definition of FUEL DEBRIS (CoC Section A.1.1) says that fuel debris will be placed in a handling container but never defines the term or gives specifications for this container.

This information is required for compliance with 10 CFR 72.236(a).

NAC International Response

The handling container for fuel debris is a stainless steel device (or container) designed to facilitate handling and loading of fuel debris, fuel rods and rod segments during DFC loading operations. The container is configured for drainage and drying and is provided with a handling mechanism. The handling container is considered dunnage and no credit is taken for use of the handling container in any of the analyses qualifying the NAC-MPC storage system or any of its components.

CoC Section A1.1 states that "FUEL DEBRIS may be loaded in a handling container" not "will be placed" as the RAI question states.

Revised SAR amendment change pages include the addition of the definition of "handling container" to Table 1.A-1 (page 1.A-5) and Section A1.1 of the CoC (page 12.A-5) and revision to the definition of "Fuel Debris" in Table 1.A-1 (page 1.A-3) to include "(may be placed into a handling container)." Also, in Section 2.A.1.3, page 2.A.1-3, changed "(including optional fuel debris handling canister)" to "(including optional fuel debris handling container)."

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.7 Define a "fuel bundle."

Item 5 of the LACBWR DAMAGED FUEL ASSEMBLY definition contains the term fuel bundle (CoC Sec. A1.1).

This information is required for compliance with 10 CFR 72.236 (a).

NAC International Response

NAC agrees to change Item 5 of the LACBWR DAMAGED FUEL ASSEMBLY definition in Appendix 1.A and Section A1.1 of the CoC to state:

"The SNF is no longer in the form of an INTACT FUEL ASSEMBLY (e.g., consists of or contains debris such as loose fuel pellets, individual fuel rods or rod segments)."

This change is made to pages 1.A-2 and 12.A-6 of the MPC-LACBWR amendment, Revision 09C.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.8 Revise the definition of "Structural Damage."

The current definition leaves ambiguity in the meaning of intact fuel. Fuel assemblies with structural damage CANNOT be classified as "intact" as indicated in the current definition in Section A1.1 of the CoC.

The information is required for compliance with 10 CFR 72.236(a).

NAC International Response

NAC has revised the last sentence of the STRUCTURAL DAMAGE definition in Section A1.1 of the CoC to read:

"Yankee and CY fuel assemblies with STRUCTURAL DAMAGE are considered to be INTACT FUEL ASSEMBLIES provided that they do not have failed or DAMAGED FUEL RODS."

The last sentence of the STRUCTURAL DAMAGE definition does not apply to LACBWR fuel contents.

This change is made to page 12.A-9 of the MPC-LACBWR amendment, Revision 09C.

Revised SAR amendment text is provided.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

MATERIALS

8.9 Add the dye penetrant testing (PT) weld inspection criterion for the entire load bearing welds to the drawing of the damaged fuel can. Specify the size of the mesh that will be used at the ends of the damaged fuel cans.

This information is required for compliance with 10 CFR 72.236(j).

NAC International Response

Dye penetrant testing (PT) weld inspection criterion for the load bearing welds of the damaged fuel can is not required due to the inclusion of a weld quality factor of 0.5 in the structural analysis of the weld. This value of weld quality factor is representative of visual examination as currently defined on Drawing 630045-902, Revision 0. The structural analysis of the DFC load bearing weld is presented in Section 3.A.4.3.4.

Item 6 and item 14 on Drawing 630045-902, Revision 0, are the filter screens located in the damaged fuel can bottom plate and the closure lid, respectively. The description included on the Bill of Materials identifies these items as NEWARK PAC 250 × 250 × 0.0016 for the defined mesh.

Based on the content of this response, no further change to the SAR amendment text is required.

**NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION**

MATERIALS

8.10 Clarify the phrase "stainless steel covered BORAL" (SAR Section 1.A.2.1.1). Is the BORAL completely isolated from the pool water when loading the TSC? Evaluate the potential to generate hydrogen by the interaction of any exposed aluminum in the basket, such as the BORAL and heat transfer disks, with the borated pool water during loading. What steps are taken to mitigate the effects of any generated hydrogen?

Any exposed aluminum surfaces in the TSC will react with the borated pool water when the fuel is being loaded. The result of the interaction will be the generation of hydrogen. If sufficient steps are not taken after the lid is put on the TSC, there may be a hydrogen buildup that can possibly be explosive during the welding process.

This information is required for compliance with 10 CFR 72.236(b).

NAC International Response

The phrase "stainless steel clad covered BORAL" is representative of the fuel tube assembly presented in drawing 630045-881, Revision 0. Item 3, CLADDING, is a 26 GA stainless steel sheet that is placed over item 2, neutron absorber, BORAL, that provides support and placement control for the BORAL sheet. As presented in the fuel tube assembly drawing, each corner of the cladding cover sheet is cut at a 45-degree angle to provide venting and draining of the BORAL enclosure. The design of this fuel tube with the stainless steel cladding cover enclosing the BORAL neutron absorber is identical in configuration to the fuel tube designs used in the certified NAC-MPC and NAC-UMS® dry storage systems. Differences in the tube design between the different systems are limited to cross-section and length for the specific basket and fuel content.

It is noted that a natural oxidation layer develops on the surface of both the BORAL neutron absorber sheets, and the aluminum heat transfer disks and provides a protective coating from borated pool water and reduces the potential of hydrogen generation. At the current time, NAC has loaded more than 220 dry storage systems having a similar basket and fuel tube neutron absorber material design as the current proposed amendment and has not experienced any hydrogen generation issue. Similar to the operating controls implemented on these other systems, SAR Operating Procedures, Section 8.A.11, steps 36 and 37, address hydrogen monitoring to be performed during the MPC-LACBWR closure lid welding operation.

NAC International Response to RAI 8.10 (cont'd)

It is further noted that the MPC-LACBWR amendment is specific to the Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor. The DPC pool does not contain borated water and, therefore, the issue of hydrogen generation from exposure to a borated pool is not relevant.

Based on the content of this response, no further change to the SAR amendment text is required.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RADIATION PROTECTION

11.1 Clarify the information in Table 10.A.3-1.

Provide additional clarifying information on the Exposure calculated in Table 10.A.3-1. The values in Step 1 appear inconsistent with the values calculated in Steps 7 and 8. Clarify if the Exposure is listed per person or for the total of all personnel and correct the discrepancy.

This information is needed to ensure that the regulations in 10 CFR 20.1201 are met.

NAC International Response

Dose (exposure) in Table 10.A.3-1 is intended to represent total exposure (sum over all personnel) for the step. Steps 7 and 8 did not contain the correct average dose rate (they are the values obtained by assuming one person involved in the task). The average dose rate is corrected in Steps 7 and 8 to allow the multiplication of number of personnel times task duration times average dose rate to achieve the listed exposure. The table is the result of collapsing a large number of subtasks with various numbers of personnel in various radiation field strengths. The dose rate column in the summary table was generated by calculating step/task average dose rates, based on a fixed number of personnel and the overall time for completing all subtasks.

A SAR amendment text change is made to Table 10.A.3-1.

September 2009

Revision 09C

NAC-MPC

NAC Multi-Purpose Canister

FINAL SAFETY ANALYSIS REPORT

MPC-LACBWR Amendment

Volume 1 of 2

Docket No. 72-1025

NAC INTERNATIONAL

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

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Chapter 1
Appendix 1.A

1.A GENERAL DESCRIPTION OF THE MPC-LACBWR STORAGE SYSTEM

NAC International (NAC) is amending the design of the Multi-Purpose Canister system (NAC-MPC) for the long-term storage of Dairyland Power Cooperative La Crosse Boiling Water Reactor (LACBWR) spent nuclear fuel. The MPC-LACBWR system consists of a transportable storage canister, vertical concrete cask, and a transfer cask.

The transportable storage canister is designed and fabricated to meet the requirements for transport in the NAC Storage Transport Cask (NAC-STC).

In long-term storage, the transportable storage canister is installed in a vertical concrete cask, which provides passive radiation shielding and natural convection cooling. The vertical concrete storage cask also provides protection during storage for the transportable storage canister under adverse environmental conditions. Employing the guidance in ISG-18, Rev 1, the MPC-LACBWR closure design uses a single lid with a redundant weld closure to preclude loss of contents and to preserve the general health and safety of the public during long-term storage of spent fuel.

The transfer cask is used to move the transportable storage canister from the work stations where the canister is loaded and closed to the vertical concrete cask. It is also used to transfer the canister from the vertical concrete cask to the NAC-STC for transport.

This Safety Analysis Report demonstrates the ability of the NAC-MPC system to satisfy the Nuclear Regulatory Commission (NRC) requirements for the storage of spent fuel, as prescribed by 10 CFR 72.

This Appendix provides a general description of the major components of the MPC-LACBWR system and a description of the system operation. The terminology used throughout this Appendix is summarized in Table 1.A-1. Table 1.A.5-1 provides a compliance matrix to the regulatory requirements and acceptance criteria specified in NUREG-1536. This matrix describes how the MPC-LACBWR Safety Analysis Report complies with each requirement and criterion listed in NUREG-1536. Table B.3-1 of Appendix 12.B of the Technical Specifications provides a list of the alternatives from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) that are applicable for the canister.

Table 1.A-1 Terminology

NAC-STC Cask	The licensed spent-fuel transport cask consisting of a spent fuel storable transport cask body with dual closure lids and energy-absorbing impact limiters (Certificate of Compliance No. 71-9235).
Confinement System	The components of the transportable storage canister intended to retain the radioactive material during storage.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, closure lid, closure ring and redundant port covers that holds the spent fuel in the canister basket.
MPC-LACBWR Contents	Up to 68 Dairyland Power Cooperative La Crosse BWR spent fuel assemblies, including up to 32 Damaged Fuel Cans.
LACBWR Damaged Fuel Assembly	Spent nuclear fuel (SNF) that cannot fulfill its fuel-specific or system-related function. SNF is classified as a LACBWR damaged fuel assembly under the following conditions. 1. There is visible deformation of the rods in the SNF assembly. Note: This is not referring to the uniform bowing that occurs in the reactor; this refers to bowing that significantly opens up the lattice spacing. 2. Individual fuel rods are missing from the assembly and the missing rods are not replaced by a solid dummy rod that displaces a volume equal to, or greater than, the original fuel rod. 3. The SNF assembly has missing, displaced or damaged structural components such that either radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch); or the assembly cannot be handled by normal means (i.e., crane and grapple). 4. Any SNF assembly that contains fuel rods for which reactor operating records (or other records or tests) cannot support the conclusion that they do not contain gross breaches. Note: Breached fuel rods with minor cladding defects (i.e., pinhole leaks or hairline cracks that will not permit significant release of particulate matter from the spent fuel rod) are classified as undamaged. 5. The SNF is no longer in the form of an INTACT FUEL ASSEMBLY (e.g., consists of or contains debris such as loose fuel pellets, individual fuel rods or rod segments).

LACBWR Undamaged Fuel Assembly	A spent nuclear fuel assembly that can meet all fuel-specific and system-related functions. A LACBWR undamaged fuel assembly is spent nuclear fuel that is not a LACBWR damaged fuel assembly, as defined herein, and does not contain assembly structural defects that adversely affect radiological and/or criticality safety. As such, a LACBWR undamaged fuel assembly may contain breached spent fuel rods (i.e., rods with minor defects up to hairline cracks or pinholes), but cannot contain grossly breached fuel rods.
LACBWR Damaged Fuel Can	A stainless steel container that is similar to an enlarged fuel tube that confines a LACBWR Fuel Assembly, Damaged Fuel Assembly, or debris. A damaged fuel can is closed on its bottom end by a stainless steel bottom plate having screened openings and on its top end by a stainless steel lid that also has screened openings. The screened openings allow gaseous and liquid media to escape, but minimizes the dispersal of gross particulate.
Fuel Debris	Fuel in the form of particles, loose pellets and fragmented rods or assemblies (may be placed into a handling container).
Canister Basket	The structure placed in the transportable storage canister to support the fuel assemblies (fuel basket).
-Support Disk	A circular stainless steel plate with square holes machined in a symmetrical pattern that provides the primary lateral load-bearing component of the canister basket.
-Heat Transfer Disk	A circular aluminum plate with square holes machined in a symmetrical pattern. The heat transfer disk enhances heat transfer in the fuel basket.
-Fuel Tube	A stainless steel tube having a square cross-section that may have BORAL neutron poison material on its exterior surfaces. Certain MPC-LACBWR fuel tubes may have an aluminum sheet in place of a BORAL sheet on one side.
-Tie Rod	A stainless steel rod used to align the support disks and heat transfer disks in the fuel basket structure.
-Split Spacer	Spacers installed on the tie rod between the support disks to properly position, and provide axial support for, the support disks and the heat transfer disks.
Closure Lid	The primary confinement boundary for the canister. It is located directly above the canister basket and provides the lifting point for the loaded canister.

-Drain Port	A penetration located in the closure lid to permit draining of the canister cavity.
-Vent Port	A penetration located in the closure lid to aid in draining and backfilling the canister cavity.
-Port Cover	The stainless steel dual covers that close the vent and drain ports, which are welded in place following draining, drying, and backfilling operations and provide both a confinement boundary and redundant closure .
-Quick Disconnect	The quick-disconnect valved nipple used in the vent and drain ports to facilitate operations.
Closure Ring	The steel bar that is positioned and welded above the closure lid to canister shell weld that provides redundant confinement.
Vertical Concrete Cask (Concrete Cask) (Storage Cask)	A reinforced concrete cylinder closed at the top end by a shield plug and lid that holds the transportable storage canister during storage. The vertical concrete cask is formed around a steel inner liner and base.
- Lid	A thick carbon steel bolted closure with incased concrete for shielding for the storage cask. The lid precludes access to the canister and provides radiation shielding to reduce skyshine.
- Liner	A thick carbon steel shell that forms the annulus of the concrete storage cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the inlet air vents` and the pedestal that supports the canister inside the storage cask.
Transfer Cask	A shielded lifting device for the empty and loaded canister. It is used for the vertical transfer of the canister between work stations and the storage cask or the transport cask. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks.
- Lifting Trunnions	Carbon steel trunnions used to lift and move the transfer cask.
Adapter Plate	A carbon steel plate that attaches to the top of the transport or storage cask to facilitate the installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined and is defined as the allowable stress of a material divided by its actual (calculated) stress.

MPC-LACBWR	MPC-LACBWR is a NAC-MPC system having a fuel basket designed to accommodate Dairyland Power Cooperative La Crosse BWR (LACBWR) reactor spent fuel. The MPC-LACBWR meets the NAC-MPC system requirements.
Handling Container	A stainless steel device (or container) designed to facilitate handling and loading of FUEL DEBRIS, fuel rods and rod segments during DFC loading operations.

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Chapter 2
Appendix 2.A

2.A.1.3 MPC-LACBWR Damaged Fuel Can

The LACBWR Damaged Fuel Can (DFC) is required to be preferentially loaded because it can only be installed in the fuel positions that are oversized. Available DFC positions are identified as "B" and "C" locations in Figure 2.A.1-1.

Single fuel assemblies with damage to structural components that prevents handling the fuel assembly in the normal manner, fuel assemblies that have one or more damaged fuel rods, and fuel debris (including optional fuel debris handling container) must be loaded in DFCs. However, an undamaged fuel assembly may also be loaded into a DFC.

The DFC is closed on the bottom end by a 0.5-inch-thick plate that is welded to the can shell. The plate has drilled holes in each corner to allow water to drain from the DFC. A screen covers the holes to preclude the release of gross particulate material from the fuel assembly. The DFC lid is not secured to the DFC shell, but is held in place when the closure lid is installed in the canister. The DFC lid also has four drilled and screened holes. The damaged fuel assembly is inserted in the DFC. Lifting lugs in the DFC shell allow the loaded DFC to be lifted and placed in the canister basket. Alternately, the DFC may be inserted in a basket corner position before the designated fuel assembly is inserted into the DFC. The DFC lid is installed after fuel loading is completed.

The MPC-LACBWR DFC design and fabrication specification summary is provided in Table 1.A.3-3. The major physical design parameters are provided in Table 1.A.3-2. The structural evaluation is provided in Sections 3.A.4.4.2.4, 11.A.2.11.2.2 and 11.A.2.12.2.2.

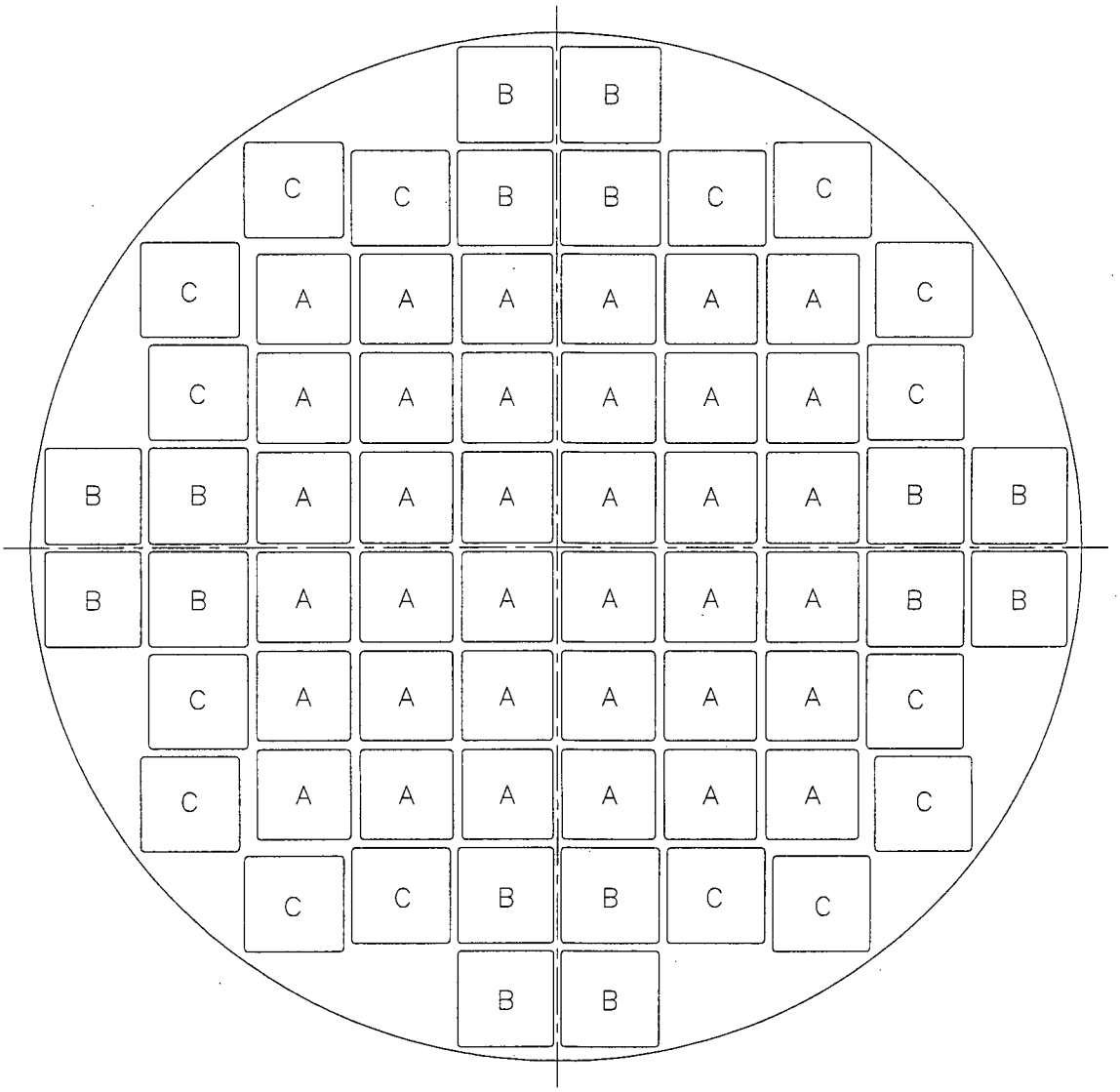
2.A.1.4 LACBWR Nonfuel Hardware

No nonfuel hardware is included in the MPC-LACBWR system design. BWR channels are not permitted for storage.

2.A.1.5 LACBWR Damaged Fuel

A transportable storage canister configured for damaged fuel holds up to 32 damaged fuel cans (DFCs) located in the "B" and "C" positions of the basket shown in Figure 2.A.1-1. A DFC may contain either an undamged or a damaged LACBWR spent fuel assembly of the types described in Table 2.A.1-1, and may also contain fuel debris. To analytically bound loose fuel material potentially trapped in a damaged fuel assembly, but not originating from the damaged fuel assembly, evaluations increased the spent fuel rod material quantity inside the DFC by 5% over that of a LACBWR undamaged assembly. The structural, thermal, shielding, confinement, and criticality effects of the damaged fuel cans are separately evaluated in the appropriate chapters.

Figure 2.A.1-1 Loading Diagram for MPC-LACBWR Fuel Basket



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.
Damaged Allis Chalmers fuel maximum enrichment 3.64 wt % ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.
Damaged Allis Chalmers fuel maximum enrichment 3.94 wt % ^{235}U.

Table 2.A.1-1 LACBWR Fuel Parameters

	Allis Chalmers	Exxon
ASSEMBLY CONFIGURATION		
Assembly Length (in)	102.5	102.5
Assembly Width (in)	5.6	5.6
Assembly Array	10x10	10x10
Assembly Weight (lb)	400	400
Enrichment-wt. % ^{235}U		
Maximum	3.64/3.94[1]	3.71[2]
Minimum	3.6	3.6
Initial Fuel Weight ($KgUO_2$/Assembly)	138	127
Initial Heavy Metal (KgU/Assembly)	122	112
Max. Burnup (MWd/MTU)	22,000	21,000
Min. Cool Time (yr)	28	23
Max. Decay Heat (W)	63	62
FUEL ROD CONFIGURATION		
Fuel Rod Pitch (in)	0.565	0.557
Active Fuel Length (in)	83	83
Rod OD (in)	0.396	0.394
Clad Thickness (in)	0.02	0.022
Pellet OD (in)	0.35	0.343
Rods per Assembly	100	96
Fuel Material	UO_2	UO_2
Clad Material	Stainless Steel	Stainless Steel
INERT RODS		
Displacement Rod Material	N/A	SS Clad / Zirc Slug
Displacement Rod Diameter (in)	N/A	0.394
Number Per Assembly	N/A	4

1. Enrichments are for Type 1 and Type 2 Allis Chalmers fuel, respectively.
2. Represents maximum planar average enrichment.

Note: Values represent unirradiated fuel conditions.

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Chapter 3

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Appendix 3.A STRUCTURAL EVALUATION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR 3.A-i

Fuel Baskets," and using the additional guidance contained in ASME Code Section III, Article NF-3000 and in ASME Code Section III, Appendix F. The structural evaluation of the CY-MPC Reconfigured Fuel Assembly is presented in Section 11.4.2.

The CY-MPC reconfigured fuel assembly is 8.9 inches square, which is slightly larger than a standard fuel assembly and, thus, requires that it be loaded in one of the oversized loading positions in the basket. These are positions 1, 4, 23 and 26 in the 26-assembly basket (Figure 2.1-2) or positions 1, 4, 21 and 24 in the 24-assembly basket (Figure 2.1-3). The weight of a loaded reconfigured fuel assembly (approximately 1,150 pounds) is less than the weight of other Connecticut Yankee fuel assemblies. The maximum temperature of the Reconfigured Fuel Assembly components is determined by the thermal analyses presented in Section 4.5.3.

The CY-MPC reconfigured fuel assembly has been evaluated for a normal condition of storage end impact of 20g and a side impact resulting in a deceleration of 20g. These analyses, presented in Section 3.4.4.5, bound the design conditions of the reconfigured fuel assembly for normal conditions of storage.

The CY-MPC reconfigured fuel assembly has also been evaluated for accident conditions, i.e., an end impact of 60g and a side impact of 60g. The analysis, presented in Section 11.4.2, bounds the design conditions of the CY-MPC reconfigured fuel assembly for accident conditions of storage.

Therefore, the structural evaluations of the NAC-MPC system containing other Connecticut Yankee fuel assemblies (Chapters 3.0 and 11.0) bound those of the NAC-MPC system containing one or more CY-MPC reconfigured fuel assemblies.

CY-MPC Damaged Fuel Cans

The CY-MPC damaged fuel can is installed in the basket fuel positions that are oversized. These are positions 1, 4, 23, and 26 of the 26-assembly basket and positions 1, 4, 21 and 24 of the 24-assembly basket. It is designed to hold a single fuel assembly that may be classified as intact or as damaged. The CY-MPC damaged fuel can is shown in Figure 1.3-3 and in Drawings 414-901 and 414-902. As shown in the drawings, the can is 141.5 inches in length, and has an internal square dimension of 8.7 inches. The shell of the damaged fuel can, which has the shape of a square tube, is 18 gauge stainless steel.

The can is closed on the bottom end by a 0.5-inch thick plate that is welded to the can shell. The plate has drilled holes in each corner to allow water to drain from the can. A screen covers the holes to preclude the release of gross particulate material from the fuel assembly. A lid having an overall depth dimension of 2.3 inches closes the can. The lid is not secured to the can shell, but is held in place when the shield lid is installed in the canister. The lid also has four drilled and screened holes. The damaged fuel assembly is inserted in the can and the can lid is installed. Lifting lugs in the can shell allow the loaded can to be lifted and installed in the canister basket. Alternately, the fuel can may be inserted in a basket corner position before the damaged fuel assembly is inserted in the fuel can. The loaded weight of the damaged fuel can is 1,500 pounds.

The CY-MPC damaged fuel can design and fabrication specification summary is provided in Table 1.3-6. As shown in that table, the damaged fuel can and the reconfigured fuel assembly are designed to the same criteria. The major physical design parameters are provided in Table 2.1-5. The structural evaluation of the damaged fuel can is provided in Sections 3.4.4.6 and 11.4.3.

3.1.1.3 Description of the MPC-LACBWR Storage System

The MPC-LACBWR storage system, which is designed to store up to 68 La Crosse Boiling Water Reactor (LACBWR) spent fuel assemblies, is similar to the Yankee-MPC storage system in most respects. Of the 68 fuel assemblies, 32 of the positions may hold damaged fuel cans that contain primarily Allis Chalmers fuel assemblies. The discussion, description and details of the MPC-LACBWR design are contained in Appendix Section 3.A.1.1.

3.1.2 Design Criteria

The NAC-MPC structural design criteria are specified in Section 2.2. The load combinations of normal, off-normal, and accident loadings have been evaluated in accordance with ANSI 57.9 and ACI 349 for the concrete cask, and in accordance with the ASME Code, Section III, Division I, Subsection NB for Class 1 components for the canister, as listed in Tables 2.2-1 and 2.2-2, respectively. The basket is evaluated in accordance with ASME Code, Section III, Subsection NG, and NUREG-6322. The transfer cask and the lifting yoke are lifting devices that are designed to NUREG-0612 and ANSI N14.6.

Chapter 3
Appendix 3.A

Figure 3.A.4.4.1-1 MPC-LACBWR Canister Shell Assembly Finite Element Model

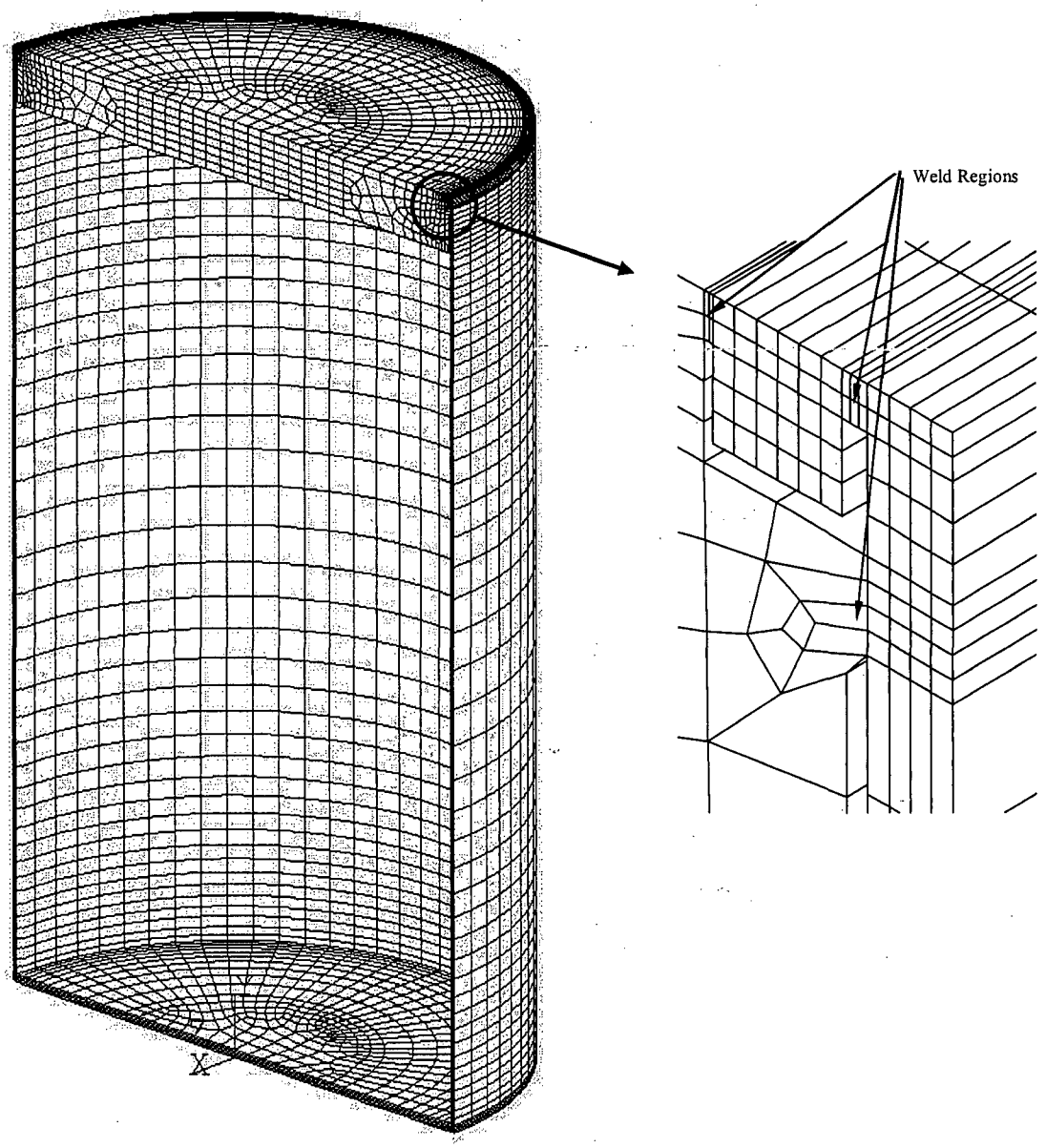
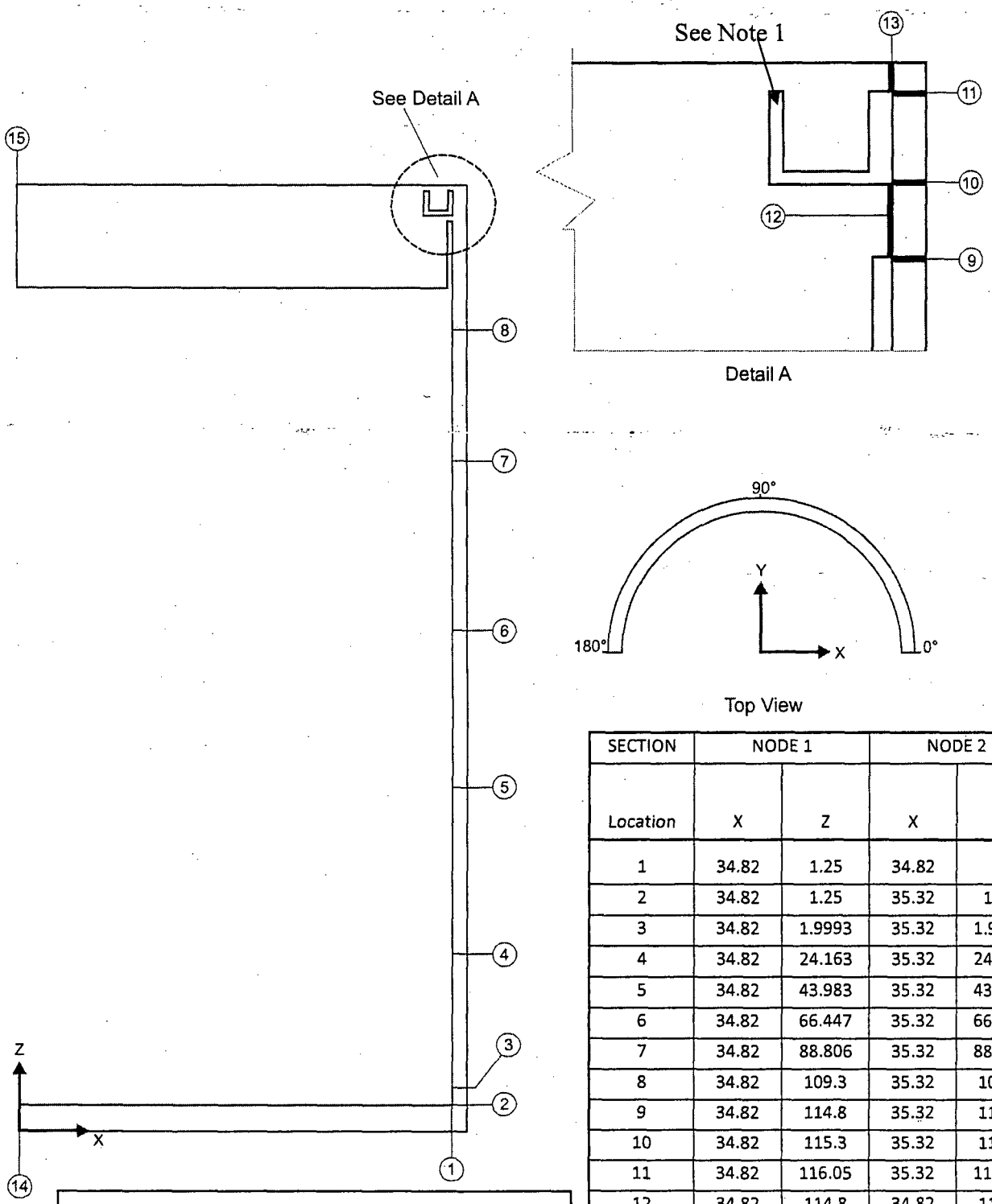


Figure 3.A.4.4.1-2 MPC-LACBWR Canister Finite Element Model Stress Section Locations



SECTION	NODE 1		NODE 2	
Location	X	Z	X	Z
1	34.82	1.25	34.82	0
2	34.82	1.25	35.32	1.25
3	34.82	1.9993	35.32	1.9993
4	34.82	24.163	35.32	24.163
5	34.82	43.983	35.32	43.983
6	34.82	66.447	35.32	66.447
7	34.82	88.806	35.32	88.806
8	34.82	109.3	35.32	109.3
9	34.82	114.8	35.32	114.8
10	34.82	115.3	35.32	115.3
11	34.82	116.05	35.32	116.05
12	34.82	114.8	34.82	115.3
13	34.82	116.05	34.82	116.3
14	0	1.25	0	0
15	0	116.3	0	109.3

Note 1: The Section 13 evaluation of the stresses in the closure ring weld bounds the stresses in the inner weld of the closure ring.

Chapter 4

4.0 THERMAL EVALUATION

The NAC-MPC is provided in three configurations. The first is designed to safely store up to 36 intact Yankee Class spent fuel and reconfigured fuel assemblies and is referred to as the Yankee-MPC. The second is the Connecticut-Yankee MPC, referred to as the CY-MPC, is designed to store up to 26 Connecticut Yankee fuel assemblies, CY-MPC reconfigured fuel assemblies and CY-MPC damaged fuel cans. The third is the La Crosse BWR MPC, referred to as the MPC-LACBWR, designed to store up to 68 La Crosse fuel assemblies, including MPC-LACBWR damaged fuel cans.

The Yankee-MPC system is designed to store Yankee class spent fuel with a maximum heat load of 12.5 kW (12.5 kW/36 assemblies = 0.347 kW per fuel assembly) and reconfigured fuel assemblies with a maximum heat load of 0.102 kW per assembly. The temperatures produced by the design basis fuel bound the temperature effects due to the reconfigured fuel assemblies.

The CY-MPC system is designed to store Connecticut Yankee spent fuel with a maximum total heat load of 17.5 kW, or an average heat load of 0.674 kW per assembly. The maximum heat load of a CY-MPC damaged fuel can, as well as CY-MPC reconfigured fuel assembly, is 0.674 kW.

The MPC-LACBWR system is designed to store Dairyland Power Cooperative La Crosse BWR spent fuel with a maximum total heat load of 4.5 kW, or an average heat load of 66.2 W per assembly for all locations with or without damaged fuel can confinement.

The thermal evaluation of the Yankee-MPC configuration for normal conditions of storage is presented in Section 4.4. The thermal evaluation for normal conditions of the CY-MPC configuration is presented in Section 4.5. The thermal evaluation for the MPC-LACBWR configuration for normal conditions is presented in Section 4.A.3 of Appendix 4.A.

4.1 Discussion

The significant thermal design feature of the NAC-MPC system is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the vertical concrete cask (storage cask) through four inlets. Heated air exits through the four outlets at the top of the storage cask. Radiant heat transfer also occurs from the canister shell to the concrete

cask liner. Consequently, the liner also heats the convective air flow. Conduction does not play a substantial role in heat removal from the canister surface. This natural circulation of air inside the vertical concrete cask (storage cask), in conjunction with radiation from the canister surface, maintains the fuel cladding temperature and all of the storage cask component temperatures below their design limits.

The thermal evaluation considers normal, off-normal, and accident conditions of storage. Each of these conditions can be described in terms of the environmental temperature, use of solar insolance, and the condition of the air inlets and outlets, as shown in Table 4.1-1. The design conditions for transfer for the Yankee-MPC and CY-MPC configurations are presented in Table 4.1-2.

This evaluation applies different component temperature limits and different material stress limits for long-term (steady-state) conditions and for short-term (transient) conditions. Normal storage is considered to be a steady-state condition. Off-normal and accident events, as well as the vacuum condition that temporarily occurs during the preparation of the canister while it is in the transfer cask and the time the canister is in the transfer cask while filled with helium, are evaluated as transient conditions. The maximum allowable material temperatures for long-term and for transient conditions are provided in Table 4.1-3.

Table 4.1-4 summarizes the results of the thermal evaluation of the Yankee-MPC system and presents the maximum temperatures for principal components. The thermal results and principal component temperatures for the CY-MPC system are shown in Table 4.1-5. Thermal results for the MPC-LACBWR system are presented in Appendix 4.A.

As shown in these tables, the calculated temperatures are well below the allowable component temperatures for normal (long-term) storage conditions and for short-term events.

Chapter 4
Appendix 4.A

APPENDIX 4.A THERMAL EVALUATION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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4.A THERMAL EVALUATION

4.A.1 Thermal Evaluation of the MPC-LACBWR System

The MPC Storage System for Dairyland Power Cooperative La Crosse Boiling Water Reactor, referred to as the MPC-LACBWR, is designed to safely store up to 68 Allis Chalmers (AC) and Exxon Nuclear (EN) undamaged and damaged spent fuel assemblies. Among the 68 loaded fuel assemblies, 32 of them can be loaded as damaged fuel assemblies in damaged fuel cans (DFCs).

The MPC-LACBWR system is designed to store the spent fuel with a maximum canister heat load of 4.5 kW, and each single fuel assembly with a maximum heat load of 66.2 W. The maximum heat load limit of 66.2 W per assembly also applies to the 32 DFCs in the outer slots of the basket.

The thermal evaluation of the MPC-LACBWR for normal conditions of storage is presented in Section 4.A.3.

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4.A.3 Thermal Evaluation of the MPC-LACBWR for Normal Conditions

The significant thermal design feature of the MPC-LACBWR system is the passive convective air flow up along the side of the canister. Cool (ambient) air enters at the bottom of the vertical concrete cask (storage cask) through four inlets. Heated air exits through the four outlets at the top of the storage cask. Radiant heat transfer also occurs from the canister shell to the concrete cask liner. In the thermal analyses performed in this appendix, the air in the inlets, annulus region and the outlets is conservatively not modeled. Consequently, the convection and conduction heat transfer due to the air flow and air are conservatively neglected. Radiation alone, from the canister to the concrete cask, maintains the fuel cladding temperature and all of the storage cask component temperatures below their design limits.

The design basis decay heat for the MPC-LACBWR is 4.5 kW with an axial distribution of the decay heat as shown in Figure 4.A.3-1. This distribution is applied to the fuel assemblies that are not in the 32 DFCs. The fuel in the 32 DFCs is considered to be compacted 50% and has an axially uniform heat load. The center of the compacted fuel in a DFC corresponds to the center of the basket. This position of the compacted fuel bounds any other possible configuration of the compacted fuel, as well as fuel in the DFC that is not damaged. A maximum decay heat for each fuel assembly of 66.2 W is applied to the basket.

The thermal evaluation considers normal, off-normal, and accident conditions of storage. Each of these conditions can be described in terms of the environmental temperature, use of solar insolance, and the condition of the air inlets and outlets, as shown in Table 4.A.3-1. The design conditions for transfer condition are presented in Table 4.A.3-2. Since no air or air flow is modeled between the concrete cask and the loaded canister, the analysis result for the normal storage condition with an ambient temperature of 75°F bounds the analysis results for the off-normal, half-inlet-blocked condition and the accident conditions of all vents block and the cask burial.

Normal storage is considered to be a steady-state condition. Off-normal and accident events, the vacuum condition that temporarily occurs during the preparation of the canister while it is in the transfer cask, and the time the canister is in the transfer cask while filled with helium are also evaluated as steady-state conditions due to the low heat load. The maximum allowable material temperatures for long-term and for transfer conditions are provided in Table 4.A.3-3.

Table 4.A.3-3 summarizes the results of the thermal evaluation of the MPC-LACBWR system and presents the maximum temperatures for principal components. As shown in the table, the calculated temperatures are well below the allowable component temperatures for normal storage and transfer conditions.

Table 4.A.3-1 Summary of Thermal Design Conditions for Storage

DESIGN CONDITION	ENVIRONMENTAL TEMPERATURE (°F)	SOLAR INSOLANCE (1)	STATUS OF INLETS AND OUTLETS
Normal	75	Yes	All open
Off-Normal (2)	75	Yes	Two inlets blocked
Off-Normal - Severe Heat	105	Yes	All open
Off-Normal - Severe Cold	-40	No	All open
Accident - Extreme Heat	125	Yes	All open
Accident (2)	75	Yes	All blocked
Accident (2) - Cask Burial Under Debris (3)	75	Yes	All blocked

(1) Solar Insolance per 10 CFR 71:
Curved Surface: 400 g cal/cm^2 (1475 Btu/ft^2) for a 12-hour period.
Flat Horizontal Surface: 800 g cal/cm^2 (2950 Btu/ft^2) for a 12-hour period.

(2) This condition is bounded by the normal condition since no air flow is modeled in the analysis for the normal condition.

(3) In the burial under debris condition, the inlets/outlets are blocked and, in addition, the solar insolance is included in the analysis. These are highly conservative assumptions.

Table 4.A.3-2 Summary of Thermal Design Conditions for Transfer Condition

CONDITION[1]	DURATION (Hours)
Design Basis Heat Load (kW)	4.5
Water Filled	600[2]
Vacuum Drying	600[2]
Canister filled with Helium	600[2]

(1) The canister is inside the transfer cask, with an ambient temperature of 75°F.

(2) Duration is unlimited. The 600 hours (25 days) is based on the 30-day time test for abnormal regimes as described in PNL-4835.

Table 4.A.3-3 Summary of Thermal Evaluation for the MPC-LACBWR System

Design Conditions	Material Temperature (°F)									
	Concrete[1]	Fuel Clad	6061- T651 Al Alloy[2]	NS-4-FR[3]	Lead[3]	SA693 630 SS[2]	SA240 SA182 304 SS[4]	SA240 SA182 304L SS[4]	A36 Steel[1]	A588/A350 LF2 Steel[3]
Allowable		SS								
Long-Term	150 (Bulk) 200 (Local)	806	650	300	600	650	800	800	700	700
Short-Term	350	806	700	300	600	800	800	800	700	700
Long-Term Conditions										
Normal (75°F Ambient)	133 (Bulk) 165 (Local)	449	441	N/A	N/A	443	412	356	165	N/A
Short-Term Conditions										
Off-Normal -Half Inlets Blocked (75°F Ambient)	168	449	441	N/A	N/A	443	412	356	168	N/A
Off-Normal -Severe Heat (105°F Ambient)	196	464	457	N/A	N/A	459	428	373	196	N/A
Off-Normal -Severe Cold (-40°F Ambient)	5	383	374	N/A	N/A	376	345	288	5	N/A
Accident -Extreme Heat (125°F Ambient)	228	476	469	N/A	N/A	470	440	384	228	N/A
Transfer -Vacuum Drying	N/A	329	321	140	141	323	288	236	N/A	156
Transfer -Backfilled with Helium	N/A	329	321	140	141	323	288	236	N/A	156

(1) Concrete cask components: Concrete cask and steel liner (ASTM A36). The bounding concrete temperatures for MPC-LACBWR (MPC FSAR Table 4.1-4) with the heat load of 12.5 kW are used.
(2) Fuel basket components: Heat transfer disks (6061- T651) and support disks (ASME SA 693, Type 630 stainless steel).
(3) Transfer cask components: Shells (ASTM A588); bottom doors (A350LF2), neutron shield (NS-4-FR); and, gamma shield (lead)/concrete cask lid shield.
(4) Canister components: Closure Lid (ASME SA 240/182, Type 304/304L stainless steel), shell (ASME SA 240, Type 304/304L stainless steel), bottom plate (ASME SA 240, Type 304/304L stainless steel), and damaged fuel can (ASME SA 240, Type 304 stainless steel).

4.A.3.1 MPC-LACBWR Thermal Models

As follows, the thermal evaluation of the MPC-LACBWR system utilizes seven finite element models, generated by the ANSYS program, for normal conditions of storage.

1. Three-Dimensional Concrete Cask and Canister Model
2. Three-Dimensional Transfer Cask and Canister Model
3. Two-Dimensional Fuel Model
4. Four Two-Dimensional Fuel Tube Models

The three-dimensional concrete cask and canister model includes: the concrete storage cask, fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, canister shell, lids and bottom plate. Based on symmetry, only one-fourth of the concrete cask and the loaded canister are modeled, and the plane of symmetry is considered to be adiabatic. The air in the inlets annulus between the canister and the concrete cask and the outlets is conservatively not modeled. A conservative distribution of the Boral plates is modeled with a total number of 24 Boral plates.

In the three-dimensional models, fuel regions and the fuel tubes are modeled using effective conductivities, which are determined by the two-dimensional fuel model and the two-dimensional fuel tube models. For the 32 DFCs, the fuel is considered to be 50% compacted, but it conservatively uses the same fuel conductivity as for the other 36 fuel assemblies. The compacted fuel would have a larger effective conductivity due to the increased contact between segments of the fuel rods. To determine the length over which the fuel assembly is distributed (in the DFC), the physical volume (V) of the fuel assembly is first computed using fuel weight data and material densities. One hundred percent (100%) of the fuel volume (V) corresponds to a length (L) determined by V/A, where A is the cross-section of the fuel can. A condition of 50% compaction is achieved when the modeled length (L_m) of the compacted fuel region is computed by L/(0.5). The 66.2 W per assembly is distributed over the modeled length (L_m) for each of the 32 DFCs. The center of the compacted fuel region in the DFC is positioned at the midplane of the basket and represents a system-bounding thermal configuration since the maximum temperature of the undamaged fuel occurs at the basket axial midplane. The conduction properties of the compacted region were conservatively taken to be the thermal properties of the undamaged fuel, since the undamaged fuel has lower conductance due to lack of contact of the fuel segments. Outside the compacted region, the volume inside the DFC was considered to be helium gas or water without any consideration of radiation across the DFC region.

The three-dimensional transfer cask and canister model comprise the transfer cask and the loaded canister. The canister and its internals are modeled the same as the three-dimensional concrete cask and the canister model. The model is used to perform steady-state analyses for the transfer condition when the canister is in the loading, vacuum drying, and helium backfill processes. Since the steady state temperatures calculated for water, vacuum and helium conditions are less than the limiting component allowable temperatures, no transient analyses are needed for the transfer conditions.

The effective conductivity of the fuel is determined using the two-dimensional fuel model, which is a detailed two-dimensional thermal model of the fuel assembly. The model includes the fuel pellets, the cladding, the helium gas between the fuel pellets and the fuel rod cladding, and the media occupying the space between the fuel rods. For normal operational conditions, the media between the fuel rods is considered to be helium. For the canister loading and transfer operations, the media is also considered to be water or vacuum. Based on Reference [A2], helium properties are used for the vacuum media since the pressure of the LACBWR during vacuum drying is in the low pressure range where helium gas thermal conductivity is not reduced as a variable of pressure.

The two-dimensional fuel tube models are used to determine the effective conductivities of the tube walls (with or without BORAL plate) and fuel tube walls (normal fuel tube or fuel tubes in the basket slots containing DFC). There are four models for the fuel tubes. The media is in the gaps between the layers comprising the fuel tube and BORAL.

These models are described in Sections 4.A.3.1.1 through 4.A.3.1.4.

Figure 4.A.3-1 Axial Power Distribution for MPC-LACBWR Fuel not in the DFC

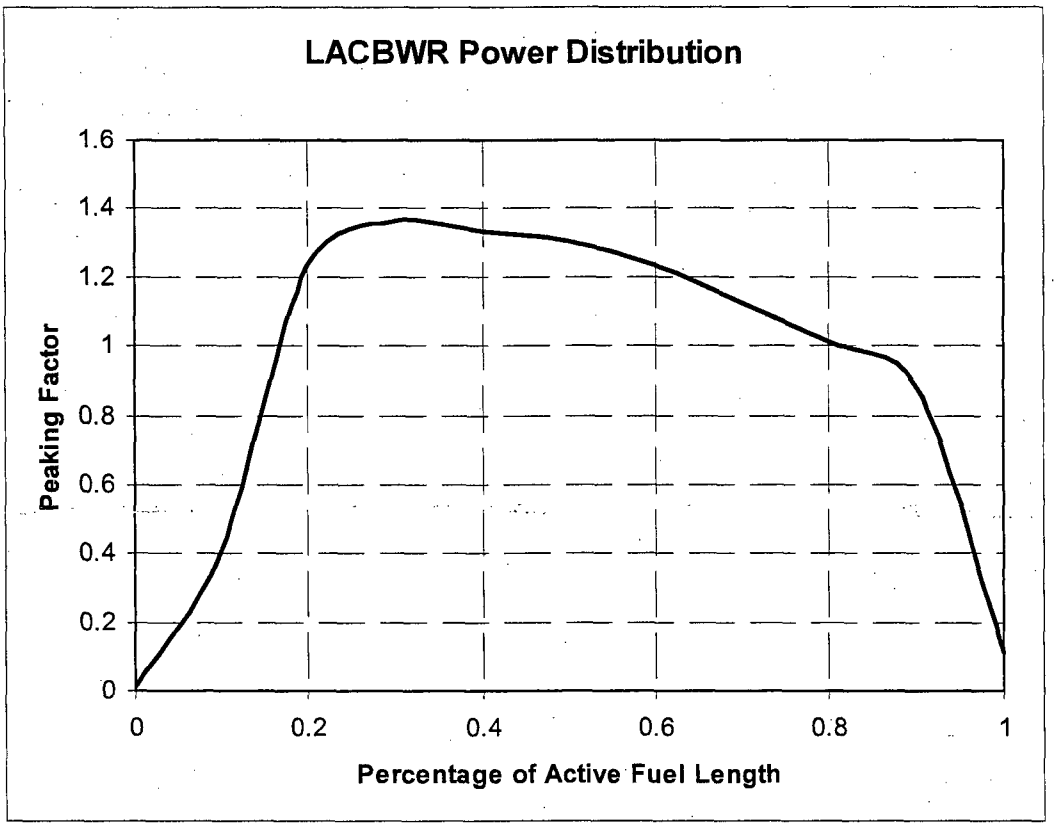
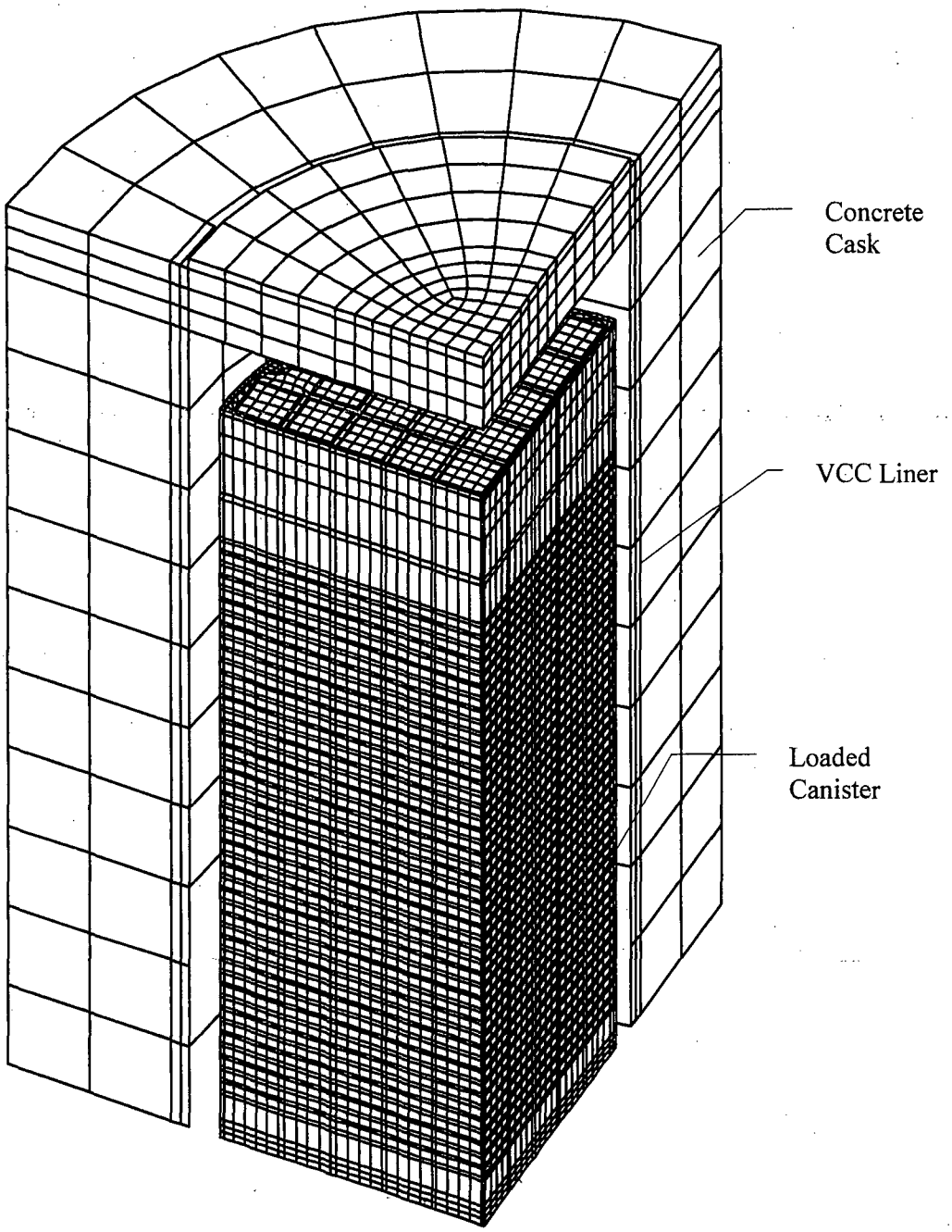


Figure 4.A.3-2 MPC-LACBWR Three-Dimensional Concrete Cask and Canister Model



4.A.3.1.2 Three-Dimensional MPC-LACBWR Transfer Cask and Canister Model

The three-dimensional MPC-LACBWR transfer cask and canister model is shown in Figures 4.A.3-5 and 4.A.3-6. ANSYS SOLID70 three-dimensional conduction elements and MATRIX50 radiation super elements are used to construct the model. The model includes the fuel assemblies, the fuel basket (fuel tubes, support disks, heat transfer disks, and top and bottom weldment plates), the canister (shell, bottom plate, and closure lid), the air gap between the canister and the transfer cask, and the transfer cask (top and bottom plates, inner and outer shells, neutron and gamma shields, and doors). The loaded canister in this model is modeled identically to the loaded canister model in the concrete cask and the canister model described in Section 4.A.3.1.1. Based on symmetry, only one-fourth of the transfer cask and canister are modeled, and the plane of symmetry is considered to be adiabatic.

A 0.43-inch air radial gap, based on the nominal dimensions of the canister shell and the transfer cask inner shell, is included. For normal transfer operations, when either vacuum or helium exists in the canister, only conduction and radiation are considered across the gaps inside the canister and convection is conservatively neglected. Conduction helium properties are utilized during the low-pressure drying process [A1]. When the canister is filled with water at the start of the transfer operation, natural circulation of the water is taken into account by adjusting the effective conductivities in the fuel and water regions based on a classical energy balance calculation of the canister content.

The heat load of 4.5kW is conservatively used in the analyses of the transfer conditions. For the heat load analyzed, a volumetric heat generation (Btu/hr-in^3) is applied to 36 fuel assemblies (not contained in a DFC) based on an active fuel length of 83 inches, along with an axial power distribution as shown in Figure 4.A.3-1. For the fuel assemblies in the 32 DFCs, the heat generation is applied to a length corresponding to 50% compaction, as described in Section 4.A.3.1. For the 32 DFCs, the heat generation is treated as being constant over the compacted region in the DFC.

Natural convection and radiation heat transfer modes are considered at the surfaces of the transfer cask and on the top of the canister lid. The same methodology as described in Section 4.A.3.1.1 for concrete cask surfaces is used. The same convection film correlation ($Hc = 0.00132\Delta T^{1/3}$ Btu/hr-in^2-°F) is used at the transfer cask outer surfaces and the canister lid. Solar insolance is applied at the surfaces of the transfer cask and on the top of the canister lid for helium condition, while solar insolance is not considered for the water condition since the draining process occurs inside a building. The same incident solar energy is applied as described in MPC FSAR Section

4.4.1.1. An ambient temperature of 75°F is assumed. The ambient temperature is applied to the bottom surface of the transfer cask doors as a fixed temperature boundary condition.

Since the steady-state temperatures calculated for water, vacuum and helium conditions are less than the limiting component allowable temperatures, transient analyses are not needed for the transfer conditions.

Table 4.A.3-4 MPC-LACBWR Maximum Component Temperatures for the Normal Conditions of Storage

Component	Maximum Temperature (°F)	Allowable Temperature (°F)(1)
Fuel Cladding	449	806
Aluminum Disk	441	700
Support Disk	443	650
Canister	356	800
Concrete Liner (steel)	165(2)	700
Concrete	165 (local)(2) 133 (bulk) (2), (3)	200 (local) 150 (bulk)

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.
(2) Concrete temperatures are from MPC FSAR Table 4.1-4, which is for a heat load of 12.5 kW.
(3) The average temperature of the concrete region is used as the bulk concrete temperature.

Table 4.A.3-5 MPC-LACBWR Maximum Component Temperatures for the Transfer Condition – Helium and Vacuum in Canister

Component	Maximum Temperature (°F)	Allowable Temperature (°F)(1)
Fuel Cladding	353	806
Gamma Shield (Lead)	166	600
Neutron Shield	165	300
Heat Transfer Disk (Aluminum)	344	700
Support Disk	346	800
Canister	261	800
Transfer Cask Inner Shell	180	700

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.

Table 4.A.3-6 MPC-LACBWR Maximum Component Temperatures for the Transfer Condition – Water in Canister

Component	Maximum Temperature (°F)	Allowable Temperature (°F)[1]
Fuel Cladding	144	806
Gamma Shield (Lead)	98	600
Neutron Shield	98	300
Heat Transfer Disk (Aluminum)	143	700
Support Disk	143	800
Canister	130	800
Transfer Cask Inner Shell	100	700

(1) The allowable temperatures are defined and referenced in MPC FSAR Table 4.1-3.

4.A.3.4 MPC-LACBWR Minimum Temperatures

MPC FSAR Section 11.1.4 provides the temperature distribution for the off-normal severe cold environmental conditions of -40°F. At this extreme condition, the components are above their minimum material limits.

4.A.3.5 Maximum MPC-LACBWR Internal Pressure for Normal Conditions

The MPC-LACBWR canister is backfilled with helium to atmospheric pressure (0.0 psig) and closed by welding. Normal condition pressure comprises the pressure due to the heating of the backfilled helium plus the pressure due to the postulated failure of 1 percent of the stored fuel rods with the subsequent release of 70 percent of the fission gas and all of the rod charge gas to the canister cavity, at temperature, from those failed rods. All of the gases, except the fission gases are helium. The total pressure for each volume is found by calculating the molar quantity of each gas and summing those directly. The calculated average normal condition temperature of the canister cavity gas is 359°F based on the thermal analysis results using the three-dimensional canister model described in MPC FSAR Section 4.4.1.2. The canister pressure is calculated using the Ideal Gas Law and applying a conservative average temperature of 370°F. The gas constant, R, is 0.0821 (atm x liters)/(Mole °K). The design basis fuel assembly for the internal pressure calculation is the Allis Chalmers (AC) assembly. This assembly has the highest burnup (22,000 MWd/MTU) and the highest fuel mass, and therefore generates the maximum fission gas quantities.

The number of moles of the canister and rod backfill gases is calculated using the Ideal Gas Law, PV = NRT. Backfill gas for the canister is assumed to be initially at 1 atmosphere absolute. The LACBWR fuel rod backfill pressure is also 1 atmosphere. The quantity of fission gas is derived from the SAS2H generated isotopics of the AC fuel assembly. For normal operating conditions, 1 percent of the fuel rods are assumed to fail, releasing 70 percent of their total fission gas and all of the backfill helium.

Rod Backfill Gas

The fuel rod plenum volume is:

$$V_1 = \pi r^2 L - \frac{M_{Spring}}{\rho}$$

where

M_{Spring} = the mass of the springs

ρ = the density of stainless steel.

The pellet clad gap volume is:

$$V_2 = \pi L\left(r^2_{Clad\ ID} - r^2_{Pellet\ OD}\right)$$

The total fuel rod backfill volume is:

$$V_{Rod\ Back\text{-}Fill} = V_1 + V_2$$

For the loaded canister, the total backfill gas volume is:

$$V = \text{Total Back-Fill}$$

$$= V_{Rod\ Back\text{-}Fill}\ \text{inches}^3 \times 100\frac{\text{Rods}}{\text{Assembly}} \times 68\frac{\text{Assemblies}}{\text{Canister}} \times \left(2.54\frac{\text{cm}}{\text{inch}}\right)^3 \times \frac{0.001\,\ell}{\text{cm}^3}$$

From the rod backfill volume and pressure, the quantity of rod backfill gas is calculated using the ideal gas law and a temperature of 68°F.

$$N = \frac{Pv}{RT}$$

$$N = 2.46\frac{\text{moles}}{\text{Canister}}$$

Conservatively rounded to:

$$N = 3\frac{\text{moles}}{\text{Canister}}$$

Fuel Fission Gas

The number of moles of fission gas per assembly is:

Isotope	Atomic Weight [grams/mole]	Mass [grams]	Number of Moles
H	3	0.0006	0.0002
HE	4	0.1360	0.0340
KR	82	0.0344	0.0004
KR	83	3.9400	0.0475
KR	84	9.5700	0.1139
KR	85	0.2190	0.0026
KR	86	16.3000	0.1895
I	127	3.0300	0.0239
I	129	13.9000	0.1078
XE	128	0.1090	0.0009
XE	129	0.0004	0.0000
XE	130	0.3210	0.0025
XE	131	38.7000	0.2954
XE	132	80.8000	0.6121
XE	134	122.0000	0.9104
XE	136	163.0000	1.1985
Total:	--	--	3.5396

For a full cask of 68 assemblies, the number of moles of fission gas per cask is 68 times that of the single assembly.

$$N = 68\frac{\text{Assemblies}}{\text{Cask}} \times 3.5396\frac{\text{Moles}}{\text{Assembly}} = 241\frac{\text{Moles of Fission Gas}}{\text{Cask}}$$

Canister (TSC) Backfill Gas

Canister backfill quantity is based on the canister free volume and backfill temperature. Backfill temperature applied is 68°F, which is conservative for a canister containing a heat-generating spent fuel payload.

$$V^{TSC}_{\text{Free Gas Volume}} = V_{\text{Canister}} - \left(V^{TSC}_{\text{Basket}} + V_{\text{Fuel}}\right)$$

$$V_{Canister} = \pi \frac{d^2}{4} \left(L_{Canister} - L_{TSC\ Bottom\ Plate}\right)$$

$$V_{Canister} = 411{,}160\ \text{inches}^3$$

Conservatively rounded to:

$$V_{Canister} = 410{,}000\ \text{inches}^3$$

$$V_{Basket}^{TSC} = 61{,}286.87\ \text{inches}^3$$

Conservatively rounded to:

$$V_{Basket}^{TSC} = 65{,}000.00\ \text{inches}^3$$

$$V_{Free\ Gas\ Volume}^{TSC} = 410{,}000 - \left(65{,}000\ \text{inches}^3 + 78{,}250\ \text{inches}^3\right)$$

$$V_{Free\ Gas\ Volume}^{TSC} = 266{,}750 \frac{\text{inches}^3}{\text{Canister}}$$

$$V_{Free\ Gas\ Volume}^{TSC} = 266{,}750 \frac{\text{inches}^3}{\text{Canister}} \times \frac{1\ \ell}{61.02\ \text{inches}^3} = 4{,}371 \frac{\ell}{\text{Canister}}$$

$$N = \frac{1\ \text{atm} \times 4{,}371\ \frac{\ell}{\text{Cask}}}{0.0821 \frac{\text{atm}\ \ell}{\text{Mole K}} \times 293\ \text{K}} = 181.62\ \frac{\text{Moles of TSC Backfill Gas}}{\text{Canister}}$$

Canister Pressure

The maximum normal operating pressure (MNOP) in the canister is calculated using the ideal gas law where:

$$N = N_{TSC\ Back\text{-}Fill} + 0.01\left(N_{Rod\ Back\text{-}Fill}\right) + 0.70(0.01)\left(N_{Fission\ Gas}\right)$$

$$N = 183.3 \frac{\text{Moles}}{\text{Canister}}$$

Therefore, the maximum normal operating condition canister internal pressure is:

$$P = \frac{\left(183.3\ \frac{\text{Moles}}{\text{Canister}}\right) \times \left(0.0821 \frac{\text{atm}\,\ell}{\text{mole K}}\right) \times 465\,\text{K}}{\left(4{,}371 \frac{\ell}{\text{Canister}}\right)} = 1.6\,\text{atm} \approx 23.5\,\text{psia} \approx 8.8\,\text{psig}$$

4.A.3.6 Maximum MPC-LACBWR Thermal Stresses for Normal Conditions

The canister and concrete storage cask thermal stresses are evaluated in MPC FSAR Section 3.4.4.3 and Section 3.4.4.4.

4.A.3.7 Evaluation of MPC-LACBWR Performance for Normal Conditions

As shown in the preceding sections, the MPC-LACBWR system operates within the thermal design limits. Therefore, no degradation due to temperature effects on material or components is expected over the lifetime of the cask.

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September 2009

Revision 09C

NAC-MPC

NAC Multi-Purpose Canister

FINAL SAFETY ANALYSIS REPORT

MPC-LACBWR Amendment

Volume 2 of 2

Docket No. 72-1025

NAC INTERNATIONAL

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

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Chapter 5
Appendix 5.A

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5.A.1 Shielding Discussion and Results for the MPC-LACBWR Storage System

This section provides a summary of the results of the shielding evaluation of the MPC-LACBWR system. Results are provided for the transfer cask and vertical concrete cask components.

The MPC-LACBWR storage system is comprised of a transportable storage canister, a transfer cask, and a vertical concrete storage cask. License drawings for these items are provided in Section 1.A.7. The transfer cask containing the canister and the basket is loaded under water in the spent fuel pool. Once filled with fuel, the closure lid is placed on top of the canister and the transfer cask is removed from the pool. After draining approximately 50 gallons of water from the canister, the closure lid is welded, the closure ring is inserted and welded, and the canister is drained and dried. Finally, the port covers are welded in place. The transfer cask is then used to transfer the canister to the storage cask where it is stored dry until transport. Shielding evaluations are performed for the transfer cask with both a wet and dry canister cavity as would occur during the welding of the closure lid. Shielding evaluations are performed for the storage cask with the cavity dry.

The MPC-LACBWR transfer cask has a multi-wall radial shield comprised of 0.75 inch of carbon steel, 3.5 inches of lead, 2 inches of solid borated polymer (NS-4-FR), and 1.25 inches of carbon steel. An additional 0.5 inch of stainless steel shielding is provided radially by the canister shell. Gamma shielding is provided primarily by the steel and lead layers, and neutron shielding is provided primarily by the NS-4-FR. The transfer cask bottom shield design is a solid section of 9.5 inches of low alloy steel. The top shielding is provided by the stainless steel closure lid, which is 7 inches thick. Temporary shielding may be used during welding, draining, drying, and helium backfill operations but is not credited in the shielding analysis. Temporary shielding is removed prior to storage.

The storage cask radial shield design is comprised of a 2.5-inch-thick carbon steel inner liner surrounded by a 22-inch thickness of concrete. Gamma shielding is provided by both the carbon steel and the concrete. Neutron shielding is provided primarily by the concrete. As in the transfer cask, an additional 0.5-inch thickness of stainless steel radial gamma shielding is provided by the canister shell. The storage cask top shielding design is comprised of 7 inches of stainless steel from the canister closure lid, 1.875 inches of carbon steel from the storage cask lid and 8 inches of concrete from the storage cask lid. Since the bottom of the storage cask sits on a concrete pad, the storage cask bottom shielding is comprised of the 1-inch thick stainless steel canister bottom plate, the 2-inch-thick carbon steel weldment base plate and its 0.25-inch-thick stainless steel cover, and the 1-inch thick carbon steel cask bottom plate. The base plate and bottom plate are

structural components that position the canister above the air inlets. The cask bottom plate supports the storage cask during lifting and forms the cooling air inlet channels at the cask bottom.

The MPC-LACBWR accommodates up to 68 stainless steel clad LACBWR spent fuel assemblies. LACBWR fuel assemblies were fabricated by two vendors, Allis Chalmers (AC) and Exxon Nuclear Company (ENC). The loading pattern evaluated for these fuel assemblies is illustrated in Figure 5.A.1-1. The physical parameters of the LACBWR fuel assemblies are presented in Table 5.A.2-1.

The fuel inventory at LACBWR contains a significant quantity of Allis Chalmers fuel classified as damaged due to concerns on clad stability. The MPC-LACBWR basket was, therefore, designed to contain up to 32 damaged fuel cans positioned in the peripheral locations in the basket. Due to criticality constraints, the Allis Chalmers fuel will not be permitted for loading in the canister interior locations. Fuel inventory at the LACBWR site allows for the split of Exxon fuel into the canister interior locations (Slot A in Figure 5.A.1-1) and Allis Chalmers fuel into the exterior locations (Slot B in Figure 5.A.1-1). The fuel inventory division resulted in initial shielding evaluations being based on undamaged Exxon fuel in the Slot A locations, with undamaged Allis Chalmers fuel in Slot B locations. The result of this evaluation is summarized in Section 5.A.1.1.

Undamaged Allis Chalmers fuel is evaluated, as the typical Allis Chalmers fuel assembly retains its nominal shape and retains the majority of the spent fuel within the fuel rod clad. As damaged fuel may not retain its geometry during transfer and storage operations, Section 5.A.1.2 results include dose rates for reconfigured/damaged Allis Chalmers fuel in Slot B locations. To allow for the contingency of loading Exxon fuel into an exterior slot/damaged fuel can, Section 5.A.4.7 provides the justification that the relative source change and, therefore, dose effects are minor. While this discussion is limited to one Exxon fuel assembly per basket, the minor change in source is not considered significant and additional Exxon fuel assemblies may be placed into the basket periphery.

The MPC-LACBWR damaged fuel can may hold a complete fuel assembly. Since the shielding evaluation conservatively assumes that the damaged fuel cans are not present in the canister, the additional shielding provided by the wall of the can would serve to reduce external dose rates.

Shielding evaluations of the MPC-LACBWR transfer and storage casks are performed using the MCNP5, Release 1.30, Monte Carlo transport code [A1]. Fuel source terms are developed using the SCALE isotopics sequence SAS2H (Herman). Source terms include fuel neutron, fuel

gamma, fuel n-gamma, and activated hardware gamma. Dose rate evaluations include the effect of axial fuel burnup peaking on fuel neutron and gamma source terms.

The resulting dose rate profiles, along with the maximum and average radial and axial dose rates are presented for the storage cask and transfer cask analyses in Section 5.A.4.

5.A.1.1 Undamaged Fuel Dose Rates

The maximum dose rates for the storage cask with undamaged fuel are summarized in Table 5.A.1-1 and Table 5.A.1-2. The standard deviation resulting from the Monte Carlo evaluation used by MCNP (1σ) is indicated in the tables. The storage cask maximum side dose rate is 28.9 (1.3%) mrem/hr slightly below the fuel midplane elevation. The maximum storage cask top axial surface dose rate is 18.7 (6.9%) mrem/hr on the top lid surface just above the annulus between the canister and the storage cask liner. Since the storage cask is vertical during normal storage operation, the bottom is inaccessible. Therefore, no bottom axial dose rates are presented. The average dose rates at the inlets and outlets are 38.3 (1.4%) mrem/hr and 2.0 (0.5%) mrem/hr, respectively.

Under accident conditions involving a projectile impact and a loss of 6 inches of concrete, the surface dose rate increases to 278 mrem/hr at the impact location and 105 mrem/hr at a distance of 1 meter from the surface. There are no design basis accidents that result in a tip-over of the MPC-LACBWR storage cask.

The maximum dose rates for undamaged fuel in the transfer cask for the wet and dry canister cavity configurations encountered during canister closure operations are presented in Table 5.A.1-3 and Table 5.A.1-4, respectively.

With a wet canister cavity (no port covers), the maximum dose rates are 68.2 (2.3%) and 471.2 (1.7%) mrem/hr on the radial surface and top axial surface, respectively. The bottom surface of the cask has a maximum dose rate of 23.8 (2.2%) mrem/hr.

With a dry canister cavity (port covers installed), the maximum dose rates are 102.2 (5.2%) mrem/hr and 598.7 (1.4%) mrem/hr on the radial and top axial surfaces, respectively. The bottom surface of the cask has a maximum dose rate of 54.2 (2.2%) mrem/hr.

5.A.1.2 Damaged Fuel Dose Rates

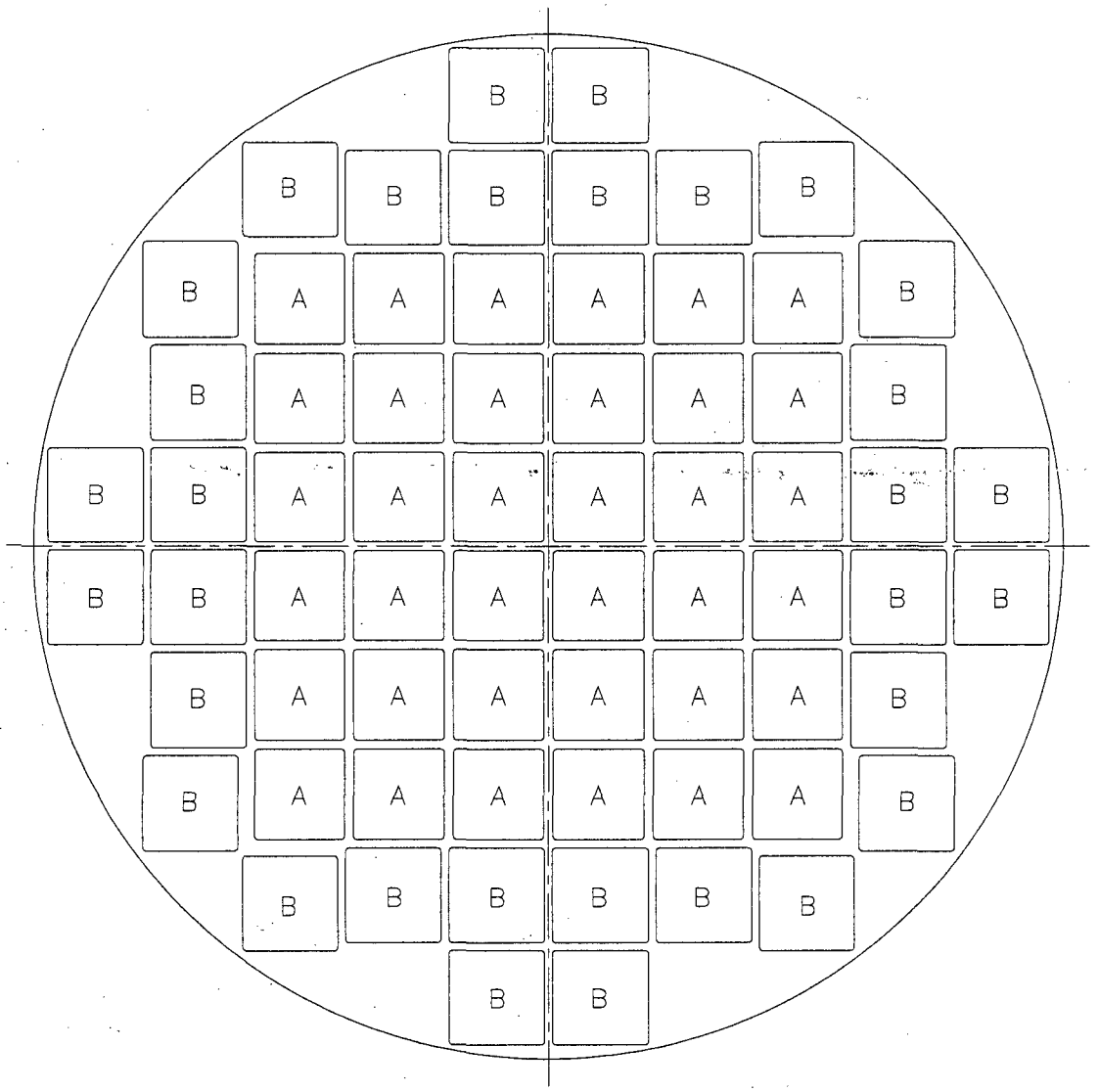
To ensure that the worst case configuration is considered, two damaged fuel scenarios are evaluated for the 32 peripheral basket locations.

The first scenario assumes the damaged fuel collects over the active fuel length of the fuel assembly. This scenario is modeled by filling the fuel rod interstitial volume with UO_2 and increasing the fuel neutron, gamma, and n-gamma source consistent with this increase in mass. A comparison of dose rate profiles for the 68 assembly intact fuel results and 36 intact and 32 damaged assemblies in Section 5.A.4 demonstrates that the damaged fuel model dose rates are less due to the increase in self-shielding from the 32 peripheral assemblies compensating for the increase in source strength.

In the second scenario, damaged fuel is assumed to migrate from the active fuel into the lower end fitting region of the fuel assembly, filling all the modeled void space. However, no credit is taken for the reduction in lower end fitting hardware dose rate due to the added UO_2 mass and self-shielding nor for the reduction in fuel mass migrated from the active fuel region. In this case, storage cask inlet and transfer cask bottom surface dose rates increase due to the addition of damaged fuel. The storage cask inlet dose rate increase is 36.7 mrem/hr, effectively doubling the air inlet dose rate. The transfer cask bottom axial dose rate increase is 22.1 mrem/hr, increasing the bottom axial dose rate by approximately 41%. Note that the radial location of the maximum dose rate at the bottom of the transfer cask differs between the undamaged and damaged fuel models.

Damaged fuel maximum dose rates are summarized in Table 5.A.1-5 and Table 5.A.1-6.

Figure 5.A.1-1 MPC-LACBWR Loading Pattern



Slot A: Undamaged Exxon fuel

Slot B: Undamaged or damaged Exxon or Allis Chalmers fuel

Fuel Type	Max. Burnup [MWd/MTU]	Min. Initial Enrichment [wt % ^{235}U]	Min. Cool Time [years]
Allis Chalmers	22,000	3.6	28
Exxon Nuclear Company	21,000	3.6	23

Table 5.A.1-1 Summary of MPC-LACBWR Storage Cask Maximum Dose Rates – Undamaged Fuel

Location	Source	Surface		1 meter	
		mrem/hr	FSD	mrem/hr	FSD
Top Axial	Neutron	0.5	6.9%	0.1	6.8%
	Gamma	18.2	7.1%	5.5	3.6%
	Total	18.7	6.9%	5.6	3.5%
Side (Normal)	Neutron	0.2	2.0%	0.1	1.5%
	Gamma	28.7	1.3%	11.4	0.9%
	Total	28.9	1.3%	11.5	0.9%
Side (Accident)	Neutron	0.7	1.7%	0.2	1.4%
	Gamma	277.1	1.1%	104.5	0.8%
	Total	277.8	1.1%	104.7	0.8%

Table 5.A.1-2 Summary of MPC-LACBWR Storage Cask Inlet/Outlet Maximum Dose Rates – Undamaged Fuel

Source	Inlet Average		Outlet Average	
	mrem/hr	FSD	mrem/hr	FSD
Fuel Neutron	1.3	1.4%	0.1	1.2%
Fuel Gamma	6.8	6.5%	0.1	7.6%
Fuel Hardware	3.2	3.6%	0.1	2.6%
Lower End Fitting	27.0	0.9%	--	--
Upper Plenum	--	--	1.2	0.4%
Upper End Fitting	--	--	0.5	0.5%
Total	38.3	1.4%	2.0	0.5%

Table 5.A.1-3 Summary of MPC-LACBWR Transfer Cask Maximum Dose Rates – Wet (Port Covers Off) – Undamaged Fuel

Location	Source	Surface		1 meter	
		mrem/hr	FSD	mrem/hr	FSD
Top	Neutron	0.2	8.5%	0.1	6.9%
	Gamma	471.0	1.7%	205.7	2.0%
	Total	471.2	1.7%	205.8	2.0%
Side	Neutron	22.0	4.9%	4.7	3.7%
	Gamma	46.2	2.5%	16.7	1.5%
	Total	68.2	2.3%	21.4	1.4%
Bottom	Neutron	0.1	13.6%	0.1	10.4%
	Gamma	23.7	2.2%	10.0	1.5%
	Total	23.8	2.2%	10.1	1.4%

Table 5.A.1-4 Summary of MPC-LACBWR Transfer Cask Maximum Dose Rates – Dry (Port Covers On) – Undamaged Fuel

Location	Source	Surface		1 meter	
		mrem/hr	FSD	mrem/hr	FSD
Top	Neutron	7.9	3.1%	2.4	2.5%
	Gamma	590.8	1.4%	251.7	1.4%
	Total	598.7	1.4%	254.1	1.4%
Side	Neutron	19.6	3.3%	4.4	2.9%
	Gamma	82.6	6.4%	31.1	2.7%
	Total	102.2	5.2%	35.5	2.4%
Bottom	Neutron	17.6	2.0%	4.9	1.2%
	Gamma	36.6	3.1%	15.4	1.1%
	Total	54.2	2.2%	20.3	0.9%

Table 5.A.1-5 Summary of MPC-LACBWR Storage Cask Maximum Dose Rates – Damaged Fuel

Source	Inlet Average	
	mrem/hr	FSD
Undamaged Fuel	38.3	1.4%
Damaged Neutron	2.3	0.3%
Damaged Gamma	34.4	3.5%
Total	75.0	1.8%

Table 5.A.1-6 Summary of MPC-LACBWR Transfer Cask Maximum Dose Rates – Dry (Port Covers On) – Damaged Fuel

Location	Source	Surface	
		mrem/hr	FSD
Bottom	Undamaged Fuel	45.1	0.8%
	Damaged Neutron	28.5	0.4%
	Damaged Gamma	2.6	8.6%
	Total	76.2	0.6%

5.A.3 MPC-LACBWR Model Specification

The transfer and concrete casks are evaluated using the MCNP three-dimensional Monte Carlo code. In the MCNP fuel assembly model, the fuel and hardware source regions are homogenized within a volume defined by the fuel assembly width and height. This volume is subdivided axially into active fuel, upper plenum, and upper and lower end fitting source regions. Within these axial volumes, the material masses of the fuel assembly are homogenized. In all models, the cask and TSC shield thicknesses and axial extents are explicitly represented, including streaming paths. Surface detectors are used to estimate the dose profiles at the cask surface and at distances of 1 foot, 1 meter, 2 meter, and 4 meters from the cask surface. The MCNP code employs an automated biasing technique for the Monte Carlo calculation based on weight window adjustments in mesh cells. Radial biasing is performed to estimate dose rates at the transfer cask radial surface and concrete cask radial surface, including air inlets and outlets. Axial biasing is used for cask top and bottom surface dose rates. Angular biasing components are used to capture azimuthal variations in the concrete cask air inlets and outlets.

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

5.A.3.1 MPC-LACBWR Fuel Assembly Model

Based on the fuel assembly parameters provided in Table 5.A.2-1 and Table 5.A.2-6, homogenized treatments are developed for fuel assembly source regions. The homogenized fuel assembly is represented in the model as a stack of boxes with width equal to the fuel assembly width. The height of each box corresponds to the modeled height of the corresponding assembly region, as shown in Figure 5.A.2-1. The fuel model is based on the Allis Chalmers radial and axial geometry. The homogenization of nonfuel regions (upper end fitting, upper plenum, and lower end fitting) is based on the minimum hardware masses of the two designs.

The active fuel region homogenization is shown in Table 5.A.3-1. The interstitial material is a void under dry canister conditions and water (density 0.9982 g/cm^3) under wet canister conditions. The clad region is stainless steel (density 7.94 g/cm^3).

Assembly hardware regions are homogenized as shown in Table 5.A.3-2 based on the minimum regional masses from Table 5.A.2-6. The only material included in the homogenized region is stainless steel. Volume fractions of material are based on the modeled region volume and the volume of stainless steel present as computed from the modeled mass and density (7.94 g/cm^3).

The homogenized fuel region model is applied to both wet and dry cask cavity cases. To address changes in interaction of thermalized neutrons in the wet canister cases and to demonstrate that the evaluated configuration is conservative, a comparison of a discretely modeled fuel rod array versus homogenized material was perfomed. For the discretely modeled array, the neutron multiplication option of MCNP was activated (i.e., the NONU option was removed from the input file, along with the subcritical multiplication factor applied to the source term). The results of this analysis demonstrated an 80% reduction in calculated neutron and n-gamma dose for the explicitly (discretely) modeled fuel rod array (e.g., average neutron dose decreased from 7.8 mrem/hr in the homogenized case to 1.3 mrem/hr in the discretely modeled rod array).

5.A.3.2 MPC-LACBWR Canister and Basket Model

The MCNP description of the canister and basket elements forms a common sub-model employed in all storage and transfer cask analyses. The key features of the basket model are the detailed representation of fuel tubes, basket support and heat transfer disks, and weldment structures. The key feature of the canister model is the inclusion of the vent and drain ports in the canister shield lid.

The vent and drain ports in the canister shield lid are modeled as a series of three stacked cylinders. The port cover is also modeled, but may or may not be in place depending on the particular operational condition specified. In the axial analysis of the transfer cask, the vent and drain port covers are assumed to be installed when the canister is in a dry condition, and removed when the canister is modeled in a wet condition. Port covers are in place in all storage cask model analyses. The port cover is modeled as a solid piece of stainless steel.

The basket model conservatively omits the uppermost and lowermost support disks and models 33% less absorber sheets than specified in the License drawings.

The closure lid diameter is modeled flush with the canister inner shell diameter, not modeling the gap between the closure lid diameter and the canister shell inner diameter. However, radiation streaming through this gap is not a concern due to the following factors: a) the lid support ring will reduce streaming, b) the lid is welded to the canister shell, and c) the closure ring is welded to the canister shell and the closure lid.

The three-dimensional canister/basket model is shown in Figure 5.A.3-1.

5.A.3.3 MPC-LACBWR Transfer Cask Model

In order to estimate occupational dose rates associated with the canister sealing operation, two operational configurations of the transfer cask are considered for the three-dimensional model of the upper cask region. These include wet canister conditions without port covers and dry canister conditions with port covers.

The top configuration of the transfer cask is evaluated in detail for the welding, draining, and drying operations. Model features include:

- Vent and drain port openings in the canister shield lid.
- Upper trunnions cut through the radial shield and extend from the inner shell to the outer shell. No credit for the radial extent of the trunnions outside the cask outer shell is taken.
- Lead and neutron shielding overlap at the top as per the transfer cask drawings.
- Termination of the radial shields at the bottom plate.
- An explicit model of the bottom door assembly.

Details of the elevations and radii used in creating the three-dimensional top model are taken directly from the drawings in Section 1.A.7. Elevations associated with the transfer cask three-dimensional features are established with respect to the center bottom of the canister bottom plate for the MCNP combinatorial geometry model. The three-dimensional transfer cask model is shown in Figure 5.A.3-2.

5.A.3.4 MPC-LACBWR Storage Cask Model

The three-dimensional model of the vertical concrete cask is based on the following features of the storage cask:

- Heat transfer annulus.
- Carbon steel weldment with cutouts for inlets and outlets.
- Concrete shield with rebar.
- Inlet and outlet models including carbon steel channel walls.
- Concrete and carbon steel top lid.
- Carbon steel bottom base plate and cover.
- Carbon steel support stand with four cutouts for air flow.
- Carbon steel storage cask bottom.
- Concrete pad below base plate.

Detailed model parameters used in creating the three-dimensional model are taken directly from the drawings in Section 1.A.7. Elevations associated with the concrete cask three-dimensional features are established with respect to the center bottom of the canister for the MCNP combinatorial model. The three-dimensional concrete cask model is shown in Figure 5.A.3-3.

A detailed sketch of the cask outlet model is shown in Figure 5.A.3-4 in terms of labeled key points in the model. Dimensions for the key points are shown in Table 5.A.3-3.

The bottom weldment model is detailed in Figure 5.A.3-5. The figure shows two sections through the bottom weldment assembly model. The portion of the figure to the left of the centerline is a section through an inlet location. The portion to the right of the centerline is a section through a non-inlet location.

5.A.3.5 MPC-LACBWR Detector Mesh Definition

MCNP surface detectors are used to calculate dose rates at various distances from the casks. The surface tallies are subdivided using the FS tally segmentation card. A graphical illustration of the detector overlay on a cask is shown in Figure 5.A.3-6. Depicted are 1 foot, 1 meter, 2 meter, and 4 meter detector surfaces on the concrete cask. For clarity, the cask surface detector and azimuthal (angular) divisions are not shown. Typical detector grids for the transfer and concrete cask analysis are shown in Table 5.A.3-4 through Table 5.A.3-6. The dose rate maps produced by this method completely enclose the accessible cask surfaces and capture all locations necessary for the evaluation of occupational exposures.

5.A.3.6 MPC-LACBWR Shield Regional Densities

Based on the homogenization described in Section 5.A.3.1, the resulting active fuel regional densities are shown in Table 5.A.3-7. Material descriptions are shown for both wet and dry canister conditions. The upper end fitting is modeled as dry under all scenarios to simulate the partial draindown that occurs prior to welding operations.

Material compositions for the remaining structural and shield materials are shown in Table 5.A.3-8.

5.A.4 MPC-LACBWR Storage System Shielding Evaluation

Shielding calculations are performed for the transfer cask and storage cask designs using the source terms from the two design basis fuel types described in Section 5.A.2: Allis Chalmers fuel at 22,000 MWd/MTU and 28 years cool time and Exxon Nuclear Company fuel at 21,000 MWd/MTU and 23 years cool time. The Allis Chalmers fuel source is modeled in the 32 peripheral basket locations and the Exxon Nuclear Company source is modeled in the 36 interior basket locations. The resulting dose rate profiles are reported as a function of distance from the radial and axial surfaces of the MPC-LACBWR storage cask and transfer cask.

5.A.4.1 MPC-LACBWR Calculational Methods

The shielding evaluations of the transfer and concrete cask are performed with MCNP5 [A1]. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. As described in Section 5.A.2.3, these evaluations include the effect of fuel burnup peaking on fuel neutron and gamma source terms.

The MCNP shielding models described in Section 5.A.3 are used with the source terms described in Section 5.A.2 to estimate the dose rate profiles at the cask surface and at distances of 1 foot, 1 meter, 2 meters, and 4 meters. The method of solution is continuous energy Monte Carlo, with a Monte Carlo based weight window generator to accelerate code convergence. Pointwise, continuous energy, neutron libraries generated from ENDF/B-V and VI data [A1] and the MCPLIB04 photoatomic [A1] libraries are employed in the MCNP shielding analysis. Default libraries for each material are obtained from the MCNP XSDIR file and are isotope/element dependent (e.g., the latest ^{10}B data released with the code version used in the analysis is ENDF/B-VI Release 6, 5010.66c, while the ^{238}U data is obtained from ENDF/B-VI Release 9, 92238.69c). Radial or axial biasing is performed, depending on the desired dose location. Azimuthal components are included in the weight window mesh to account for the angular variations in the bulk shielding properties of the concrete cask at the inlets and outlets and at the TSC lid ports (transfer evaluation only).

Significant validation literature is available for MCNP, as it is an industry standard tool for spent fuel cask evaluations. Available literature covers a range of shielding penetration problems ranging from slab geometry to spent fuel cask geometries [A2-A6]. Confirmatory calculations against other validated shielding codes [SCALE (ORNL) and MCBEND (Serco Assurance)] on NAC casks have further validated the use of MCNP for shielding evaluations.

5.A.4.2 MCNP Flux-to-Dose Rate Conversion Factors

The ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors are used in all cask shielding evaluations. Neutron and gamma dose conversion factors are listed in Table 5.A.4-1 and Table 5.A.4-2, respectively.

5.A.4.3 MPC-LACBWR Storage Cask Three-Dimensional Dose Rates

5.A.4.3.1 Undamaged Fuel

Storage cask three-dimensional model side dose rates are presented in Figure 5.A.4-1. The figure shows the dose rate profiles at various radial distances from the cask surface. Figure 5.A.4-2 shows the contributions to the radial surface dose rate from each source component. Figure 5.A.4-3 shows the corresponding top axial dose rate profile at various axial distances from the cask surface and Figure 5.A.4-4 gives the contributions to the top axial surface dose rate from each source type. The azimuthal dose profile at the air inlet elevation is shown in Figure 5.A.4-5. Similarly, the azimuthal dose rate profile at the cask air outlets is shown in Figure 5.A.4-6.

The maximum radial and top axial surface dose rates for the storage cask are 28.9 (1.3%) and 18.7 (6.9%) mrem/hr, respectively. At the inlet locations, the average dose rate of the four symmetric inlet openings is 38.3 (1.4%) mrem/hr. At the outlets, the average dose rate at the four symmetric outlet locations is 2.0 (0.5%) mrem/hr.

Under accident conditions involving a projectile impact and a loss of 6 inches of concrete, the surface dose rate increases to 278 mrem/hr at the impact location and 105 mrem/hr at a distance of 1 meter from the surface.

5.A.4.3.2 Damaged Fuel

To ensure that the worst case configuration is considered, two damaged fuel scenarios are evaluated for the 32 peripheral basket locations.

The first scenario assumes the damaged fuel collects over the active fuel length of the fuel assembly. This scenario is modeled by filling the fuel rod interstitial volume with UO_2 and increasing the fuel neutron, gamma, and n-gamma source consistent with this increase in mass. A comparison of dose rate profiles for the 68 assembly intact fuel results and 36 intact and 32 damaged assemblies, shown in Figure 5.A.4-7 and Figure 5.A.4-8, demonstrates that the damaged fuel model dose rates are less due to the increase in self-shielding from the 32 peripheral assemblies compensating for the increase in source strength.

In the second scenario, damaged fuel is assumed to migrate from the active fuel into the lower end fitting region of the fuel assembly, filling all the modeled void space. In this case, the combined gamma and neutron sources result in a maximum radial dose rate of 4.3 mrem/hr at the bottom of the cask (average increase of 1.0 mrem/hr), shown in Figure 5.A.4-9. The inlet dose rate increase, as shown in Figure 5.A.4-10, is 36.7 mrem/hr, effectively doubling the air inlet dose rate. However, no credit is taken for the reduction in lower end fitting hardware dose rate due to the added UO_2 mass and self-shielding nor for the reduction in fuel mass migrated from the active fuel region.

5.A.4.4 MPC-LACBWR Transfer Cask Three-Dimensional Dose Rates

The MPC-LACBWR transfer cask is analyzed under wet and dry conditions corresponding with the operational configurations associated with loading the canister with spent fuel assemblies in a fuel pool environment. Under wet conditions, the canister is assumed to be filled with water to the top of the upper plenum region. This distance accounts for the removal of water from the canister prior to welding the closure lid to the canister in order to prevent water contamination of the closure lid weld.

Transfer cask dose rates are calculated for two lid configurations. The first configuration consists of the closure lid without port covers with the canister in the wet condition. The second configuration consists of the closure lid with the canister in the dry condition and port covers are installed.

5.A.4.4.1 Undamaged Fuel

For a wet canister cavity and no port covers, dose rate profiles are shown in Figure 5.A.4-11 through Figure 5.A.4-16. The maximum dose rates are 68.2 (2.3%) and 471.2 (1.7%) mrem/hr on the radial surface and top axial surface, respectively. The bottom surface of the cask has a maximum dose rate of 23.8 (2.2%) mrem/hr.

For a dry canister cavity and installed port covers, dose rate profiles are shown in Figure 5.A.4-17 through Figure 5.A.4-22. The maximum dose rates are 102.2 (5.2%) mrem/hr and 598.7 (1.4%) mrem/hr on the radial and top axial surfaces, respectively. The bottom surface of the cask has a maximum dose rate of 54.2 (2.2%) mrem/hr.

5.A.4.4.2 Damaged Fuel

The same damaged fuel scenarios were analyzed for the transfer cask under dry conditions as those described for the storage cask. Like the storage cask (scenario 1), a comparison of dose rate profiles for the 68 assembly intact fuel results and 36 intact and 32 damaged assemblies, shown

in Figure 5.A.4-23 through Figure 5.A.4-25, demonstrates that the damaged fuel model dose rates are less due to the increase in self-shielding from the 32 peripheral assemblies compensating for the increase in source strength. In the second scenario (damaged fuel in the lower end fitting region), the combined gamma and neutron sources result in a maximum radial dose rate of 23.4 mrem/hr at the bottom of the cask (average increase of 3.7 mrem/hr), shown in Figure 5.A.4-26. The bottom axial dose rate increase is 22.1 mrem/hr, shown in Figure 5.A.4-27, increasing the bottom axial dose rate by approximately 41%. As stated previously, no credit is taken for the reduction in lower end fitting hardware dose rate due to the added UO_2 mass and self-shielding nor for the reduction in fuel mass migrated from the active fuel region.

5.A.4.5 Partial Flooding Evaluation

To confirm that the analyzed dry and wet canister conditions bound a partial flooding scenario, additional MCNP inputs were setup to model a canister filled up to 70% of the active fuel region. The results in Figure 5.A.4-28 and Figure 5.A.4-29 demonstrate that the dry canister configuration produces bounding transfer cask dose rates.

5.A.4.6 Validation of Fresh Fuel Material Composition

The validity of the dose rate results summarized herein is based on the ability to apply fresh fuel material composition based MCNP results to spent fuel. To confirm the accuracy of this assumption, radial dose rates are calculated using the SAS2H-calculated isotopic composition of the design basis Allis Chalmers fuel assembly (22,000 MWd/MTU, 3.6 wt % ^{235}U, and 28 years cool time) in the transfer and storage casks and compared to the fresh fuel results. Radial dose rate profiles for fresh and spent fuel isotopics are shown in Figure 5.A.4-30 and Figure 5.A.4-31 and demonstrate the acceptability of the fresh fuel assumption (i.e., there is no significant dose change associated with the fresh fuel model).

5.A.4.7 Justification of Exxon Fuel in DFCs

In the event that an Exxon Nuclear Company fuel assembly is found to be damaged, the assembly may be loaded in a DFC location rather than a standard location. A bounding scenario, under which a basket is loaded with 31 Allis Chalmers assemblies and 37 Exxon Nuclear Company assemblies (36 undamaged and 1 damaged), the change in 68-assembly source is as follows.

Source	31/37 vs. 32/36 Change
Fuel Neutron	-0.5%
Fuel N-Gamma	-0.5%
Fuel Gamma	0.0%
Fuel Hardware	1.0%
Lower End Fitting	1.3%
Upper Plenum	0.4%
Upper End Fitting	0.4%

While a slight azimuthal increase in gamma dose rate may be present at the radial location of the DFC under this scenario, there is no significant change in cask average dose rates when loading Exxon fuel assemblies into DFCs.

Figure 5.A.4-1 MPC-LACBWR Storage Cask Radial Dose Rate Profiles – Undamaged Fuel

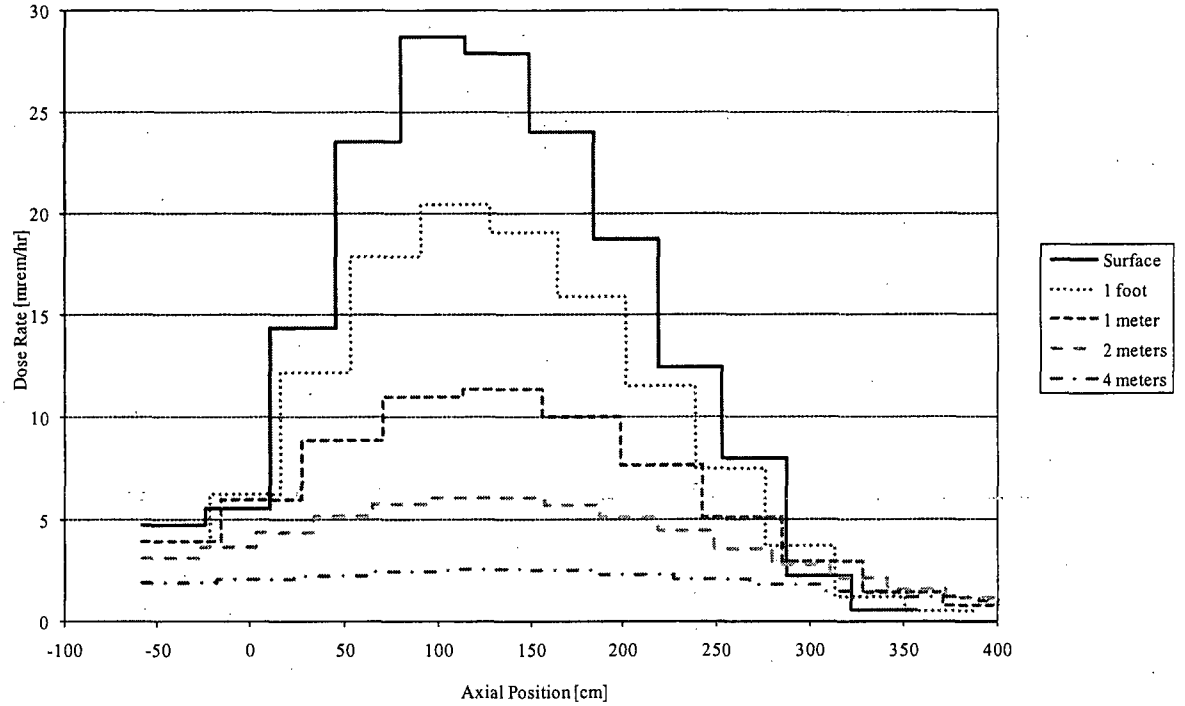
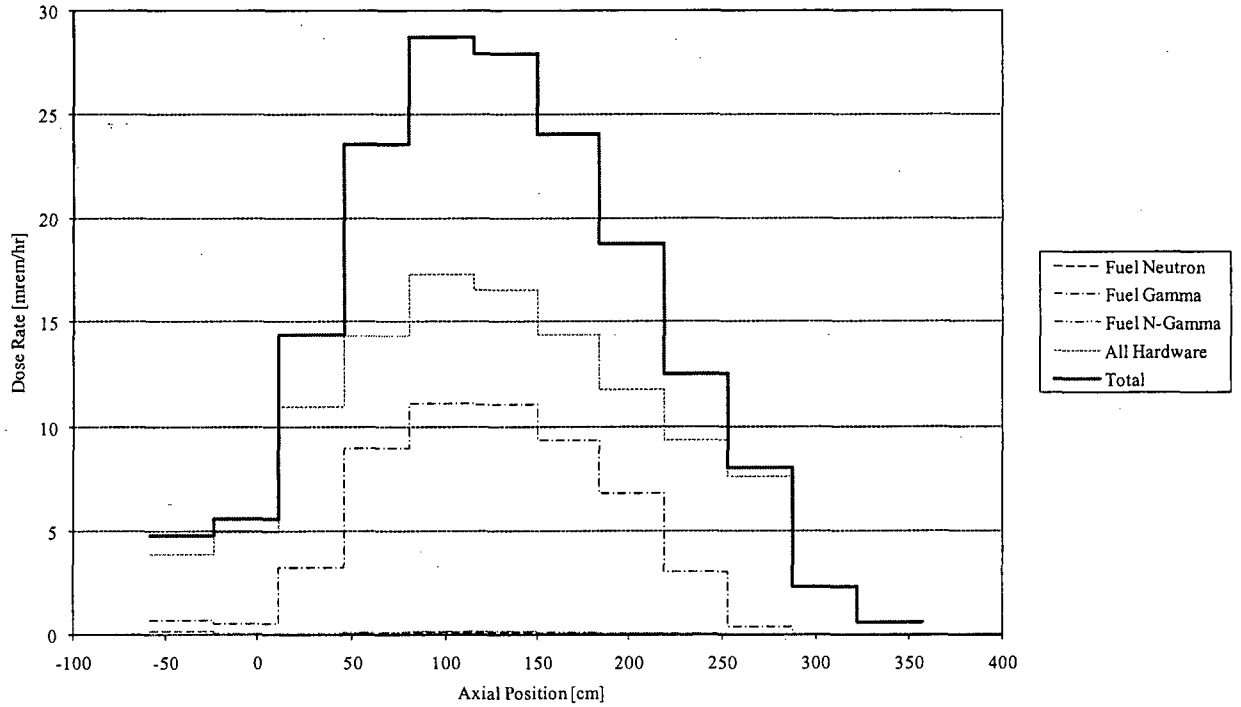


Figure 5.A.4-2 MPC-LACBWR Storage Cask Radial Surface Dose Rate Profile by Source Type – Undamaged Fuel



Chapter 6
Appendix 6.A

Appendix 6.A CRITICALITY EVALUATION – MPC-LACBWR MPC STORAGE SYSTEM FOR DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR

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6.A.1 Discussion and Results

The cask system consists of a TSC (Transportable Storage Canister), a transfer cask, and a concrete cask. The system is designed to safely store up to 68 LACBWR fuel assemblies of which up to 32 may be classified as damaged and be placed into damaged fuel cans (DFCs). The TSC is comprised of a stainless steel canister and a basket within which fuel is loaded. The DFC provides a screened container to prevent gross fissile material release into the TSC cavity from failed fuel rod clad. The TSC is loaded into the concrete cask for storage. A transfer cask is used for handling the TSC during loading of spent fuel. Fuel is loaded into the TSC contained within the transfer cask underwater in the spent fuel pool. Once loaded with fuel, the TSC closure lid is welded and the TSC is drained, dried and backfilled with helium. The transfer cask is then used to move the TSC into or out of the concrete cask. The transfer cask provides shielding during the TSC loading and transfer operations.

Under normal conditions, such as loading in a spent fuel pool, moderator (water) is present in the TSC during the initial stages of fuel transfer. During draining and drying operations, moderator with varying density is present. Thus, the criticality evaluation of the transfer cask includes a variation in moderator density and a determination of optimum moderator density. Normal, off-normal, and accident condition optimum moderator density studies cover pellet clad flooding, preferential flooding (i.e., independent variation in the DFC and TSC and outside the TSC) and partial flooding (i.e, variations in moderator elevations). Normal condition structural analysis in Section 3.A and off-normal and accident structural analysis of the fuel, basket, TSC and cask in Section 11.A demonstrate that no operating condition induces geometry variations in the system beyond those allowed by the manufacturing tolerances.

Structural analyses demonstrate that the TSC confinement boundary remains intact through all storage operating conditions. Therefore, moderator is not present in the TSC while it is in the concrete cask. However, access to the concrete cask interior environment is possible via the air inlets and outlets and the heat transfer annulus between the TSC and the cask steel liner. This access provides paths for moderator intrusion during a flood. Under off-normal and accident conditions, moderator intrusion into the convective heat transfer annulus is evaluated.

System criticality control is achieved through the use of neutron absorber sheets (BORAL®) attached to the exterior faces of the fuel tubes. Individual fuel assemblies are held in place by stainless steel structural disks. The basket design includes 68 fuel tubes, one tube per fuel assembly or DFC, with the DFC tubes having a slightly larger (oversized) opening.

Criticality evaluations rely on modeled neutron absorber ^{10}B loadings of 0.015 g/cm^2. The modeled areal density is arrived at by multiplying the minimum 0.02 g/cm^2 ^{10}B areal density specified for the absorber by a 75% efficiency factor.

MCNP [A3], a three-dimensional Monte Carlo code, is used in the system criticality analysis. Evaluations are primarily based on the ENDF/B-VI continuous energy neutron cross-section library [A4] available in the MCNP distribution. Nuclides for which no ENDF/B-VI data is available are set to the latest cross-section sets available in the code distribution. The code and cross-section libraries are benchmarked by comparison to a range of critical experiments relevant to light water reactor fuel in storage and transport casks. An upper subcritical limit (USL) for the system is determined based on guidance given in NUREG/CR-6361 [A9].

Key assembly physical characteristics and maximum initial enrichment for the loading of the two LACBWR fuel assembly types are shown in Table 6.A.1-1, with the allowed loading configuration shown in Figure 6.A.1-1. Maximum enrichment is defined as planar-average enrichment for the variably enriched Exxon (EX) assemblies.

Undamaged fuel assemblies are evaluated with a full, nominal set of fuel rods. Fuel rod (lattice) locations may contain filler rods. A filler rod must occupy, at a minimum, a volume equivalent to the fuel rod it displaces. Filler rods may be placed into the lattice after assembly in-core use or be designed to replace fuel rods prior to use. The undamaged Exxon assembly must contain its nominal set of inert rods.

The maximum multiplication factors (k_{eff} +2σ) are calculated, using conservative assumptions, for the transfer and concrete casks. The USL applied to the analysis results is 0.9372 per Section 6.A.5. Maximum reactivities are produced by the damaged fuel payloads. The results of the analyses are presented in detail in Section 6.A.3.4 and are summarized as follows.

Cask Body	Operating Condition	Water Density (g/cc)			k_{eff}+2σ
		TSC Interior	DFC Interior	TSC Exterior	
Transfer	--	0.0001	0.0001	0.0001	0.35333
Transfer	--	0.9982	0.9982	0.0001	0.87655
Transfer	--	0.9982	0.9982	0.9982	0.87636
Transfer	--	0.0001	0.9982	0.9982	0.91423
Transfer	--	0.0001	0.9982	0.0001	0.93014
Storage	Normal	0.0001	0.0001	0.0001	0.34222
Storage	Accident	0.0001	0.0001	0.9982	0.33691

The maximum reactivity was established for different baseline configurations, one in which the fuel retained its preirradiation configuration and one where damage occurred, resulting in the potential reconfiguration of fissile material.

Maximum reactivity of the system with fuel in an undamaged condition, i.e., fuel geometry is retained, was calculated to be one where system tolerances are combined to form a worst case, maximum reactivity geometry (also referred to as combined tolerance model). In this configuration, the following tolerances are applied:

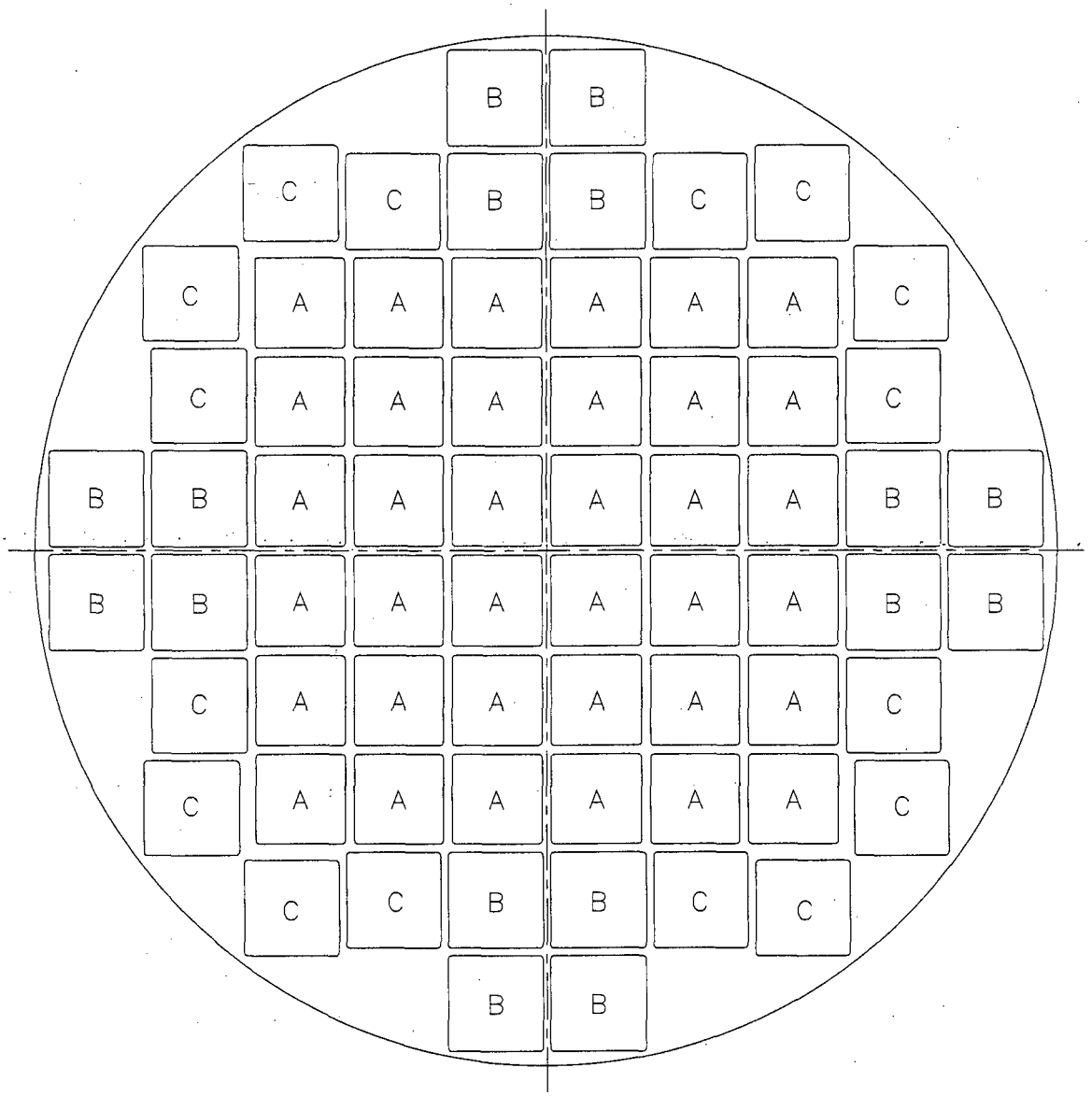
- Minimum absorber width and maximum absorber thickness
- Maximum the width and minimum tube thickness
- Maximum size disk opening at the minimum radial location
- Minimum disk pitch at maximum thickness

Fuel assemblies and fuel tubes are moved radial to the center of the canister with fuel assembly clad to pellet gap flooded. This configuration results in a maximum k_{eff} of approximately 0.85.

Maximum system reactivity was achieved for the configuration in which damaged fuel is located in the exterior 32 locations of the basket as noted in Figure 6.A.1-1. In this configuration, undamaged Exxon fuel is located in the 36 interior locations. The maximum damaged fuel configuration is one where fuel clad, in addition to any other assembly steel hardware, is assumed to be removed from the DFC, and the fuel pellet stacks are assumed to be floating in the DFC at maximum square pitch available in the DFC opening. The DFCs are flooded with full density water to maximize neutron moderation, while the canister cavity is dry, reducing neutron absorber effectiveness. As there is no moderator in the canister cavity in the dry canister, preferential flood configuration, system manufacturing tolerances, such as disk thickness, tube thickness, and absorber size, do not affect system reactivity to any statistical extent. Storage casks are evaluated in an infinite array of casks by applying reflective boundary conditions, while transfer cask reactivities are based on a single cask model.

Analysis of moderator density in the canister shows a monotonic decrease in reactivity with decreasing moderator density for undamaged fuel. The full moderator density TSC interior condition bounds any off-normal or accident condition with the exception of the preferentially flooded DFC case. Analysis of moderator intrusion into the concrete cask heat transfer annulus with the dry TSC shows a slight decrease in reactivity from the completely dry condition.

Figure 6.A.1-1 Allowed Loading Configuration for LACBWR Fuel



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Damaged Allis Chalmers fuel maximum enrichment 3.64 wt % ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Damaged Allis Chalmers fuel maximum enrichment 3.94 wt % ^{235}U.

Table 6.A.1-1 Bounding Fuel Assembly Loading Criteria

Fuel ID		AC	EX
Array		10×10	10×10
Number of Fuel Rods		100	96
Max. Active Length	[in]	83	83
Rod Pitch	[in]	0.565	0.557
Rod Diameter	[in]	0.396	0.394
Pellet Diameter	[in]	0.350	0.343
Clad Thickness	[in]	0.020	0.0220
Max. MTU(5)	[MTU]	0.1214	0.1119
Number of Inert Rods(1,2)		0	4
Listed Inert Rod OD	[in]	N/A	0.3940
Max. Enrichment	[wt.% ^{235}U]	3.64/3.94(4)	3.71(3)

Notes:

(1) Not required for fuel assemblies located in DFC.
(2) Inert rods comprised of stainless steel clad tube containing zirconium alloy slug.
(3) Planar average enrichment.
(4) Two AC fuel types, Type 1 at an enrichment of 3.64 wt % ^{235}U and Type 2 at 3.94 wt % ^{235}U.
(5) Damaged fuel cans are allowed to contain an additional 5% fissile material to account for loose pellets not necessarily associated with the as-built assembly.

General Note: All dimensions represent nominal, cold, unirradiated values.

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6.A.3 Model Specification

6.A.3.1 Description of Calculation Model

The MCNP code is used to model the storage and transfer casks containing a full load of fuel assemblies. The system contains up to 68 fuel assemblies, with up to 32 assemblies being damaged. The MCNP code package uses combinatorial geometry, with the option to divide the model into self-contained Universes. The self-contained Universe structure can be used to separate canister, cask, and fuel into individual components that can be easily modified and checked.

The basic component of the geometry package is a set of general surfaces. To reduce the required user input MCNP includes simplified expressions for cylinders and planes perpendicular to system axes and "macrobodies" (cubes, finite cylinders, wedges, etc). Models are constructed by combining geometry components (surfaces) into cells. Cells may be embedded in individual Universes to simplify modeling. A given Universe may be included in different positions within the geometry by translation. Translation allows movement in the x, y, and z directions and rotation using direction cosines.

Finite cask/canister/basket/fuel models (termed cask model henceforth) are constructed for the storage and transfer system. The cask models are constructed in a set of distinct phases. In the first phase a fuel assembly is constructed from the basic components of the fuel assembly, i.e., fuel rod, inert rod (Exxon fuel), and nozzle (end-fitting). Next the basket structure is placed within the canister cavity. The basket structure is comprised of a set of stainless steel tubes and aluminum/B_4C (BORAL®) neutron absorber panels within a stacked set of stainless steel and aluminum disks. Fuel assemblies are placed into the canister cavity with the basket structure superimposed on the cavity surrounding the assemblies using the Universe structure. The canister shell and lid are placed around the loaded basket. The complete canister is then placed into either the transfer or storage cask overpack. For damaged fuel models an additional phase is the construction of the damaged fuel can (DFC) which surrounds the fuel material in the outer 32 fuel tubes.

Components within the fuel region are evaluated for manufacturing tolerance effects.

6.A.3.1.1 Fuel Assembly Models

Fuel assemblies are constructed of a top and bottom end-fitting, also referred to as top and bottom nozzles, inert rods (EX fuel only), and fuel rods that are held via grid spacers between the top and bottom nozzles. LACBWR assemblies do not contain channels.

The majority of fuel rods are not attached to either the top or bottom end-fitting. A limited number of fuel rods function as tie rods that connect the top and bottom end-fittings. Fuel rods may also be segmented. To simplify model construction all fuel rods are modeled as "floating" between the top and bottom end-fittings. All fuel rods are modeled at the full active fuel length. The number of fuel rods depends on the fuel design being evaluated. A fuel rod consists of: (1) a bottom end cap, (2) an active fuel region (including clad), (4) a top plenum, and (5) a top end cap. Spacing associated with the insulator disks is included in the end-cap definition. As stated in the assumptions, grid spacers are not modeled.

For each fuel assembly type selected a set of Universes is constructed. Assembly Universes are typically fuel rod, inert rod, tube/rod array, and the complete fuel assembly. Table 6.A.3-1 shows the geometry input summary sheet for the AC fuel type. Similar data is developed for the EX model with inert rods replacing four interior fuel rods.

Partial flooding evaluations are limited to cases with a void region above the active fuel region and canister cavity moderator from the top of the active fuel region to the canister bottom. This configuration allows a potential reactivity increase by neutron reflection from the steel lid, while minimizing neutron leakage in the moderated canister volume. As optimum moderator studies in Section 6.A.4 have demonstrated that the active fuel region achieves maximum system reactivity at the fully flooded (0.9982 g/cc) condition for the nonpreferentially flooded DFC and a void TSC (0.0001 g/cc) for the preferentially flooded DFC, moderator density changes as a function of elevation in the active fuel region are not required (i.e., decreasing moderator density in the active fuel region at any elevation will reduce reactivity, as the region has been demonstrated to be undermoderated).

6.A.3.1.2 Fuel Tube Model

The basket contains 68 fuel tubes, 36 standard size tubes and 32 DFC tubes. Each standard tube contains up to two absorber sheets, each held to a tube face by a stainless steel cover sheet. One standard size fuel tube is modeled for each absorber configuration which is then replicated using the MCNP "fill" option to fill the basket. DFCs contain either three or four absorbers. Similarly, one DFC tube is modeled for each required absorber configuration and duplicated using the "fill" option. A sketch of a fuel tube is shown in Figure 6.A.3-1.

Key dimensions for the fuel tube are shown in shown in Table 6.A.3-2. Manufacturing tolerances are applied within the calculated parameters.

6.A.3.1.3 DFC Model

The basket may contain up to 32 DFCs. Each DFC is comprised of a stainless steel rectangular box with a lid and bottom plate structure. The DFC is designed to freely drain and fill, thereby avoiding the potential of preferentially flooding the DFC. Key dimensions for the DFC are shown in Table 6.A.3-3.

DFC contents included in these evaluations are:

- Undamaged clad fuel assemblies with functional grids (i.e., fuel assemblies retaining clad and fuel rods in their as-designed configuration);
- A hypothetical fuel configuration, where the clad (stainless steel) is conservatively removed from the model and rods are allowed to achieve increased pitch; and
- A homogenized fuel/water mixture model.

For modeling the unclad array the fuel assembly input structure is repeated, this time with no clad or end-fitting. Homogenized fuel is modeled as a simple rectangular volume with the maximum mixture elevations set as a variable (compact fuel to full DFC cavity height). Material composition is then calculated based on the fuel mass within the DFC, DFC cross-section, and moderator density within the mixture volume.

6.A.3.1.4 Basket Model

Key dimensions for the basket are shown in shown in Table 6.A.3-4. A sketch of a support disk is shown in Figure 6.A.3-2. Manufacturing tolerances are applied within the calculated parameters. An MCNP Universe composed of the basket disks is constructed with no fuel or openings. Into this Universe the disk opening, tube, DFC (if required), and payload are inserted. Up to three fuel types may be implemented within the analysis, one for interior assemblies and up to two in the exterior DFC slots.

The fuel assemblies (or DFC payload) are placed into the basket at their respective radial and axial locations via the "fill" option. Fuel assembly location, DFC location, and disk opening locations and sizes may be adjusted to any desired manufacturing tolerance and shift direction.

6.A.3.1.5 Canister Model

Criticality evaluations for the storage and transfer configurations model the system with lid and port covers in place. A sketch of the canister model, composed of a canister cavity body, surrounded by lid, canister shell weldment and bottom plate, is shown in Figure 6.A.3-3. To simplify the geometry, the lid is modeled as a single cylinder with no penetrations. Parameters required in the canister model are listed in Table 6.A.3-5.

6.A.3.1.6 Transfer Cask Model

Model parameters for the transfer cask shown in Figure 6.A.3-4 are listed in Table 6.A.3-6. The cask body model is an arrangement of cylindrical shells. The origin of the transfer cask Universe (Universe 0) is the top of the shield door in the center of the cask, corresponding to the center bottom of the canister bottom plate. The canister filling the cask cavity must, therefore, be raised by the canister bottom thickness. The shield door is modeled by a combination of macro bodies (RHP and RPP). The retaining ring is not included in the model. Since the transfer cask is open to the cask exterior the cask exterior material is employed in the canister-to-cask gap.

6.A.3.1.7 Storage Cask Model

Model parameters for the storage cask shown in Figure 6.A.3-5 are listed in Table 6.A.3-7. The cask body model is an arrangement of a cylindrical carbon steel shell surrounded by concrete. Cask model complexity is associated with the pedestal stand and the air inlet and outlet structure. The cask body is composed of a lower section, housing the canister and air inlet structure, and an upper section composed of the cask lid and air outlet structure. The origin of the concrete cask Universe (Universe 0) is set to the top of the bottom weldment pedestal plate in the center of the cask, corresponding to the center bottom of the canister bottom plate. The canister filling the cask cavity must, therefore, be raised by the canister bottom thickness. Since the storage cask is open to the cask exterior the cask exterior material is employed in the canister-to-cask gap and air inlet and outlet structure.

6.A.3.1.8 Monte Carlo Sampling and Result Convergence

To assure proper spatial fission site distribution, 30 inactive generations (cycles) are tracked prior to starting the active cycles employed in the eigenvalue calculation. The standard deviation (σ) for all cases will be less than 0.002. Upon completion of the analysis, a reactivity comparison is made to the upper subcritical limit (USL). These comparisons are made using the reported "final estimated combined collision/absorption/track-length k_{eff}" identified in each output. Each output is checked for sampling of fissile material areas in the model and normal distribution of the k_{eff} values.

6.A.3.1.9 Sample Model Files

Sample input files for the transfer cask and storage cask models for undamaged and damaged fuel are provided in Section 6.A.7. The inputs provided represent the maximum reactivity configuration case and additional transfer and storage cases.

Table 6.A.3-9 Basket, TSC, and Cask Material Densities and Compositions

Material	Density g/cm^3	Element/ Isotope	Density atom/barn-cm
Carbon Steel	7.821	Fe	8.35E-02
		C	3.92E-03
Stainless Steel	7.94	Cr	1.75E-02
		Fe	5.95E-02
		Ni	7.74E-03
		Mn	1.74E-03
Aluminum	2.702	Al	6.03E-02
Lead	11.344	Pb	3.30E-02
NS-4-FR	1.632	H	5.8508E-02
		^{10}B	9.1385E-05
		^{11}B	3.3665E-04
		C	2.2600E-02
		N	1.3904E-03
		O	2.6107E-02
		Al	7.8003E-03
Concrete	2.322	H	1.3879E-02
(145 lb/ft^3)		O	4.6522E-02
		Na	1.7643E-03
		Al	1.7625E-03
		Si	1.6783E-02
		Ca	1.5356E-03
		Fe	3.5063E-04
Neutron Absorber	1.965	^{10}B	7.10E-03
(Core)	(75% of	^{11}B	3.01E-02
	Actual)	C	9.30E-03
		Al	2.48E-02

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6.A.4 Criticality Calculation

6.A.4.1 Calculation Method

System reactivity evaluations are performed with the MCNP5 three-dimensional Monte Carlo code and continuous neutron energy cross-sections [A3, A4]. The Monte Carlo code and neutron cross-section libraries are validated for use in fuel transport and storage cask applications through a series of calculations based on critical experiments. Validation detail is presented in Section 6.A.3.4. Validation and cask analysis apply a consistent cross-section set to ensure applicability of the experimental benchmark analysis to the cask analysis. Cross-sections used in both the benchmark and cask analysis are shown in the cask MCNP sample input file material descriptions in Section 6.A.7.

The criticality analysis of the system is performed in several steps.

- For Undamaged Fuel
  - Establish initial reactivities effects of gap conditions and fuel types for LACBWR fuel assemblies and justify the use of planar average versus heterogeneous enrichments.
  - Evaluate basket mechanical perturbations and basket geometric tolerances.
  - Construct optimum moderator density curve(s).
  - Based on the maximum reactivity configuration, demonstrate that the system reactivity remains below the USL.
- For Damaged Fuel
  - Establish maximum reactivity configuration for damaged fuel.
  - Evaluate optimum moderator density considering preferential flooding of DFC.
  - Justify that maximum reactivity configuration for undamaged fuel applies to the damaged fuel configuration.
  - Demonstrate that the reactivity for the maximum reactivity damaged fuel configuration remains below the USL.

6.A.4.2 Fuel Loading Optimization

The fuel loading is optimized in the criticality models using the following:

- Fresh fuel at 96% theoretical density
- Bounding fuel assembly characterization
- The most reactive cask configuration

The maximum reactivity cask configuration considers basket fabrication tolerances, component shifting, and moderator density evaluations. Each of these effects is evaluated individually and in combination to assure that the highest reactivity configuration is documented. Fabrication tolerances and shift effect are evaluated for each of the LACBWR fuel types.

Casks are evaluated at various interior and exterior flood conditions with reflective boundary conditions. Reflective boundary conditions are applied to an independently generated cylindrical body surrounding the cask body. This allows the modeling of infinite cask arrays at various cask spacings. Space between the cask surface and the reflecting body may be flooded at various moderator densities. Given the low neutron fluxes of either concrete or transfer cask bodies, no significant effect is observed from conditions outside the cask body.

The cask reactivity evaluation is divided into two primary sections. The first section details the determination of the maximum reactivity configuration for a payload of undamaged fuel assemblies. The maximum reactivity configuration is a function of the bounding parameters of the fuel, fuel location within the basket, manufacturing tolerances of the basket, and optimum moderator level. The second section details the extension of the analysis to consider damaged fuel cans (DFCs).

6.A.4.2.1 Undamaged Fuel

The maximum reactivity configuration is the result of fabrication tolerances, component shifting, and moderator density evaluations. Each of these effects is initially evaluated separately to determine the highest reactivity basket configuration. An evaluation combining tolerance and shift effects builds on the individual evaluations. These analyses are performed with the canister placed into the transfer cask. Fabrication tolerances and shift effects are evaluated using each of the primary fuel types.

Fuel Type and Enrichment Variation Study

LACBWR fuel is composed of two primary fuel types: Allis Chalmers (AC) fuel that was produced as Type I with a uniform enrichment of 3.64 wt% ^{235}U and as a Type II with a 3.94 wt% ^{235}U enrichment, and Exxon fuel with radial enrichment variations. Radial enrichments varied from 3.12 to 4.05 wt% ^{235}U, with an average enrichment of 3.70 wt% ^{235}U. A maximum planar average enrichment of 3.71 wt% ^{235}U is evaluated. Each of the fuel types is individually evaluated to determine their relative reactivity within the LACBWR cask system. Wet and dry pellet to clad gaps are evaluated to demonstrate the effect of minor clad failure on system reactivity. Radial enrichments were discretely modeled to allow a comparison to a homogenized enrichment.

Fabrication Tolerances

The basket is composed of a set of fuel tubes, held in place by disks which in turn are held in place by a set of tie rods, spacers and washers. Tube location in the basket is controlled by the disk opening cut-out in the structural disk (see Figure 6.A.3-2). Tolerances on the location and

size of the opening in the structural disk evaluated in Section 6.A.4.3 are included in Figure 6.A.3-2. Moderator space between fuel tubes is controlled by a number of items including: fuel tube size, fuel tube thickness, absorber thickness and width, and locations of the disk opening and tube within the disk opening.

Component Shift

In addition to the component tolerances, a reactivity study on component shifts is required. Based on the tube in disk arrangement the radial shifts evaluated are the movement of the fuel assembly within the tubes, and the tubes within the disk opening. A sketch of the basket tube cross-section is included in Figure 6.A.3-1, with the structural disk shown in Figure 6.A.3-2.

Combined Basket Tolerances and Shift

In addition to the individual effect studies, the combination of the various basket tolerances with the maximum reactivity shift configuration (center shift) are considered in the results section of the chapter.

Moderator Density Study

Optimum moderator density is evaluated by considering TSC uniform water density variations ranging from void (0.0001 g/cm^3) to full density (0.9982 g/cm^3). More moderator density evaluations are performed with a void interior and various exterior water densities, and simultaneous variations in interior and exterior water densities, for the damaged fuel evaluations.

6.A.4.2.2 Damaged Fuel Evaluations

Starting with the most reactive basket configuration determined for undamaged fuel, various damaged fuel configurations are evaluated. Base evaluations for DFC designs are comprised of undamaged EX assemblies in the basket interior with EX or AC fuel assemblies in the DFCs. No AC assemblies are allowed in the non DFC basket locations.

To bound the potential of additional fissile material in the DFCs the total fissile mass in each DFC is increased by 5%. For mixture cases, the mass is increased by adjusting the fuel volume in the mixture. For pellet stack cases, the active fuel height is increased. As demonstrated in the various moderator density and rod pitch evaluations, the fuel rod lattice is under-moderated. Placing additional fissile material within the active fuel region displaces moderator and reduces reactivity. An increased active fuel height configuration was chosen.

Clad Rods – Grid Undamaged

DFC evaluations are initiated by placing undamaged fuel assemblies into the DFC to demonstrate the effect of the additional stainless steel shell. Table 6.A.4-8 demonstrates that

starting from an all EX fuel payload, the DFC material results in a minor decrease in system reactivity. Next, fuel rods are removed from the lattice to evaluate various "missing rod geometries." The missing rod pattern was chosen to increase potential reactivity by not removing adjacent fuel rods, i.e., not generating large moderator volumes that would result in exceeding the H/U peak and result in excess water neutron absorption; and by initially removing fuel rods near the center of the assembly, i.e., away from the neutron absorbers and in the area where increases in the moderator ratio are most likely to increase system reactivity.

Unclad Rods – Loose Pellet Stack

Damaged fuel is postulated to lose its cladding. As stainless steel presents a significant parasitic absorber within the fuel region, the removal of cladding from the array is evaluated. The next stage in the analysis is an increase in pitch for the unclad array. As the assembly array showed an increase in reactivity as a function of moderator density, increased rod pitches are also evaluated.

Fuel/Water Mixture

In addition to the heterogeneous (rod array) evaluations, a set of mixture analysis are run. Mixtures are designed to fill various fractions of the DFC canister and simulate small fuel rubble inside the canister.

The fuel mixture studies model fuel material at various material fractions, expressed as mixture height (i.e., fuel/water mixture height in DFC). At the low end fraction, the material is modeled as a solid mass at preirradiation density (10.522 g/cc or 96% theoretical), while at the high end, the fuel water mixture fills the DFC cavity. Per Table 6.A.4-10, a compacted mixture at full density occupies approximately 25% of the DFC. Therefore, the full height mixture will result in a fuel density approximately 25% of the initial case (i.e., 2.6 g/cc). This study is, therefore, an optimum moderator density study for the fuel/water mixture. Water within the mixture is full density in all cases.

Optimum Moderator

In the final stage of the maximum reactivity configurations analysis, moderator density variations are considered. Including the damaged fuel optimum moderator configuration studies, a preferential flooding of the damaged fuel cans is evaluated (i.e., DFCs may be flooded with a dry canister cavity and vice versa).

Maximum Reactivity Configuration Confirmation Studies

Maximum reactivity geometry for undamaged fuel was based on a fully flooded TSC. For preferentially flooded geometry, the fissile material shift and tolerance studies are repeated. Additional evaluations to assess the movement of the active fuel towards the top and bottom of the DFCs must also be included for damaged fuel contents.

6.A.4.2.3 Storage Cask Evaluations

Additional runs for the maximum reactivity assembly are performed for the single storage cask and array of casks under normal and accident conditions. These studies evaluate the effects of differences in reflector material on canister reactivity.

6.A.4.3 Criticality Results

The cask reactivity evaluation is divided into two primary sections. The first section details the results of undamaged fuel elevations with the second section detailing the extension of the analysis to consider damaged fuel cans (DFCs).

6.A.4.3.1 Undamaged Fuel Maximum Reactivity Basket Configuration

The maximum reactivity configuration is the result of fabrication tolerances, component shifting, and moderator density evaluations. Each of these effects is initially evaluated separately to determine the highest reactivity basket configuration. An evaluation combining tolerance and shift effects builds on the individual evaluations. As stated in the assumptions, these analyses are performed with the canister placed into the transfer cask.

Fabrication tolerances and shift effects are evaluated using each of the primary fuel types.

Fuel Type and Enrichment Variation Study

Each of the fuel types is individually evaluated to determine their relative reactivity within the LACBWR cask system. Radial enrichments were discretely modeled to allow a comparison to a homogenized enrichment. Results for the individual cases are shown in Table 6.A.4-1 and indicate a substantially higher reactivity for the AC fuel type. Wet and dry pellet to clad gaps are evaluated with a wet clad being bounding.

As shown in the result table, the radial and homogenized enrichment results are statistically identical. The variable enriched fuel may therefore be represented by the homogenized "average" enrichment description.

Fabrication Tolerances – Centered Components

The results of the tolerance evaluation for centered fuel assemblies and tubes are included in Table 6.A.4-2 and Table 6.A.4-3. As indicated in the tables, little statistically significant information is available from this study. Viewed independently, none of the fabrication related tolerance studies with the exception of fuel tube thickness produce significant reactivity increases across both fuel types. A combined model is therefore constructed based on theoretical maximum reactivity configuration. Maximum reactivity configuration is one that minimizes the amount of moderator and neutron absorber (BORAL and tube material) within the radial plane

between disks (i.e., maximum tube opening, minimum tube wall thickness, minimum absorber width and maximum absorber thickness). Disk thickness increase and decreased disk pitch increases the amount of parasitic absorber within the center (axial) fuel region while simultaneously decreasing water space between fuel tubes, thereby reducing the BORAL effectiveness. Structural disk thickness and pitch tolerances, therefore, invoke offsetting effects. A reduced flux trap (maximum disk thickness and minimum disk pitch) configuration is chosen for the maximum reactivity model. As shown in the tolerance study result tables, the combined tolerance model significantly increases system reactivity and produces the maximum reactivity configuration. Further evaluations of the component tolerance evaluation are performed in conjunction with the shifted component configuration.

Component Shift – Nominal Basket Configuration

The results of the shift evaluation are shown in Table 6.A.4-4 and Table 6.A.4-5 for Allis Chalmers (AC) and Exxon (EX) fuel and indicate that moving the fuel assembly and tube towards the basket center ("in" shift) clearly increases system reactivity.

Combined Basket Tolerances and Shift

Previous sections evaluated basket tolerances and fuel shifts as separate effects. This section evaluates the effect of combining various basket tolerances with the maximum reactivity shift configuration (center shift). The results for these evaluations are shown in Table 6.A.4-6 and Table 6.A.4-7. Similar to the results of the independent basket tolerance evaluations, applying fabrication tolerances to the basket components provides limited statistically significant change in system reactivity. A combined shifted/toleranced model is, therefore, constructed containing minimum absorber/flux trap space.

The maximum reactivity configuration is, therefore, based on:

a) Minimum absorber width and maximum absorber thickness
b) Maximum tube width and minimum tube thickness;
c) Maximum size disk openings at their minimum radial location;
d) Minimum disk pitch at maximum disk thickness

The minimum fuel assembly separation modeled in this configuration reduces the amount of moderation and the corresponding effectiveness of the absorber sheet, which depend on the ^{10}B neutron capture cross-section in the thermal energy range.

Moderator Density Study

The moderator density variation studies are limited to the AC (most reactive) fuel assembly type as it produces maximum system reactivity. The results of these analyses are shown in Figure

6.A.4-1. Maximum reactivity occurs at full density water inside the canister cavity. This analysis also demonstrates the low reactivity, k_{eff}<0.4, for the dry basket system.

More moderator density evaluations are performed with a void interior and various exterior water densities, and simultaneous variations in interior and exterior water densities, for the damaged fuel can evaluations.

6.A.4.3.2 Damaged Fuel Can (DFC) Evaluations

Starting with the most reactive basket configuration determined for undamaged fuel, various damaged fuel configurations are evaluated. Base evaluations for DFC designs are comprised of undamaged EX assemblies in the basket interior with EX or AC fuel assemblies in the DFCs. No AC assemblies are allowed in the non DFC basket locations.

Clad Rods – Grid Undamaged

Table 6.A.4-8 demonstrates that starting from an all EX fuel payload, the DFC material results in a minor decrease in system reactivity. Next, fuel rods are removed from the lattice to evaluate various "missing rod geometries." System reactivity for the AC and EX elements is not significantly affected by the removal of fuel rods.

Unclad Rods – Loose Pellet Stack

As listed in Table 6.A.4-9, system reactivity increases with rod pitch. A significant increase occurs for the midpoint pitch (midpoint between nominal assembly pitch and maximum pitch in the DFC) for both fuel types. Rod removal from the unclad rod maximum pitch array does not statistically affect or reduce system reactivity for either fuel type.

Fuel/Water Mixture

Shown in Table 6.A.4-10 are the results of the mixture analysis. Results for the homogenous mixture cases are lower than those of the rod array studies.

Optimum Moderator

In the final stage of the maximum reactivity configurations, analysis moderator density variations are considered for damaged fuel canisters. Similar to the undamaged fuel analysis, a uniform reduction in moderator density reduces reactivity. Maximum reactivity is achieved with maximum density moderator within the DFC and void (low density moderator) in the TSC cavity. This configuration reduces neutron absorber effectiveness by hardening the neutron spectrum. Moderator density cases are illustrated in Figure 6.A.4-2. Maximum reactivity plus two sigma for a 3.71 wt % ^{235}U Exxon interior loading and 3.94 wt % ^{235}U AC exterior DFC pattern is 0.9373, which is above the USL. To reduce system reactivity, the AC fuel is divided between 3.64 and 3.94 wt % ^{235}U enriched assemblies. The optimum moderator density study is

repeated for a loading pattern of 16 high and 16 low enriched AC assemblies, with the lower enriched material restricted to the "on-axis" locations. The "off-axis" higher, 3.94 wt % ^{235}U enriched AC loading bounds a payload of either 3.64 or 3.94 wt % ^{235}U AC fuel since the higher enrichment increases reactivity. Maximum reactivity ($k_{eff}+2\sigma$) of 0.92636 is below the USL and represents the allowed loading configuration as illustrated in Figure 6.A.4-3. Figure 6.A.4-4 illustrates the allowed loading configuration. Within the scope of the moderator density evaluations, a partial drain-down of the TSC is evaluated. Draining the cask to the top of the active fuel region changes the axial reflection from water to steel TSC lid. The reactivity change for this configuration is not statistically resolvable at Δk of 0.003 for the wet TSC case (k_{eff} 0.879) and -0.002 (k_{eff} 0.9250) for the dry TSC and wet DFC case. Drain down further than the top of the active fuel region decreases reactivity, as demonstrated by the optimum moderator density stuides. Justification for loading Exxon fuel in the DFCs is included in the following section.

Maximum Reactivity Configuration Confirmation Studies

As maximum reactivity geometry was based on a fully flooded TSC, the fissile material shift and tolerance studies are repeated at the preferential flood configuration. Shift models contain two additional options. The additional options move DFC fissile material into close contact with each other. Results from studies, shown in Table 6.A.4-11, demonstrate that the shifted radial towards canister center ("in") pattern represents the most reactive fissile material configuration. Since the dry TSC removes the inter-assembly moderator, the minor geometry variations do not significantly affect system reactivity. Similarly, the small manufacturing tolerances on the system do not significantly affect system reactivity in the dry TSC cavity, see Table 6.A.4-12.

While all Exxon fuel is expected to be undamaged, the potential exists for Exxon assemblies to be characterized as damaged prior to placement in the TSC. Table 6.A.4-13 contains the reactivities for models containing Exxon fuel in some or all DFC locations. There is no increase in system reactivity when considering Exxon assemblies in DFCs.

All previous evaluations rely on models where the active fuel region in the DFCs is aligned with the active fuel region of the undamaged fuel in the basket interior locations. Evaluations to assess the movement of the active fuel towards the top and bottom of the DFCs are considered. These evaluations expose small amounts of fissile material outside the axial absorber coverage. As shown in Table 6.A.4-14, the effect of the fuel location within the DFC is minor for the dry TSC and insignificant for the wet TSC models. Maximum uncertainty adjusted (2σ) reactivity for the transfer cask is 0.93014. This model represents a partially flooded DFC with DFC fuel material arranged in an unclad pellet array at the bottom of the DFC.

6.A.4.3.3 Storage Cask Evaluations

Additional runs for the maximum reactivity assembly are performed for the single storage cask and array of casks under normal and accident conditions. Results of these evaluations are shown in Table 6.A.4-15. Results include a comparison to the single transfer cask evaluations. Both axially aligned fuel and DFC bottom models are evaluated for the storage cask. Transfer cask results are significantly higher than those of the storage cask at identical TSC configuration.

6.A.4.3.4 Allowed Fuel Specifications and Comparison to USL Range of Applicability

Fuel assembly characteristics evaluated for content within the MPC-LACBWR system are listed within Table 6.A.4-16. The variables are based on the maximum reactivity model for a preferentially flooded DFC with a maximum reactivity configuration payload. A comparison of the maximum reactivity case parameters to the MCNP range of applicability is shown in Table 6.A.4-17.

Figure 6.A.4-1 Undamaged Fuel Reactivity versus Water Density (AC Fuel, 3.94 wt % ^{235}U)

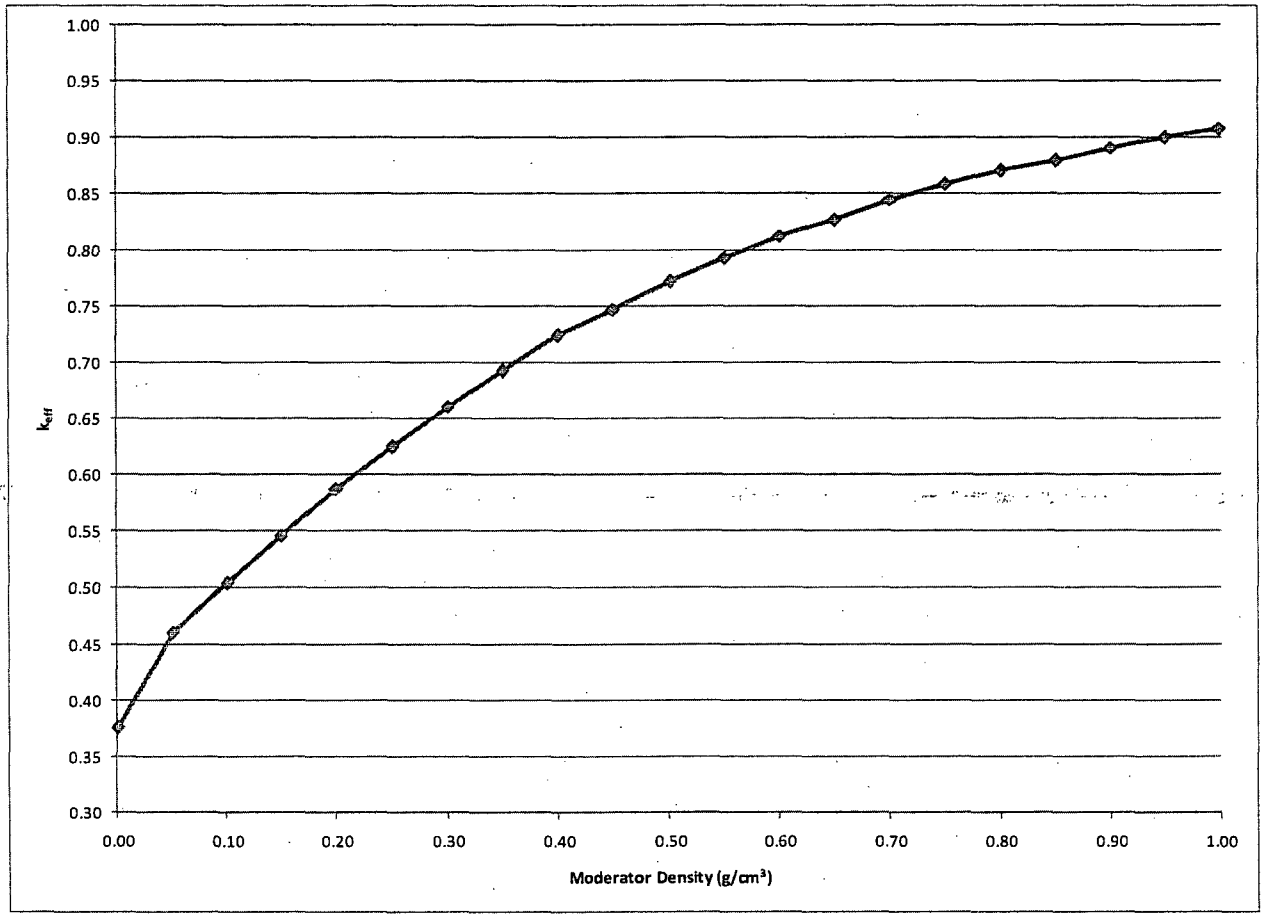


Figure 6.A.4-2 DFC Cask System Reactivity versus Water Density
(3.71 wt % EX Interior Locations, 3.94 wt % ^{235}U AC DFC Locations)

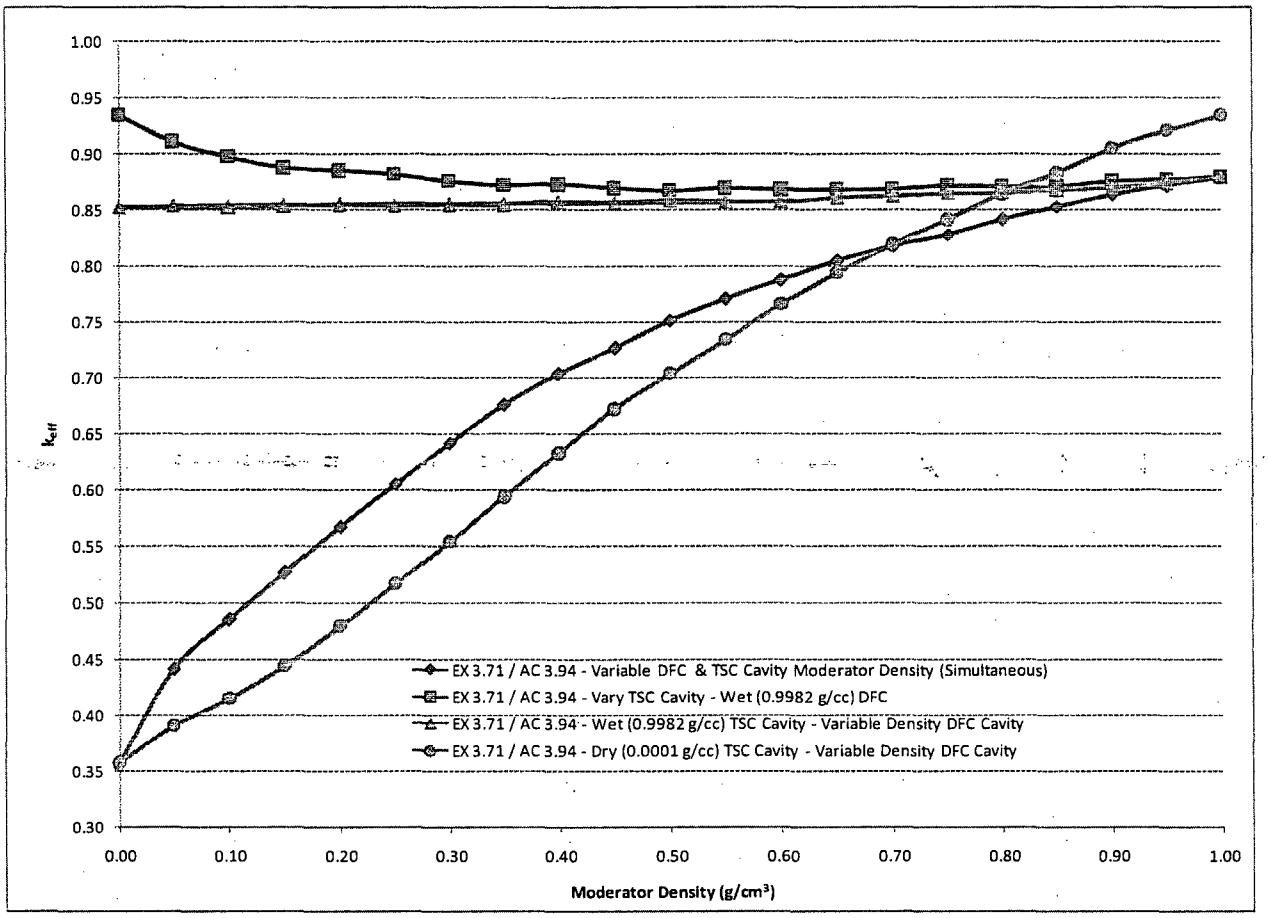


Figure 6.A.4-3 DFC Cask System Reactivity versus Water Density
(3.71 wt % EX Interior Locations, 3.94/3.64 wt % ^{235}U AC DFC Locations)

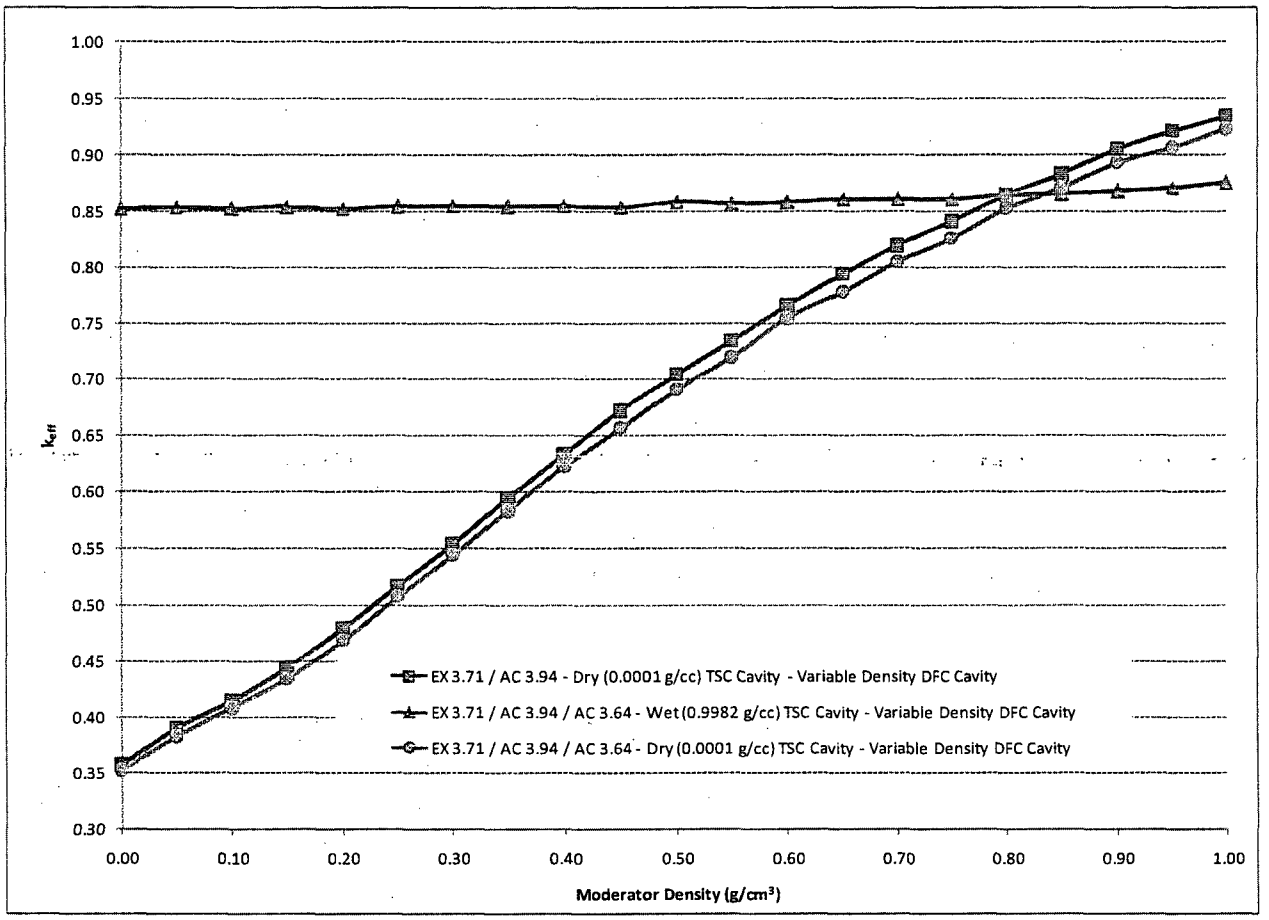
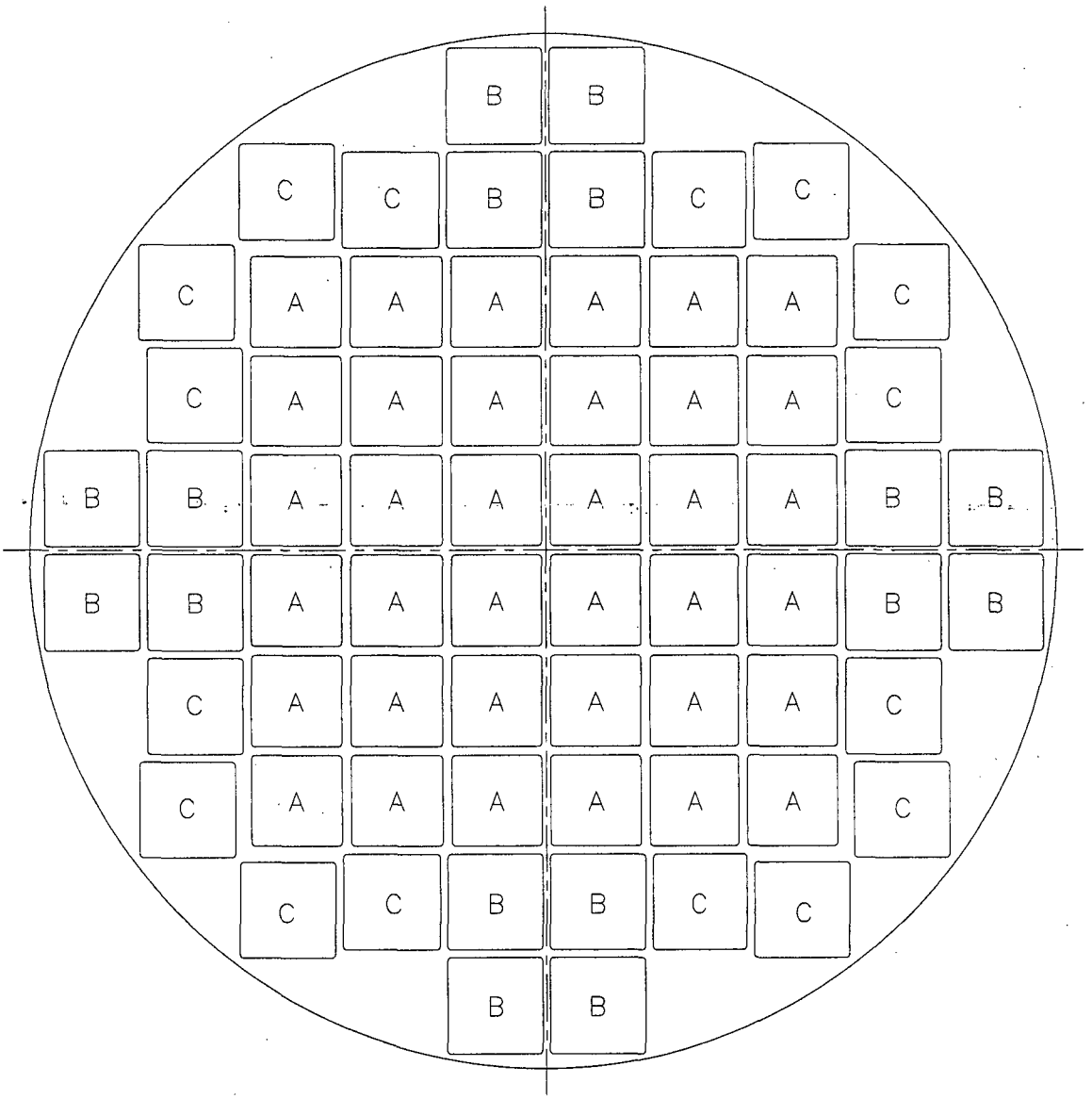


Figure 6.A.4-4 Allowed Loading Configuration for LACBWR Fuel



Slot A: Undamaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Slot B: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Damaged Allis Chalmers fuel maximum enrichment 3.64 wt % ^{235}U.

Slot C: Undamaged or damaged Exxon fuel maximum planar average enrichment 3.71 wt % ^{235}U.

Damaged Allis Chalmers fuel maximum enrichment 3.94 wt % ^{235}U.

Table 6.A.4-1 Baseline Reactivity Evaluation Results – Typical Tube and Disk Basket Maximum Reactivity Configuration

Fuel Type	Discrete Enrichment	Pellet / Clad Gap	Enrichment % ^{235}U	k_{eff}	Δk
AC	N/A	Dry	3.64	0.88298	--
AC	N/A	Dry	3.94	0.90521	--
EX	N/A	Dry	3.71	0.85525	--
EX	N/A	Dry	3.70	0.85365	--
EX	Yes	Dry	3.70	0.85313	-0.00052
AC	N/A	Wet	3.64	0.88704	--
AC	N/A	Wet	3.94	0.90794	--
EX	N/A	Wet	3.71	0.86033	--

Table 6.A.4-2 Component Tolerance Study – No Shift – Allis Chalmers Fuel (AC) at 3.94 wt % ^{235}U

Tube Tol.		Absorber Tol.		Disk Tol.						
Outer Width	Thickness	Width	Thickness	Op_Width	Location	Thickness	Spacing	k_{eff}	Δk	Δk/σ
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.88983	--	--
Min	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.88980	-0.00003	0.0
Max	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.89144	0.00161	1.4
Nom	Min	Nom	Nom	Nom	Nom	Nom	Nom	0.89662	0.00679	6.0
Nom	Max	Nom	Nom	Nom	Nom	Nom	Nom	0.88524	-0.00459	-4.0
Nom	Nom	Min	Nom	Nom	Nom	Nom	Nom	0.89243	0.00260	2.3
Nom	Nom	Max	Nom	Nom	Nom	Nom	Nom	0.88889	-0.00094	-0.9
Nom	Nom	Nom	Min	Nom	Nom	Nom	Nom	0.89041	0.00058	0.5
Nom	Nom	Nom	Max	Nom	Nom	Nom	Nom	0.89027	0.00044	0.4
Nom	Nom	Nom	Nom	Min	Nom	Nom	Nom	0.88937	-0.00046	-0.4
Nom	Nom	Nom	Nom	Max	Nom	Nom	Nom	0.89041	0.00058	0.5
Nom	Nom	Nom	Nom	Nom	Min	Nom	Nom	0.88911	-0.00072	-0.6
Nom	Nom	Nom	Nom	Nom	Max	Nom	Nom	0.88988	0.00005	0.0
Nom	Nom	Nom	Nom	Nom	Nom	Min	Nom	0.88959	-0.00024	-0.2
Nom	Nom	Nom	Nom	Nom	Nom	Max	Nom	0.88931	-0.00052	-0.5
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Min	0.89031	0.00048	0.4
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Max	0.89061	0.00078	0.7
Max	Min	Min	Max	Max	Min	Max	Min	0.90051	0.01068	9.6

Table 6.A.4-3 Component Tolerance Study – No Shift – Exxon Fuel (EX) at 3.71 wt % ^{235}U

Tube Tol.		Absorber Tol.		Disk Tol.						
Outer Width	Thickness	Width	Thickness	Op_Width	Location	Thickness	Spacing	k_{eff}	Δk	$\Delta k/\sigma$
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.84195	--	--
Min	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.84009	-0.00186	-1.6
Max	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.84328	0.00133	1.2
Nom	Min	Nom	Nom	Nom	Nom	Nom	Nom	0.84800	0.00605	5.6
Nom	Max	Nom	Nom	Nom	Nom	Nom	Nom	0.83573	-0.00622	-5.5
Nom	Nom	Min	Nom	Nom	Nom	Nom	Nom	0.84201	0.00006	0.1
Nom	Nom	Max	Nom	Nom	Nom	Nom	Nom	0.83957	-0.00238	-2.1
Nom	Nom	Nom	Min	Nom	Nom	Nom	Nom	0.83924	-0.00271	-2.5
Nom	Nom	Nom	Max	Nom	Nom	Nom	Nom	0.84288	0.00093	0.8
Nom	Nom	Nom	Nom	Min	Nom	Nom	Nom	0.84140	-0.00055	-0.5
Nom	Nom	Nom	Nom	Max	Nom	Nom	Nom	0.84149	-0.00046	-0.4
Nom	Nom	Nom	Nom	Nom	Min	Nom	Nom	0.84094	-0.00101	-0.9
Nom	Nom	Nom	Nom	Nom	Max	Nom	Nom	0.84121	-0.00074	-0.6
Nom	Nom	Nom	Nom	Nom	Nom	Min	Nom	0.84150	-0.00045	-0.4
Nom	Nom	Nom	Nom	Nom	Nom	Max	Nom	0.84342	0.00147	1.3
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Min	0.84181	-0.00014	-0.1
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Max	0.84073	-0.00122	-1.1
Max	Min	Min	Max	Max	Min	Max	Min	0.85141	0.00946	8.4

Table 6.A.4-4 Component Shift Study – No Tolerance Applied – Allis Chalmers Fuel (AC) at 3.94 wt % ^{235}U

Comp. Radial Shift				
Fuel	**Tube**	k_{eff}	Δk	$\Delta k/\sigma$
Centered	Centered	0.88983	--	--
Centered	Top	0.88874	-0.00109	-1.0
Centered	Bottom	0.88951	-0.00032	-0.3
Centered	Right	0.88977	-0.00006	-0.1
Centered	Left	0.88928	-0.00055	-0.5
Centered	Corner	0.88995	0.00012	0.1
Centered	In	0.89618	0.00635	5.6
Centered	Out	0.88445	-0.00538	-4.8
Centered	Custom	0.88866	-0.00117	-1.0
Top	Centered	0.89067	0.00084	0.7
Top	Top	0.88932	-0.00051	-0.5
Bottom	Centered	0.89094	0.00111	1.0
Bottom	Bottom	0.88929	-0.00054	-0.5
Right	Centered	0.88994	0.00011	0.1
Right	Right	0.88984	0.00001	0.0
Left	Centered	0.89233	0.00250	2.2
Left	Left	0.88972	-0.00011	-0.1
Corner	Centered	0.88965	-0.00018	-0.2
Corner	Corner	0.89040	0.00057	0.5
In	Centered	0.89248	0.00265	2.3
In	In	0.89765	0.00782	6.9
Out	Centered	0.88705	-0.00278	-2.5
Out	Out	0.88181	-0.00802	-7.2

Table 6.A.4-5 Component Shift Study – No Tolerance Applied – Exxon Fuel (EX) at 3.71 wt % ^{235}U

Comp. Radial Shift				
Fuel	**Tube**	k_{eff}	Δk	$\Delta k/\sigma$
Centered	Centered	0.84195	--	--
Centered	Top	0.84142	-0.00053	-0.5
Centered	Bottom	0.84128	-0.00067	-0.6
Centered	Right	0.84094	-0.00101	-0.9
Centered	Left	0.84025	-0.00170	-1.5
Centered	Corner	0.84256	0.00061	0.6
Centered	In	0.84580	0.00385	3.5
Centered	Out	0.83437	-0.00758	-6.8
Centered	Custom	0.84188	-0.00007	-0.1
Top	Centered	0.84185	-0.00010	-0.1
Top	Top	0.84227	0.00032	0.3
Bottom	Centered	0.84096	-0.00099	-0.9
Bottom	Bottom	0.84099	-0.00096	-0.9
Right	Centered	0.84210	0.00015	0.1
Right	Right	0.83924	-0.00271	-2.4
Left	Centered	0.84188	-0.00007	-0.1
Left	Left	0.83911	-0.00284	-2.7
Corner	Centered	0.84070	-0.00125	-1.1
Corner	Corner	0.84220	0.00025	0.2
In	Centered	0.84182	-0.00013	-0.1
In	In	0.84753	0.00558	4.9
Out	Centered	0.84047	-0.00148	-1.3
Out	Out	0.83438	-0.00757	-6.9

Table 6.A.4-6 Component Tolerance Study – In Shift Allis Chalmers Fuel (AC) at 3.94 wt % ^{235}U

Tube Tol.		Absorber Tol.		Disk Tol.						
Outer Width	**Thickness**	**Width**	**Thickness**	**Op_Width**	**Location**	**Thickness**	**Spacing**	k_{eff}	**Δk**	**Δk/σ**
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.89765	--	--
Min	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.89728	-0.00037	-0.3
Max	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.89960	0.00195	1.7
Nom	Min	Nom	Nom	Nom	Nom	Nom	Nom	0.90312	0.00547	4.8
Nom	Max	Nom	Nom	Nom	Nom	Nom	Nom	0.89114	-0.00651	-5.7
Nom	Nom	Min	Nom	Nom	Nom	Nom	Nom	0.89853	0.00088	0.8
Nom	Nom	Max	Nom	Nom	Nom	Nom	Nom	0.89477	-0.00288	-2.5
Nom	Nom	Nom	Min	Nom	Nom	Nom	Nom	0.89727	-0.00038	-0.3
Nom	Nom	Nom	Max	Nom	Nom	Nom	Nom	0.89679	-0.00086	-0.7
Nom	Nom	Nom	Nom	Min	Nom	Nom	Nom	0.89797	0.00032	0.3
Nom	Nom	Nom	Nom	Max	Nom	Nom	Nom	0.89666	-0.00099	-0.9
Nom	Nom	Nom	Nom	Nom	Min	Nom	Nom	0.89715	-0.00050	-0.5
Nom	Nom	Nom	Nom	Nom	Max	Nom	Nom	0.89735	-0.00030	-0.3
Nom	Nom	Nom	Nom	Nom	Nom	Min	Nom	0.89625	-0.00140	-1.3
Nom	Nom	Nom	Nom	Nom	Nom	Max	Nom	0.89615	-0.00150	-1.3
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Min	0.89713	-0.00052	-0.5
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Max	0.89683	-0.00082	-0.7
Max	Min	Min	Max	Max	Min	Max	Min	0.90794	0.01029	9.1

Table 6.A.4-7 Component Tolerance Study – In Shift Exxon Fuel (EX) at 3.71 wt % ^{235}U

Tube Tol.		Absorber Tol.		Disk Tol.						
Outer Width	Thickness	Width	Thickness	Op_Width	Location	Thickness	Spacing	k_{eff}	Δk	Δk/σ
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.84753	--	--
Min	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.84771	0.00018	0.2
Max	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.85028	0.00275	2.4
Nom	Min	Nom	Nom	Nom	Nom	Nom	Nom	0.85547	0.00794	6.9
Nom	Max	Nom	Nom	Nom	Nom	Nom	Nom	0.84325	-0.00428	-3.8
Nom	Nom	Min	Nom	Nom	Nom	Nom	Nom	0.84929	0.00176	1.6
Nom	Nom	Max	Nom	Nom	Nom	Nom	Nom	0.84789	0.00036	0.3
Nom	Nom	Nom	Min	Nom	Nom	Nom	Nom	0.84925	0.00172	1.5
Nom	Nom	Nom	Max	Nom	Nom	Nom	Nom	0.84800	0.00047	0.4
Nom	Nom	Nom	Nom	Min	Nom	Nom	Nom	0.84621	-0.00132	-1.2
Nom	Nom	Nom	Nom	Max	Nom	Nom	Nom	0.84798	0.00045	0.4
Nom	Nom	Nom	Nom	Nom	Min	Nom	Nom	0.84824	0.00071	0.6
Nom	Nom	Nom	Nom	Nom	Max	Nom	Nom	0.84717	-0.00036	-0.3
Nom	Nom	Nom	Nom	Nom	Nom	Min	Nom	0.84801	0.00048	0.4
Nom	Nom	Nom	Nom	Nom	Nom	Max	Nom	0.84796	0.00043	0.4
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Min	0.84936	0.00183	1.6
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Max	0.84810	0.00057	0.5
Max	Min	Min	Max	Max	Min	Max	Min	0.86033	0.01280	11.4

Table 6.A.4-8 DFC Undamaged Assembly and Missing Rod Study Results

Primary Fuel (Interior)		Secondary Fuel (Exterior)							
Type	Enrichment wt % ^{235}U	Type	Enrichment wt % ^{235}U	DFC Model	DFC Content	Missing Rods #	k_{eff}	Δk	$\Delta k/\sigma$
EX	3.71	N/A	N/A	No	N/A	N/A	0.86033	--	--
EX	3.71	N/A	N/A	Yes	IAA	N/A	0.85806	--	--
EX	3.71	N/A	N/A	Yes	IAA	1	0.86003	0.00197	1.8
EX	3.71	N/A	N/A	Yes	IAA	4	0.85946	0.00140	1.3
EX	3.71	N/A	N/A	Yes	IAA	8	0.85741	-0.00065	-0.6
EX	3.71	N/A	N/A	Yes	IAA	12	0.85886	0.00080	0.7
EX	3.71	N/A	N/A	Yes	IAA	16	0.85972	0.00166	1.5
EX	3.71	N/A	N/A	Yes	IAA	20	0.85883	0.00077	0.7
EX	3.71	N/A	N/A	Yes	IAA	30	0.85751	-0.00055	-0.5
EX	3.71	N/A	N/A	Yes	IAA	48	0.85433	-0.00373	-3.5
EX	3.71	AC	3.94	Yes	IAA	N/A	0.86144	--	--
EX	3.71	AC	3.94	Yes	IAA	1	0.86068	-0.00076	-0.7
EX	3.71	AC	3.94	Yes	IAA	4	0.86033	-0.00111	-1.0
EX	3.71	AC	3.94	Yes	IAA	8	0.86081	-0.00063	-0.6
EX	3.71	AC	3.94	Yes	IAA	12	0.86111	-0.00033	-0.3
EX	3.71	AC	3.94	Yes	IAA	16	0.86129	-0.00015	-0.1
EX	3.71	AC	3.94	Yes	IAA	20	0.86198	0.00054	0.5
EX	3.71	AC	3.94	Yes	IAA	32	0.86132	-0.00012	-0.1
EX	3.71	AC	3.94	Yes	IAA	50	0.85505	-0.00639	-5.8

Table 6.A.4-9 DFC Damaged Assembly – No Clad, Modified Pitch, Missing Rod Study Results

Primary Fuel		Secondary Fuel								
Type	Enrichment wt % ^{235}U	Type	Enrichment wt % ^{235}U	DFC Model	DFC Content	Pitch [in]	Missing Rods #	k_{eff}	Δk	Δk/σ
X	3.71	AC	3.94	Yes	URA	--	N/A	0.87209	--	--
EX	3.71	AC	3.94	Yes	URA	0.5825	N/A	0.87595	0.00386	3.6
EX	3.71	AC	3.94	Yes	URA	0.5999	N/A	0.87930	0.00721	6.4
EX	3.71	AC	3.94	Yes	URA	0.5999	N/A	0.87930	--	--
EX	3.71	AC	3.94	Yes	URA	0.5999	1	0.88051	0.00121	1.1
EX	3.71	AC	3.94	Yes	URA	0.5999	4	0.87886	-0.00044	-0.4
EX	3.71	AC	3.94	Yes	URA	0.5999	8	0.87760	-0.00170	-1.5
EX	3.71	AC	3.94	Yes	URA	0.5999	12	0.87747	-0.00183	-1.6
EX	3.71	AC	3.94	Yes	URA	0.5999	16	0.87521	-0.00409	-3.7
EX	3.71	AC	3.94	Yes	URA	0.5999	20	0.87090	-0.00840	-7.2
EX	3.71	AC	3.94	Yes	URA	0.5999	32	0.86681	-0.01249	-10.8
EX	3.71	AC	3.94	Yes	URA	0.5999	50	0.86071	-0.01859	-16.7
EX	3.71	EX	3.71	Yes	URA	0.557	N/A	0.86752	--	--
EX	3.71	EX	3.71	Yes	URA	0.5789	N/A	0.87208	0.00456	4.2
EX	3.71	EX	3.71	Yes	URA	0.6007	N/A	0.87275	0.00523	4.6
EX	3.71	EX	3.71	Yes	URA	0.6007	N/A	0.87275	--	--
EX	3.71	EX	3.71	Yes	URA	0.6007	1	0.87319	0.00044	0.4
EX	3.71	EX	3.71	Yes	URA	0.6007	4	0.87174	-0.00101	-0.9
EX	3.71	EX	3.71	Yes	URA	0.6007	8	0.87075	-0.00200	-1.8
EX	3.71	EX	3.71	Yes	URA	0.6007	12	0.86867	-0.00408	-3.6
EX	3.71	EX	3.71	Yes	URA	0.6007	16	0.86906	-0.00369	-3.1
EX	3.71	EX	3.71	Yes	URA	0.6007	20	0.86673	-0.00602	-5.2
EX	3.71	EX	3.71	Yes	URA	0.6007	30	0.86065	-0.01210	-10.7
EX	3.71	EX	3.71	Yes	URA	0.6007	48	0.85479	-0.01796	-15.6

Table 6.A.4-10 DFC Damaged Assembly – Fuel Mixture Study Results

Primary Fuel		Secondary Fuel							
Type	Enrichment wt % ^{235}U	Type	Enrichment wt % ^{235}U	DFC Model	DFC Content	Mix. Height	k_{eff}	Δk	$\Delta k/\sigma$
EX	3.71	N/A	N/A	No	N/A	N/A	0.86033	--	--
EX	3.71	N/A	N/A	Yes	MIX	0.25	0.85102	-0.00931	-8.3
EX	3.71	N/A	N/A	Yes	MIX	0.3	0.85200	-0.00833	-7.8
EX	3.71	N/A	N/A	Yes	MIX	0.4	0.85778	-0.00255	-2.3
EX	3.71	N/A	N/A	Yes	MIX	0.5	0.85994	-0.00039	-0.3
EX	3.71	N/A	N/A	Yes	MIX	0.6	0.86369	0.00336	2.9
EX	3.71	N/A	N/A	Yes	MIX	0.7	0.86599	0.00566	5.1
EX	3.71	N/A	N/A	Yes	MIX	0.8	0.86621	0.00588	5.3
EX	3.71	N/A	N/A	Yes	MIX	0.9	0.86521	0.00488	4.2
EX	3.71	N/A	N/A	Yes	MIX	1	0.86485	0.00452	4.0
EX	3.71	AC	3.94	Yes	MIX	0.25	0.85118	-0.00915	-8.3
EX	3.71	AC	3.94	Yes	MIX	0.3	0.85096	-0.00937	-8.5
EX	3.71	AC	3.94	Yes	MIX	0.4	0.85511	-0.00522	-4.8
EX	3.71	AC	3.94	Yes	MIX	0.5	0.86074	0.00041	0.4
EX	3.71	AC	3.94	Yes	MIX	0.6	0.86393	0.00360	3.2
EX	3.71	AC	3.94	Yes	MIX	0.7	0.86783	0.00750	6.6
EX	3.71	AC	3.94	Yes	MIX	0.8	0.86864	0.00831	7.2
EX	3.71	AC	3.94	Yes	MIX	0.9	0.86780	0.00747	6.7
EX	3.71	AC	3.94	Yes	MIX	1	0.86659	0.00626	5.7

Table 6.A.4-11 Preferentially Flooded System Shift Study – (AC Fuel, 3.94/3.64 wt % ^{235}U DFC Locations, EX Fuel, 3.71 wt % ^{235}U Interior Locations)

Comp. Radial Shift				
Fuel	**Tube**	k_{eff}	Δk	$\Delta k/\sigma$
In	In	0.92286	--	--
DFC_Close_A	DFC_Close_A	0.92045	-0.00241	-1.0
DFC_Close_B	DFC_Close_B	0.92183	-0.00103	-0.4

Table 6.A.4-12 DFC Shift Study – Preferential Flood Tolerance Evaluations Based on DFC_Close_B Shift Pattern

Tube Tol.		Absorber Tol.		Disk Tol.						
Outer Width	**Thickness**	**Width**	**Thickness**	**Op_Width**	**Location**	**Thickness**	**Spacing**	k_{eff}	**Δk**	**Δk/σ**
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.92414	--	--
Min	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.92371	-0.00043	-0.2
Max	Nom	Nom	Nom	Nom	Nom	Nom	Nom	0.92117	-0.00297	-1.3
Nom	Min	Nom	Nom	Nom	Nom	Nom	Nom	0.91849	-0.00565	-2.5
Nom	Max	Nom	Nom	Nom	Nom	Nom	Nom	0.92373	-0.00041	-0.2
Nom	Nom	Min	Nom	Nom	Nom	Nom	Nom	0.92429	0.00015	0.1
Nom	Nom	Max	Nom	Nom	Nom	Nom	Nom	0.92288	-0.00126	-0.5
Nom	Nom	Nom	Min	Nom	Nom	Nom	Nom	0.92149	-0.00265	-1.2
Nom	Nom	Nom	Max	Nom	Nom	Nom	Nom	0.92201	-0.00213	-0.9
Nom	Nom	Nom	Nom	Min	Nom	Nom	Nom	0.92180	-0.00234	-1.0
Nom	Nom	Nom	Nom	Max	Nom	Nom	Nom	0.92311	-0.00103	-0.5
Nom	Nom	Nom	Nom	Nom	Min	Nom	Nom	0.92550	0.00136	0.6
Nom	Nom	Nom	Nom	Nom	Max	Nom	Nom	0.92160	-0.00254	-1.1
Nom	Nom	Nom	Nom	Nom	Nom	Min	Nom	0.92189	-0.00225	-1.0
Nom	Nom	Nom	Nom	Nom	Nom	Max	Nom	0.92253	-0.00161	-0.7
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Min	0.92638	0.00224	1.0
Nom	Nom	Nom	Nom	Nom	Nom	Nom	Max	0.92124	-0.00290	-1.3

Table 6.A.4-13 Exxon Placement in DFC

Primary Fuel		Secondary Fuel		Third Fuel				
Type	Enrichment wt % ^{235}U	Type	Enrichment wt % ^{235}U	Type	Enrichment wt % ^{235}U	k_{eff}	Δk	$\Delta k/\sigma$
EX	3.71	AC	3.94	AC	3.64	0.92286	--	--
EX	3.71	AC	3.94	EX	3.71	0.92175	-0.00123	-0.5
EX	3.71	EX	3.71	AC	3.64	0.90577	-0.01753	-7.5
EX	3.71	N/A	N/A	N/A	N/A	0.91357	-0.00977	-4.2

Table 6.A.4-14 Active Fuel Location Study for Wet TSC and Dry TSC

DFC Fuel Location	TSC	k_{eff}	Δk	$\Delta k/\sigma$
Active Fuel	Dry	0.92286	--	--
Bottom	Dry	0.92680	0.00394	1.6
Top	Dry	0.92502	0.00216	0.9
Active Fuel	Wet	0.87505	--	--
Bottom	Wet	0.87601	0.00096	0.8
Top	Wet	0.87254	-0.00251	-2.3

Table 6.A.4-15 Storage Cask Analyses Results

DFC Fuel Loc.Axial	Cask Model	Cask Array	Density (H_2O) Lattice	Exterior	DFC	k_{eff}	σ	$k_{eff}+2\sigma$	Δk
Active Fuel	Transfer	SC[1]	0.0001	0.0001	0.9982	0.92286	0.00175	0.92636	--
Active Fuel	Storage	SC	0.0001	0.0001	0.9982	0.90915	0.00161	0.91237	-0.01371
Active Fuel	Storage	SC	0.0001	0.0001	N/A	0.33965	0.00054	0.34073	-0.56950
Active Fuel	Storage	InfArray	0.9982	0.0001	N/A	0.87621	0.00083	0.87787	-0.04665
Active Fuel	Storage	InfArray	0.0001	0.0001	0.9982	0.91159	0.00163	0.91485	-0.01127
Active Fuel	Storage	InfArray	0.0001	0.0001	N/A	0.33869	0.00057	0.33983	-0.58417
Active Fuel	Storage	InfArray	0.0001	0.9982	N/A	0.33438	0.00069	0.33576	-0.58848
Active Fuel	Storage	InfArray	0.9982	0.9982	N/A	0.87447	0.00082	0.87611	-0.04839
Bottom	Transfer	SC	0.0001	0.0001	0.9982	0.92680	0.00167	0.93014	--
Bottom	Storage	SC	0.0001	0.0001	0.9982	0.91418	0.00170	0.91758	-0.01262
Bottom	Storage	SC	0.0001	0.0001	N/A	0.33994	0.00057	0.34108	-0.57424
Bottom	Storage	InfArray	0.9982	0.0001	N/A	0.87578	0.00082	0.87742	-0.05102
Bottom	Storage	InfArray	0.0001	0.0001	0.9982	0.91571	0.00164	0.91899	-0.01109
Bottom	Storage	InfArray	0.0001	0.0001	N/A	0.34100	0.00061	0.34222	-0.58580
Bottom	Storage	InfArray	0.0001	0.9982	N/A	0.33553	0.00069	0.33691	-0.59127
Bottom	Storage	InfArray	0.9982	0.9982	N/A	0.87363	0.00076	0.87515	-0.05317

[1] Single cask model.

Table 6.A.4-16 Fuel Assembly Limiting Characteristics

Fuel ID		AC	EX
Array		10×10	10×10
Number of Fuel Rods		100	96
Max. Active Length	[in]	83	83
Rod Pitch	[in]	0.565	0.557
Rod Diameter	[in]	0.396	0.394
Pellet Diameter	[in]	0.350	0.343
Clad Thickness	[in]	0.020	0.0220
Max. MTU(5)	[MTU]	0.1214	0.1119
Number of Inert Rods(1,2)		0	4
Listed Inert Rod OD	[in]	N/A	0.3940
Enrichment	[wt % ^{235}U]	3.64/3.94(4)	3.71(3)

Notes:

(1) Not required for fuel assemblies located in DFC.
(2) Inert rods comprised of stainless steel clad tube containing zirconium alloy slug.
(3) Planar average enrichment.
(4) Two AC fuel types, Type 1 at an enrichment of 3.64 wt % ^{235}U and Type 2 at 3.94 wt % ^{235}U.
(5) Damaged fuel cans are allowed to contain an additional 5% spent fuel rod material to account for loose pellets not necessarily associated with the as-built assembly.

Table 6.A.4-17 MCNP Validation – Range of Applicability

Parameter	Minimum	Maximum	MPC-LACBWR
Enrichment (wt % ^{235}U)	2.350%	4.738%	3.94%
Fuel rod pitch (cm)	1.30	2.54	1.524 (AC Loose Pellets)
Fuel pellet outer diameter (cm)	0.790	1.265	0.889 (AC Loose Pellets)
Fuel rod diameter (cm)	0.9400	1.4172	N/A
H/^{235}U atom ratio	72.7	403.9	195 (AC Loose Pellets)
Soluble boron (ppm by weight)	0	4986	0
Cluster Gap (cm)	1.206	13.750	1.2(1)
Boron (^{10}B) plate loading (g/cm^2)	0.0000	0.0670	0.020
EALCF (eV)	0.09781	0.77219	0.4215

Note: (1) Represents minimum flux trap between DFCs. Corner DFCs are separated by up to 2.8 cm. The maximum reactivity payload cluster gap is less than 1% lower than the range evaluated. This will not affect the USL applicability to the MPC-LACBWR basket.

linear regression is performed on each system parameter, and the line with the greatest correlation is used to functionalize β.

Section 6.A.5.2 contains the extensive list of LWR critical benchmarks employed in the validation of MCNP with its continuous energy neutron cross-section libraries. The range of parameters included in the benchmarks is shown in Table 6.A.5-1.

Included in Section 6.A.5.2 are linear fits of reactivity (k_{eff}) to each of the system parameters. Experiments were chosen to reflect the fuel and basket geometry and materials, and the spent fuel cask criticality control mechanism. This includes the use of square pitched, low enriched uranium oxide fuel rods, a rectangular arrangement of assembles, light water moderation, and criticality control by spacing, borated moderator, and/or borated absorber panels and tubes. Trending in k_{eff} was evaluated for the following independent variables: wt % ^{235}U, rod pitch, H/U volume ratio, energy of the average neutron lethargy causing fission (EALCF), ^{10}B loading of the absorber sheet, and soluble boron loading. No statistically significant trends were found for any of the system parameters. USLs are, therefore, generated for each of the independent variables. A minimum USL covering the range of applicability of the benchmark set is determined as detailed in Section 6.A.5.2.

6.A.5.2 Results of Benchmark Calculations

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. The linear correlation coefficient, R, is calculated by taking the square root of the R^2 value. In particular, the correlation coefficient, R, is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If R is +1, a perfect linear relationship with a positive slope is indicated; and if R is -1, a perfect linear relationship with a negative slope is indicated. When R is 0, no linear relationship is indicated.

Table 6.A.5-2 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter. Linear fits and correlation constants are based on the 183 data-point evaluation sets plotted in Section 6.A.5-3. The cluster gap plot is limited to the 137 data points for experiments containing multiple fuel rod clusters. Single fuel rod cluster experiments documented in LEU-COMP-THERM sets 06, 14, 35 and 50, in addition to LEU-COMP-THERM experiments 01-01, 02-01 to -03, and 08-01 to -15, were therefore excluded from the cluster gap study. The 183 data points evaluated for the remaining parameters represent the complete set of experiments listed in Section 6.A.5-3 minus the three high energy lethargy experiments above 0.35 eV (Experiments LEU-COMP-THERM 14-05, -06 and -07). Addition of these points, while not resulting in a significant linear fit, produces a noticeable slope to the

USL correlation not representative of the remaining data fits. As this increased slope results in a higher USL, it is acceptable to discard these data points. The three higher energy points are removed from all independent variables for consistency.

As there is no significant correlation to any of the independent variables, the USL for each independent variable is calculated and shown with its range of applicability in Table 6.A.5-2. A sample output for EALCF is shown in Figure 6.A.5-1. Uncertainties included in the USLSTATS evaluation are the Monte Carlo uncertainty associated with the reactivity calculation and experimental uncertainty that was provided in the literature for each of the cases.

Based on all the independent variable correlations, a lower limit constant USL of 0.9376 may be applied. The range of applicability (area of applicability) of this limit may be extended to 5 wt % enriched fuel, as the correlation shows no significant trend with enrichment between 2.35 and 4.74 wt %, and that the limited trending observed increases the USL. Extending the range of applicability for the average neutron lethargy is based on a minimal, but positive, trend of the USL versus EALCF. Studies, including additional data points up to 0.7722 eV, indicate that the trending continues to the higher energy levels.

6.A.5.3 Critical Benchmarks

From the International Handbook of Evaluated Criticality Safety Benchmark Experiments [A8], 186 experiments are selected as basis of the MCNP benchmarking. Experiments were selected for compatibility of materials and geometry with the spent fuel casks. Of particular interest are benchmarks with rectangular arrays of low enriched uranium oxide fuel rods in which reactivity is controlled by soluble boron or borated plates (tubes).

MCNP benchmark cases represent a collection of files composed of inputs directly obtained from references (with cross-section sets adjusted to those used in the cask analysis), NAC modified input files representing unique geometries based on reference input files, and input files constructed from the experimental material and geometry information. All cases were reviewed on a "preparer/checker" principle for modeling consistency with the cask models and the choice of code options. Due to large variations in the benchmark complexities, not all options employed in the cask models are reflected in each of the benchmarks (e.g., UNIVERSE structure). A review of the criticality results did not indicate any result trend due to particular modeling choices (e.g., using the UNIVERSE structure versus a single universe, or employing KSRC versus SDEF sampling).

Key system parameters, the experimental uncertainty, and calculated k_{eff} and σ for each experiment are shown in Table 6.A.5-3. The case number listed in the table corresponds to the

handbook LEU-COMP-THERM-XXX experiment identification (e.g., Case 1.07 corresponds to LEU-COMP-THERM-001 Experiment 7). Stochastic Monte Carlo error is kept within ±0.2% and each output is checked to assure that the MCNP build-in statistical checks on the results are passed and that all fissile material is sampled.

Scatter plots of k_{eff} versus system parameters for 183 data point sets (full set-minus three high lethargy points above 0.35 eV) are created (see Figure 6.A.5-2 through Figure 6.A.5-10). Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (R^2) to statistically indicate any trend or lack thereof. Scatter plates are created for k_{eff} versus the following.

Enrichment in ^{235}U (wt % ^{235}U)
Fuel rod pitch (cm)
Fuel pellet outer diameter (cm)
Fuel rod outer diameter (cm)
Hydrogen/uranium (^{235}U) atom ratio
Soluble boron (ppm by weight)
Cluster gap spacing (spacing between assemblies in cm)
Boron (^{10}B) plate loading (g/cm^2)
Energy of average neutron lethargy causing fission (eV)

Figure 6.A.5-1 USLSTATS Output for EALCF

```
*****************************************************************************
                        Version 1.4, April 23, 2003
                       Oak Ridge National Laboratory
*****************************************************************************
 Input to statistical treatment from file:enrich-183.in
 Title: keff vs enrichment
 Proportion of the population = .995
 Confidence of fit            = .950
 Confidence on proportion     = .950
 Number of observations       =  183
 Minimum value of closed band =    0.00
 Maximum value of closed band =    0.00
 Administrative margin        =    0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
2.35000E+00	9.94910E-01	3.42000E-03	2.35000E+00	9.95090E-01	3.46000E-03
2.35000E+00	9.92830E-01	3.38000E-03	2.35000E+00	9.92520E-01	3.47000E-03
2.35000E+00	9.98060E-01	3.38000E-03	2.35000E+00	9.95620E-01	3.50000E-03
2.35000E+00	9.96550E-01	3.42000E-03	2.35000E+00	9.93130E-01	3.55000E-03
2.35000E+00	9.89310E-01	3.44000E-03	2.35000E+00	9.98130E-01	3.58000E-03
2.35000E+00	9.95340E-01	3.41000E-03	2.35000E+00	9.96700E-01	3.56000E-03
2.35000E+00	9.93880E-01	3.44000E-03	2.35000E+00	9.93830E-01	3.55000E-03
2.35000E+00	9.89690E-01	3.36000E-03	2.35000E+00	9.92770E-01	3.47000E-03
4.30600E+00	9.95160E-01	2.79000E-03	2.35000E+00	9.92920E-01	3.50000E-03
4.30600E+00	9.93670E-01	2.54000E-03	2.35000E+00	9.96410E-01	3.46000E-03
4.30600E+00	9.96340E-01	2.76000E-03	2.35000E+00	9.93060E-01	3.49000E-03
4.30600E+00	9.93110E-01	2.64000E-03	2.35000E+00	9.96500E-01	3.45000E-03
4.30600E+00	9.93000E-01	2.49000E-03	2.35000E+00	9.94680E-01	3.50000E-03
2.59600E+00	9.92680E-01	2.10000E-03	2.35000E+00	9.93300E-01	3.47000E-03
2.59600E+00	9.93190E-01	2.14000E-03	2.35000E+00	9.91810E-01	3.46000E-03
2.59600E+00	9.92990E-01	2.13000E-03	2.35000E+00	9.93920E-01	3.47000E-03
2.59600E+00	9.94790E-01	2.13000E-03	2.35000E+00	9.95560E-01	3.55000E-03
2.59600E+00	9.93100E-01	2.12000E-03	2.35000E+00	9.94540E-01	3.51000E-03
2.59600E+00	9.93240E-01	2.12000E-03	2.35000E+00	9.94490E-01	3.47000E-03
2.59600E+00	9.91990E-01	2.12000E-03	2.35000E+00	9.91300E-01	3.52000E-03
2.59600E+00	9.93820E-01	2.12000E-03	2.35000E+00	9.94800E-01	3.47000E-03
2.59600E+00	9.94450E-01	2.12000E-03	2.35000E+00	9.93500E-01	3.60000E-03
2.59600E+00	9.95440E-01	2.13000E-03	2.35000E+00	9.94000E-01	3.45000E-03
2.59600E+00	9.94410E-01	2.12000E-03	2.35000E+00	9.96280E-01	3.53000E-03
2.59600E+00	9.93920E-01	2.15000E-03	2.35000E+00	9.92620E-01	3.45000E-03
2.59600E+00	9.95090E-01	2.14000E-03	2.35000E+00	9.94100E-01	3.53000E-03
2.59600E+00	9.93780E-01	2.12000E-03	2.35000E+00	9.96470E-01	3.52000E-03
2.59600E+00	9.95040E-01	2.14000E-03	2.35000E+00	9.93600E-01	3.47000E-03
2.59600E+00	9.94380E-01	2.11000E-03	2.35000E+00	9.97020E-01	3.49000E-03
2.59600E+00	9.95730E-01	2.12000E-03	2.35000E+00	9.94970E-01	3.50000E-03
2.59600E+00	9.94270E-01	2.14000E-03	2.35000E+00	9.91950E-01	3.55000E-03
2.45900E+00	9.98350E-01	1.34000E-03	2.59600E+00	9.93410E-01	1.93000E-03
2.45900E+00	9.96860E-01	1.36000E-03	2.59600E+00	9.91310E-01	2.05000E-03
2.45900E+00	9.99310E-01	1.24000E-03	4.73800E+00	9.95860E-01	4.36000E-03
2.45900E+00	9.97950E-01	1.36000E-03	4.73800E+00	9.93580E-01	4.53000E-03
2.45900E+00	9.97650E-01	1.38000E-03	4.73800E+00	9.95390E-01	4.58000E-03
2.45900E+00	9.96990E-01	1.35000E-03	4.73800E+00	9.92370E-01	4.54000E-03
2.45900E+00	9.97230E-01	1.37000E-03	4.73800E+00	9.91440E-01	4.62000E-03
2.45900E+00	9.96590E-01	1.40000E-03	4.73800E+00	9.98780E-01	4.82000E-03
2.45900E+00	9.95260E-01	1.40000E-03	4.73800E+00	9.94180E-01	4.94000E-03
2.45900E+00	9.97450E-01	1.36000E-03	4.73800E+00	9.92400E-01	4.90000E-03
2.45900E+00	9.97590E-01	1.38000E-03	4.73800E+00	9.96930E-01	4.98000E-03
2.45900E+00	9.97650E-01	1.36000E-03	4.73800E+00	9.91370E-01	5.05000E-03
2.45900E+00	9.98880E-01	1.39000E-03	2.35000E+00	9.92500E-01	2.34000E-03
2.45900E+00	9.97350E-01	1.37000E-03	2.35000E+00	9.95140E-01	2.43000E-03
2.45900E+00	9.97580E-01	1.39000E-03	2.35000E+00	9.92190E-01	2.33000E-03
2.45900E+00	9.97720E-01	1.39000E-03	2.35000E+00	9.94760E-01	2.40000E-03
2.45900E+00	9.96910E-01	1.35000E-03	2.35000E+00	9.94690E-01	3.67000E-03
4.30600E+00	9.95480E-01	2.84000E-03	2.35000E+00	9.94340E-01	2.49000E-03
4.30600E+00	9.93430E-01	2.78000E-03	2.35000E+00	9.93190E-01	2.39000E-03
4.30600E+00	9.93300E-01	2.81000E-03	4.73800E+00	9.93300E-01	1.28000E-03
4.30600E+00	9.93710E-01	2.85000E-03	4.73800E+00	9.93400E-01	1.23000E-03
4.30600E+00	9.95930E-01	2.73000E-03	4.73800E+00	9.94890E-01	1.25000E-03
4.30600E+00	9.92950E-01	2.85000E-03	4.73800E+00	9.93190E-01	1.25000E-03
4.30600E+00	9.96160E-01	2.89000E-03	4.73800E+00	9.93060E-01	1.28000E-03
4.30600E+00	9.93890E-01	2.73000E-03	2.45900E+00	9.91330E-01	2.03000E-03
4.30600E+00	9.95710E-01	2.96000E-03	2.45900E+00	9.95970E-01	2.43000E-03
4.30600E+00	9.93190E-01	2.60000E-03	2.45900E+00	9.95550E-01	2.42000E-03
4.30600E+00	9.93780E-01	2.75000E-03	2.45900E+00	9.94860E-01	2.42000E-03
4.30600E+00	9.92630E-01	2.84000E-03	2.45900E+00	9.95040E-01	2.42000E-03
4.30600E+00	9.95660E-01	2.75000E-03	2.45900E+00	9.95420E-01	2.42000E-03

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Appendix 8.A

31. Verify that the top of the closure lid is level (flush) with, or slightly above (< 0.25 inch), the top of the TSC shell.

32. At the discretion of the user, establish foreign material exclusion controls to prevent objects from being dropped into the annulus or TSC.

33. Install the welding system, including supplemental shielding if used, to the top of the closure lid.

    Note: At the discretion of the user, supplemental shielding may be installed around the transfer cask to reduce operator dose. Use of supplemental shielding shall be evaluated to ensure its use does not adversely affect the safety performance of the MPC-LACBWR system.

34. Connect a suction pump to the drain port quick-disconnect and verify venting through the vent port quick-disconnect.

35. Operate the suction pump to remove approximately 50 gallons of water from the TSC. Disconnect the suction pump.

    Note: The radiation level may increase as water is removed from the TSC cavity.

    Note: Fuel rods shall not be exposed to air during the 50-gallon pump-down.

36. Attach a hydrogen detector to the vent line. Ensure that the vent line does not interfere with the operation of the weld machine.

37. Sample the gas volume below the closure lid and observe hydrogen detector for H_2 concentration prior to commencing closure lid welding operations. Monitor H_2 concentration in the TSC until the closure lid-to-shell weld root pass is completed.

    Note: If H_2 concentration exceeds 2.4% prior to or during root pass welding operations, immediately stop welding operations. Evacuate the TSC gas volume or purge the gas volume with helium or argon. Verify H_2 levels are <2.4% prior to restarting welding operations.

38. Install shims into the closure lid-to-TSC shell gap, as necessary, to establish a uniform gap for welding. Tack weld the closure lid and shims, as required.

39. Operate the welding equipment to complete the closure lid-to-TSC shell root pass weld in accordance with the approved weld procedure.

40. At the completion of the root pass, discontinue use of the hydrogen detector. Leave the connector and vent tube installed to vent the canister.

Caution: It is necessary to continue to vent the cavity volume below the closure lid to ensure that there is no buildup of hydrogen gas until the completion of closure lid welding.

41. Perform visual and liquid penetrant (PT) examinations of the root pass and record the results.

42. Operate the welding equipment to perform the closure lid-to-shell weld to the midplane between the root and final weld surfaces. Perform visual and PT examinations for the midplane weld pass and record the results.

43. Complete welding through the completion of the final pass of the closure lid weld, perform final visual and PT examinations, and record the results.

44. Perform the hydrostatic test of the TSC as follows:

    a. Connect a drain line to the vent port and a pressure test system to the drain port.

    b. Refill the TSC with clean water until water is observed flowing from the vent port drain line. Close the vent line isolation valve.

    c. Pressurize the TSC to 15 (+2, -0) psig and isolate the TSC.

    d. Maintain the TSC test pressure for a minimum 10-minute hold period. At the end of the 10-minute hold period, visually examine the closure lid-to-TSC shell weld for water leakage, while maintaining the test pressure. The test pressure shall be maintained until the completion of the visual inspection of the closure lid-to-TSC shell weld is completed.

    e. The hydrostatic test is acceptable if there is no visible water leakage from the closure lid-to-TSC shell weld based on visual examination of the weld after a minimum 10-minute hold period, while maintaining the test pressure.

    f. Vent the TSC cavity and remove the pressure test system from the drain port and the drain line from the vent line. Reinstall a vent line to the vent port to prevent pressurization of the TSC.

45. Install and tack the closure ring in position in the closure lid-to-TSC shell weld groove.

46. Weld the closure ring to the TSC shell and to the closure lid. Perform visual and PT examinations of the final surfaces of the welds and record the results.

47. Remove the water from the TSC using one of the following methods: a) drain down using a suction pump supplemented with pressurized helium cover gas; or b) blow down using pressurized helium gas.

    Note: Fuel rods shall not be exposed to air during canister draining operations.

48. Connect a drain line with or without suction pump to the drain port connector.

49. Connect a regulated helium gas supply to the vent port connector.

Chapter 9
Appendix 9.A

9.A.2 Structural and Pressure Tests

9.A.2.1 Pressure Testing of the TSC

Following completion of the closure lid-to-TSC shell weld during the TSC preparation operations after fuel loading, the TSC shall be hydrostatically pressure tested in accordance with ASME Code, Section III, Subsection NB, NB-6000 requirements as described in operating procedures in Appendix 8.A. The minimum test pressure of 15 psig shall be applied to the drain port connection for a minimum 10-minute hold period. The minimum test pressure is 125% of the normal design pressure of 12 psig. There shall be no visible water leakage from the closure lid-to-TSC shell weld based on visual examination of the weld after a minimum 10-minute hold period, while maintaining the test pressure. Test pressure shall be maintained until the completion of the visual weld examination.

9.A.2.2 Transfer Cask Structural Load Testing

The Yankee-MPC Transfer Cask will be utilized for the loading operations of the MPC-LACBWR system. Prior to delivery to the La Crosse BWR site, the Yankee Transfer Cask lifting trunnions and bottom shield doors will be load tested in accordance with the requirements of ANSI N14.6 "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More" [A8] for nuclear materials.

The load tests shall consist of applying 300 percent of the maximum vertical loads for a minimum of 10 minutes to the lifting trunnions and bottom shield door components. The load tests will be performed in accordance with approved written procedures. The test conditions are:

	Transfer Cask Shield Doors		**Transfer Cask Trunnions**	
NAC-MPC System	**Canister Weight (lbs)**	**Test Load (lbs)**	**Loaded Cask Weight (lbs)**	**Test Load (lbs)**
MPC-LACBWR	65,875	197,625 (+5%, -0%)	146,625	439,875 (+5%, -0%)

Following completion of the lifting trunnion load tests, all trunnion welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Similarly, following completion of the bottom shield door load test, all door rail welds and all load bearing surfaces shall be visually inspected for permanent deformation, galling and cracking. Magnetic particle or liquid penetrant examinations, using the method appropriate to the material, shall be performed on accessible trunnion and shield door rail load bearing welds in accordance with

ASME Code Section V, Articles 1, 6 and/or 7, with acceptance in accordance with ASME Code Section III, NF-5340 or NF-5350, as applicable.

Any evidence of permanent deformation, cracking, galling of the load bearing surfaces or unacceptable liquid penetrant or magnetic particle examination results shall be cause for rejection of the affected component.

9.A.2.3 Damaged Fuel Can (DFC) Load Testing

The first MPC-LACBWR damaged fuel can (DFC) shall be load tested at the fabricator to 685 +50, -0 pounds to qualify the DFC design. The test load is calculated as 150% of the weight of the heaviest LACBWR fuel assembly (400 pounds) plus the weight of the damaged fuel can (55 pounds). The test load is applied and held for a minimum of 10 minutes. Following completion of the load test, all load bearing welds and surfaces shall be visually inspected for permanent deformation, galling or cracking. Load bearing welds shall be inspected using liquid penetrant examination in accordance with the ASME Code Section V, Article 6. Acceptance criteria shall be in accordance with ASME Code Section III, NG-5350.

9.A.2.4 Leakage Tests

The MPC-LACBWR confinement boundary is defined as the TSC shell weldment, closure lid, and inner vent and drain port covers. As described in Appendix 7.A.1, the confinement boundary is designed, fabricated, examined and tested in accordance with the requirements of the ASME Code, Section III, Subsection NB, except for the code alternatives listed in Table B.3-1 of the Technical Specifications.

Following welding, the TSC shell weldment shall be leakage tested using the evacuated envelope method as described in ASME Code, Section V, Article 10, and ANSI N14.5-1997 [A9] to confirm the total leakage rate is less than or equal to 1×10^{-7} ref. cm^3/s at an upstream pressure of 1 atmosphere absolute and a downstream pressure of 5 torr or less. Under these test conditions, this corresponds to a test leakage rate of 2×10^{-7} cm^3/s, helium at standard conditions.

The TSC shell weldment will be closed using a test lid installed over the top of the shell and the cavity evacuated with a vacuum pump to a vacuum of 5 torr or less. A test envelope will be installed around the TSC enclosing all of the TSC shell confinement welds, evacuated and backfilled to approximately 1 atmosphere absolute with 99.995% (minimum) pure helium. The

9.A.2.7.1 Neutron Absorber Material Sampling Plan

The neutron absorber sampling plan is selected to demonstrate a 95/95 (95% probability and 95% confidence level) statistical confidence level in the neutron absorber sheet material compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least 6 measurements (along the edges near each corner and the longitudinal centerline) on each sheet. No rejected neutron absorber sheet is used. The sampling plan is supported by written and approved procedures.

The sampling plan requires that a coupon sample be taken from each sheet of the first set of 50 sheets of absorber material. Thereafter, coupon samples are taken from 10 randomly selected sheets from each set of 50 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion), or a process change, at which time the sampling process is reinitiated as previously described. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

9.A.2.7.2 Wet Chemistry Test Performance

An approved facility with chemical analysis capability shall be selected to perform the wet chemistry tests. The tests will verify the presence of boron and provide input to the calculation of the ^{10}B areal density. Chemical test methods will be calibrated against neutron absorption methods for the BORAL® neutron absorber.

The most common method of verifying the acceptability of neutron absorber material is the wet chemistry method—a chemical analysis where the aluminum is separated from a sample with known thickness and volume. The remaining boron-carbide material is weighed and the areal density of ^{10}B is computed. A statistical conclusion about the BORAL® sheet from which the sample was taken and that batch of BORAL® sheets may then be drawn based on the test results and the established manufacturing processes previously noted.

9.A.2.7.3 Neutron Absorption Test Performance

An approved facility with a neutron source and neutron detection capability shall be selected to perform the described tests, if the neutron absorption test method is used. The tests will assure that the neutron absorption capacity of the material tested is equal to, or higher than, the given

reference value and will verify the uniformity of boron distribution. The principle of measurement of neutron absorption is that the presence of boron results in a reduction of neutron flux between the thermalized neutron source and the neutron detector—depending on the material thickness and boron content.

Determination of neutron absorber material acceptance shall be performed by neutron attenuation testing. Neutron attenuation testing of the final product or the coupons shall compare the results with those for calibrated standards composed of a homogeneous ^{10}B compound. Other calibrated standards may be used, but those standards must be shown to be equivalent to a homogeneous standard. These tests shall include a statistical sample of finished product or test coupons taken from each lot of material to verify the presence, uniform distribution and the minimum areal density of ^{10}B.

9.A.2.7.4 Acceptance Criteria

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube drawings. The neutron absorption test shall be considered acceptable if the neutron count determined for each test specimen is less than or equal to the highest permissible neutron count rate determined from the BORAL standard, which is based on the ^{10}B areal density specified on the fuel tube drawings. Any specimen not meeting the acceptance criteria for either test method shall be rejected. Nonconforming material shall be evaluated within the NAC International QA Program and shall be assigned one of the following dispositions: "use-as-is," "rework" or "reject." Only material that is determined to meet all applicable conditions of the license will be accepted.

9.A.2.8 Thermal Tests

No thermal acceptance testing of the MPC-LACBWR system is required during fabrication and construction. The thermal analysis of the MPC-LACBWR system in Appendix 4.A, which is limited to a maximum fuel content decay heat load of 4.5 kW, confirms that thermal monitoring of the MPC-LACBWR system in storage operations is not required to ensure system safety.

9.A.2.9 Cask Identification

Each MPC-LACBWR concrete cask will be appropriately identified by a stamped stainless steel nameplate as shown on Drawing No. 630045-864 that will be affixed to the outer surface of the cask.

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Appendix 10.A

- Annual visual inspection of the general condition of the concrete casks. This inspection is estimated to require 10 minutes per cask and require one technician. For each cask, three minutes of health physics support is also included.

- Corrective maintenance. As the MPC-LACBWR is a passively cooled and shielded system, no significant maintenance is expected over the lifetime of the IFSFI. To account for activities such as minor concrete repairs, air inlet and outlet cleaning, or temperature-monitoring equipment replacement, 20% of the array (one cask) is assumed to require maintenance each year. Maintenance exposure is evaluated based on two operators for 30 minutes each and one health physics technician for 10 minutes.

- Grounds maintenance performed twice a month by one maintenance technician. Grounds maintenance is assumed to require 60 minutes.

Storage operation exposures for a 5-cask array of MPC-LACBWR concrete casks loaded with TSCs containing bounding fuel assembly sources are presented in Table 10.A.3-2. ISFSI exposures are based on design basis casks.

Table 10.A.3-1 Estimated Person-mrem Exposure for Loading Operations of a Single MPC-LACBWR Storage Cask

Step	Description	# Sub-Tasks	# Personnel	Exposure Duration (min)	Average Dose Rate (mrem/hr)	Exposure (mrem)
1	Fuel Assembly Loading and Transfer Cask Removal from Pool	4	2	1605	2.0	108
2	HP Survey and Decon Top of TSC/Transfer Cask	3	1	30	8.0	4
3	Install Weld Shield/Weld Machine, and Perform Partial Drain of TSC	4	1	45	5.3	4
4	Perform Closure Lid and Ring Welding and PT Exams, Hydrostatically Test TSC	16	1	480	8.1	65
5	Drain TSC and Decontaminate Transfer Cask	5	1	230	4.2	16
6	Dry TSC Cavity, Backfill/Pressure TSC, Install Port Covers, Weld and Inspect Covers, Remove Weld Shield/Weld Machine, and Survey Cask/TSC Surfaces	13	1	505	6.5	55
7	Install Hoist Rings, Place Transfer Cask on Concrete Cask, Transfer TSC, Install Concrete Cask Lid, and Perform HP Survey	17	2	220	18.7	137
8	Move Concrete Cask to ISFSI, Position Concrete Cask on ISFSI Pad, and Install/Connect Screens and Temperature Measuring System	11	2	180	2.2	13
	Total					402

Chapter 11
Appendix 11.A

load from the canister contents is applied as a uniform pressure load acting on the inside surface of the cansiter bottom plate. The horizontal loading from the canister contents includes an acceleration load of 0.7071*g* to account for the support disks that are included in the model and applied forces to account for the remainder of the canister contents. The horizontal load from the basket components that are not modeled (i.e., fuel assemblies, fuel tubes, and heat transfer disks) is assumed to be evenly distributed between the support disks that are included in the model.

A design off-normal internal pressure is calculated based on the condition of 10 percent fuel rod failure with 70 percent fission gas release at a conservative gas temperature of 475°K. The calculated off-normal internal pressure is 11.4 psig. A bounding design internal pressure load of 12 psig is applied to the inside surfaces of canister shell, bottom plate, and lid elements in combination with dead weight and off-normal handling loads.

The linearized membrane and membrane plus bending stress components and stress intensities at each canister stress section identified in Figure 3.A.4.4.1-2 due to the combined dead weight, off-normal handling, and normal internal pressure loads are summarized in Tables 11.A.1.2.2-1 and 11.A.1.2.2-2, respectively. The allowable stresses and the corresponding margins of safety are also summarized in Tables 11.A.1.2.2-1 and 11.A.1.2.2-2. The results of the analysis show that the MPC-LACBWR canister satisfies the applicable allowable stress design criteria for the off-normal handing event.

As noted in Table 11.A.1.2.2-2, the bending stress at the bottom plate-to-shell junction is classified as secondary stress in accordance with Table NB-3217-1 if the edge moment is not required to maintain the bending stress at the center of the bottom plate to within acceptable limits. To demonstrate this, the bending stress at the center of the bottom plate is determined using hand calculations (Roark), treating it as a simply-supported circular plate subjected to a uniform pressure load. For a combined pressure load of 28.2 psi from the 1.5*g* dead weight plus off-normal handling load (i.e., 16.2 psi pressure load) and design internal pressure (12 psig), the membrane plus bending stress at the center of the bottom plate is calculated to be 28.28 ksi. The allowable primary membrane plus bending stress intensity, based on Type 304 stainless steel properties at a bounding design temperature of 400°F, is 33.48 ksi. Therefore, the edge moment is not required to maintain the bending stress at the center of the bottom plate to within acceptable limits, and the design margin for primary membrane plus bending stress intensity at the center of the bottom plate is 0.18 (= 33.48/28.28 – 1).

The maximum stresses in the MPC-LACBWR basket support disks due to the off-normal handling loads are calculated using the finite element model described in Section 3.A.4.4.2.1.

The off-normal handling horizontal acceleration is assumed to act along the +X direction. The support disk model nodes located at the centerline of each tie-rod are restrained in the vertical direction (i.e., UY=0).

The inertial loading due to the support disk self weight is accounted for by applying a horizontal acceleration loads of 0.7071*g* and a vertical acceleration load of 1.5*g* to the model. The horizontal load from the basket components that are not modeled (i.e., fuel assemblies, fuel tubes, heat transfer disks, and tie-rods) is applied as line loads on the supporting disk ligaments.

The linearized membrane and membrane plus bending stress intensities are evaluated at all critical sections of the support disk. All membrane and membrane plus bending stress intensities are conservatively classified as primary stresses and compared to the off-normal condition (i.e., Service Level C) allowable primary membrane (P_m) and membrane plus bending (P_m+P_b) stress intensity limits. The maximum calculated P_m and P_m+P_b stress intensities in the MPC-LACBWR basket assembly support disk due to the combined off-normal handling plus dead weight loading are summarized in Table 11.A.1.2.2-3 along with the corresponding accident condition allowable stresses and margins of safety. The result of the analysis show that MPC-LACBWR basket support disks satisfy the applicable allowable stress design criteria for the off-normal handing event.

11.A.1.2.3 Radiological Consequences

There are no radiological consequences for the canister off-normal handling event.

11.A.1.2.4 NAC-MPC Performance

This evaluation of the MPC-LACBWR canister for the off-normal handling event shows that the maximum stresses in the canister and fuel basket satisfy the applicable allowable stress design criteria. This event does not cause any damage to the MPC-LACBWR canister that would affect its ability to perform its design functions.

11.A.1.2.5 Recovery and/or Corrective Actions

Operations should be halted until the cause of the misalignment, interference or faulty operation is identified and corrected. Since the radiation level of the canister sides and bottom is high, extreme caution should be exercised if inspection of these surfaces is required.

11.A.2 Accident Conditions for the MPC-LACBWR Storage System

This section presents the evaluation of the MPC-LACBWR storage system for postulated accident events whose consequences could have the maximum potential impact on the immediate environment, and very low probability natural phenomena events that might occur once during the lifetime of an ISFSI. The analyses of the MPC-LACBWR storage system for accident conditions show that the owner-controlled area boundary dose limits of 10 CFR 72.106 are satisfied. In addition, the accident evaluation of the MPC-LACBWR storage system demonstrates that occupational exposure can be adequately controlled.

11.A.2.1 Accident Pressurization

This section describes the evaluation of the MPC-LACBWR storage system for an accident pressurization event in which all fuel rods in the canister are postulated to fail and release their fission and fill gases into the canister. The evaluations of the canister internal pressure load resulting from this condition and the stresses in the canister due to that internal pressure load are presented in this section. There are no radiological consequences for the accident pressurization event and no corrective actions are required for recovery from the event.

11.A.2.1.1 MPC-LACBWR Canister Maximum Internal Pressure

The analysis requires the calculation of the free volume of the canister, calculation of the quantity of fill and fission gas in the 68 fuel assemblies, and the subsequent calculation of the pressure in the canister if these gases are added to the helium pressure (initially at 1 atm) already present in the canister (Section 4.A.3.5). The quantity of fission gases was conservatively estimated assuming that 70% of the total gases present are released from the fuel. The bulk temperature of the fill gas is conservatively taken to be 400°F. This temperature bounds the calculated bulk temperature of fill gas for any of the evaluated normal, off-normal, or accident conditions. The internal pressure is a function of rod-fill, fission and canister backfill gases. All of the gases, except the fission gases, are helium. The total pressure for each volume are found by calculating the molar quantity of each gas and summing those directly. The design basis fuel assembly for the internal pressure calculation is the Allis Chalmers 10×10 fuel assembly. This assembly has the highest fuel mass (0.1201 MTU) and received the highest burnup (22,000 MWd/MTU).

The number of moles of the backfill gases is calculated using the Ideal Gas Law, PV = NRT. Backfill gases for the canister and cavity are assumed to be initially at 1 atmosphere. The

quantity of fission gas is derived from the SAS2H source term evaluation of the Allis Chalmers fuel assembly. The number of moles of gas in the canister is:

$$N = N_{TSC\ Back\text{-}Fill} + N_{Rod\ Back\text{-}Fill} + 0.70(N_{Fission\ Gas})$$

The number of moles of helium contained in the canister as backfill and the number of moles of gas in the fuel rods (as helium backfill and fission products) are calculated in Section 4.A.3.5.

Based on an assumed bounding temperature of 400°F, the maximum pressure in the canister is 3.18 atm ≈ 46.8 psia ≈ 32.1 psig.

11.A.2.1.2 MPC-LACBWR Canister Maximum Stress Due to Internal Pressure

The stresses in the MPC-LACBWR canister due to the combined loading of dead weight, normal handling, and accident pressurization are calculated using the finite element model described in Section 3.A.4.4.1.1 and shown in Figure 3.A.4.4.1-1. A bounding uniform pressure load of 32.1 psig is conservatively applied to the inside surfaces of canister shell, bottom plate, and shield lid elements. In addition, the 1.1*g* vertical acceleration loading for combined effects of dead weight and normal handling are applied to the model as described in Section 3.A.4.4.1.4.

The membrane and membrane plus bending stresses in the MPC-LACBWR canister due to the combined dead weight, normal handling, and 32.1 psig internal pressure loads are evaluated at the stress sections shown in Figure 3.A.4.4.1-2. The results show that the maximum primary membrane (P_m) and primary membrane plus bending (P_m+P_b) stress intensities in the canister due to the 32.1 psig internal pressure load, which both occur at the bottom plate-to-shell junction, are 15.19 ksi and 61.88 ksi, respectively. The corresponding allowable P_m and P_m+P_b stress intensities for accident conditions are 45.65 ksi and 65.2 ksi, respectively. Therefore, the minimum margin of safety in the MPC-LACBWR canister for the accident pressurization event is +0.05.

11.A.2.1.3 Radiological Consequences

There are no radiological consequences for this accident.

11.A.2.1.4 MPC-LACBWR Performance

This analysis demonstrates that the canister performance is not significantly affected by the increase in internal pressure that results from the hypothetical rupture of all of the fuel rods contained in the canister. There is a positive margin of safety throughout the canister.

Property	Concrete Cask Tip-Over Model	
	Yankee - MPC	LACBWR - MPC
Soil Density	100 pcf	120 pcf
Soil Elastic Modulus	70,000 psi	10,000 psi
Concrete Pad Compressive Strength	4,000 psi	6,000 psi

The results of the MPC-LACBWR concrete cask tip-over analysis show that the peak accelerations at key locations of the concrete cask liner, which are used in the evaluation of the loaded canister/basket model (Section 11.A.2.12.2 and Section 11.A.2.12.3) are:

Axial Position (Distance from Cask Bottom End)	Canister/Basket Component at Location	Peak Acceleration
122.8 in.	Top Support Disk	21.9*g*
139.4 in.	Canister Top End	24.6*g*

11.A.2.12.2.2 Analysis of the MPC-LACBWR Canister and Basket

Canister Stress Analysis

The MPC-LACBWR canister is evaluated for the concrete cask tip-over condition using a plastic analysis in accordance with Appendix F of the ASME Code. The analysis is performed using the half-symmetry finite element model shown in Figure 11.A.2.12-1. This model is similar to the model used for the MPC-LACBWR canister normal condition stress analysis, as described in Section 3.A.4.4.1.1, but has a refined mesh, gap elements that are sized based on the concrete cask annulus, and plastic material properties.

A bi-linear kinematic hardening material model, defined by a yield strength and a tangent modulus, is used to perform a plastic analysis. The material model is based on the properties of the canister Type 304 stainless steel material at a temperature of 300°F, corresponding to the temperature of the lid weld region where the highest stresses are expected to occur. The yield strength is modeled as 21.4 ksi. The tangent modulus is taken as $(S_u - S_y)/\varepsilon_p = 10.95\times10^4$ psi, where S_u is the ultimate tensile strength (66.2 ksi) and ε_p is the ultimate strain (40%).

The canister tip-over stress analysis is conservatively performed using a transverse acceleration of 25*g* over the full length of the canister. Note that this corresponds to a DLF of 1.02 (25/24.6)

at the top of the canister, which is conservative since the canister lid (7 inches thick) region is stiff due to its monolithic structure. The inertial load of the basket assembly is modeled as a pressure load on the inside surface of the canister shell. The basket pressure load distribution is assumed to be uniform over the length of the canister shell that is in contact with the support disks and is assumed to vary linearly in the circumferential direction, from a maximum value at the symmetry plane to zero at an angle of 22° from the symmetry plane. In addition, a uniform internal pressure load of 12 psig is applied to the model.

The maximum primary membrane (P_m) and primary membrane plus bending (P_m+P_b) stress components and stress intensities in the MPC-LACBWR canister due to the storage cask tip-over load are summarized in Tables 11.A.2.12-1 and 11.A.2.12-2, respectively. In accordance with Appendix F of the ASME Code, the membrane stress is limited to $0.7S_u$ and the membrane plus bending stress is limited to $0.9S_u$. An additional factor of 0.8 is applied to the allowable stresses for the lid closure weld (Section 12). The results of the analysis show that the lowest margins of safety in the cansiter shell for P_m and P_m+P_b stress intensities both occur in the lid closure weld (Section 12 in Figure 3.A.4.4.1-2) and result from the storage cask tip-over load with internal pressure loading. The lowest margins of safety for P_m and P_m+P_b stress intensity are 0.40 and 0.63, respectively.

Support Disk and Weldment Stress Analysis

The MPC-LACBWR canister fuel basket is evaluated for the concrete cask tip-over conditions using the finite element model shown in Figure 11.A.2.12-2. This finite element model is similar to the model used for the MPC-LACBWR canister normal condition stress analysis, as described in Section 3.A.4.4.1.1, but uses a coarse mesh for the canister and includes the basket support disks in order to properly model the interaction between the basket and shell under transverse loading conditions. Modeling the interaction of the basket and canister shell does not require a detail element discretization of the weld region or the lid. It is only necessary for the model to represent the stiffness that the lid applies to the canister shell at the lid-canister shell junction.

Each of the basket support disks and the top weldment are modeled using elastic shell elements (SHELL63) with thickness input as real constants. The 15 support disks that are located in the bottom-end region of the canister, where tip-over loads are lowest, are modeled using the coarse mesh shown in Figure 11.A.2.12-3. The 11 support disks located in the canister top-end region, where the tip-over loads are highest, are modeled using the fine mesh shown in Figure 11.A.2.12-4. The top weldment disk is also modeled using a fine mesh, similar to that shown in Figure 11.A.2.12-4 for the support disk. Rather than modeling the basket tie-rods, each support disk is restrained in the axial direction (i.e., UY=0) at the nodes located at the centerline of each tie-rod. The interface between the basket and canister is modeled by CONTAC52

11.A.3 Design Basis Loading of the MPC-LACBWR Transportable Storage Canister

NAC has completed analyses to verify that the MPC-LACBWR canister design satisfies 10 CFR 71 loading conditions in response to 10 CFR 72.236 (m) requirements.

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11.A.4 Evaluation of Site-Specific Fuel Components

Site-specific fuel components are structures designed to hold fuel assemblies, individual fuel rods or fuel debris configurations that are not in a standard fuel assembly. The LACBWR spent fuel assemblies are classified as LACBWR undamaged fuel assemblies and LACBWR damaged fuel assemblies that may contain fuel debris.

This section presents the evaluation of the LACBWR site-specific components used with the MPC-LACBWR storage system.

End Drop Evaluation

This section presents the buckling evaluation for the LACBWR fuel rods. The analysis shows that the maximum stresses in the fuel remain below the yield strength in the design basis event and confirm that the fuel rods will return to their original configuration. An end drop orientation is considered with an acceleration of 21g, which subjects the fuel rods to axial loading. This 21g acceleration bounds the maximum end drop acceleration calculated for the 6-inch concrete cask end drop.

In the end drop orientation, the fuel rods are laterally restrained by the grids and come into contact with the fuel assembly base. The only vertical constraint for the fuel rod is the base of the assembly.

Fuel Assembly	Cladding Diameter (in)	Cladding Thickness (in)	Fuel Rod Pitch (in)	Maximum Gap Between Fuel Assembly and Fuel Tube Wall (in)
ENC 10 × 10	0.394	0.022	0.557	0.136

Review of the design basis fuel indicates that the largest bow between a straight fuel assembly and the basket fuel tube inner wall could be 0.136 inch. A bow of 0.25 inch is conservatively used in the model. The fuel is not explicitly modeled, but the fuel pellet weight is included with the clad by increasing the density of the clad elements as required. Properties are used for the fuel pellet and the stainless steel 348H clad as shown in the following table. The mechanical properties for the clad were obtained from the PIE examination and the associated comprehensive report of the results published in 1976 [A4].

	Modulus of Elasticity (10^6 psi)	Density (lb/in^3)
Rod Clad	25.3	0.291
Fuel Pellet	-	0.396

To implement a bow of 0.25 inch into the fuel assembly, a half-symmetry ANSYS model is used in which lateral forces are applied at the 3rd grid at the approximate mid-length of the fuel to develop a 0.25-inch lateral displacement. The fuel rod is simply supported at each end. The purpose of the ANSYS model and solution is to provide the coordinates of the clad for the LS-DYNA model. This is accomplished by obtaining a static solution with the ANSYS model, and then using the option to update the coordinates of the nodes with the displacements from the solution.

Figures 11.A.4-1 and 11.A.4-2 show the LS-DYNA model. Each grid is modeled using spring elements (with a stiffness of 1.0E6 lb/in) to maintain the spacing between the fuel rods at the grid locations. No restriction in axial movement or rotation at the attachment to the fuel rods is provided by the spring elements. The LS-DYNA model employs the same nodes and elements as the ANSYS model (with the incorporation of the 0.25-inch bow). The shell elements in LS-DYNA use additional integration points to ensure that the maximum shear stress at the surface of the shell elements is accurately computed. Elastic properties used in the LS-DYNA model are the same as those used in the ANSYS model. An initial downward velocity of 68.1 in/sec (corresponding to a 6-inch drop) is assigned to all nodes in the model. The deceleration applied to the base of the model has a duration of 0.0087 second and a 21g maximum value, which provide bounding acceleration for the 6-inch end drop of the concrete cask.

The LS-DYNA analysis was performed for a duration of 0.015 second to capture the response of the fuel after the 0.0087-second loading duration. Post-processing identifies the maximum shear stress occurring at the shell surface. The maximum shear stress result from LS-DYNA is factored by two to determine the maximum stress intensity. The maximum stress intensity in the fuel cladding is shown to be 7.83 ksi, which gives a factor of safety against yield of 4.73.

The temperature of the fuel at the bottom end of the basket is bounded by 662°F (350°C) and the static yield strength for irradiated stainless at 662°F is 37.0 ksi. This conservatively neglects any strengthening effect due to the dynamic loading.

The analysis results confirm that the LACBWR fuel will remain undamaged for design basis cask end drop load condition.

11.A.6 References

The references for the MPC-LACBWR storage system accident analysis are the same as those listed in MPC FSAR Section 11.6, and supplemented as follows:

[A1] ANSI/ANS-57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)," American Nuclear Society, May 1992.

[A2] NUREG/CR-6865, "Parametric Evaluation of Seismic Behavior of Freestanding Spent Fuel Drop Cask Storage Systems," Sandia National Laboratories, February 2005.

[A3] Tennessee Valley Authority, Division of Engineering Design, Thermal Power Project, Report No. CEB 77-46, "All Projects Steel to Concrete Coefficient of Friction Preliminary Tests," December 1977.

[A4] General Electric Report No. NEDC-12633, "Post-Irradiation Evaluation of Fuel Rods from the La Crosse Boiling Water Reactor," June 1976.

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Chapter 12
Appendix 12.A

CY-MPC RECONFIGURED FUEL ASSEMBLY	A stainless steel container, having external dimensions that are slightly larger than a standard Connecticut Yankee fuel assembly, that ensures criticality control geometry and which permits gaseous and liquid media to escape while minimizing dispersal of gross particulate. It may contain a maximum of 100 INTACT FUEL RODS or DAMAGED FUEL RODS, or FUEL DEBRIS from any Connecticut Yankee spent fuel assembly.
DAMAGED FUEL ASSEMBLY (applicable for Yankee-MPC and CY-MPC fuel assembly contents only)	A fuel assembly containing at least one DAMAGED FUEL ROD or that cannot be handled by normal means, or both. Yankee class fuel assemblies containing up to 20 fuel rod positions that are either missing or that are holding DAMAGED FUEL RODS.
DAMAGED FUEL ROD	DAMAGED FUEL ROD is a fuel rod with a known or suspected cladding defect greater than a hairline crack or pinhole leak.
FAILED ROD STORAGE CANISTER	A handling container for moving up to 60 individual INTACT FUEL RODS or DAMAGED FUEL RODS in stainless steel tubes into a CY-MPC DAMAGED FUEL CAN. The steel tubes are held in place by regularly spaced plates welded in an open stainless steel frame. The FAILED ROD STORAGE CANISTER, which is closed at the top end by a bolted closure and at the bottom by a welded plate to capture the fuel rods in the tubes, must be loaded in a CY-MPC DAMAGED FUEL CAN.
FORCED AIR COOLING	Air delivered to the bottom eight ports of the TRANSFER CASK at a minimum rate of 250 CFM and a maximum air temperature of 75°F for Yankee-MPC, and at a minimum rate of 375 CFM and a maximum air temperature of 75°F for CY-MPC. The canister must be backfilled with helium.
FUEL DEBRIS	FUEL DEBRIS is fuel in the form of particles, loose pellets, and fragmented rods or assemblies. FUEL DEBRIS may be loaded in a handling container.
HANDLING CONTAINER	A stainless steel device (or container) designed to facilitate handling and loading of FUEL DEBRIS, fuel rods and rod segments during DFC loading operations.

(continued)

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within the perimeter fence licensed for storage of spent fuel within NAC-MPC SYSTEMs (see also 10 CFR 72.3).
INTACT FUEL ASSEMBLY (applicable for Yankee-MPC and CY-MPC fuel assembly contents only)	INTACT FUEL ASSEMBLY is a fuel assembly without DAMAGED FUEL RODS. Connecticut Yankee fuel assemblies with missing fuel rods, or with missing fuel rods replaced with solid filler rods, or with structural damage, are considered INTACT FUEL ASSEMBLIES provided that they have no DAMAGED FUEL RODS. Yankee Class fuel assemblies with missing fuel rods replaced with Zircaloy or stainless steel rods, or with structural damage, are considered INTACT FUEL ASSEMBLIES provided they have no DAMAGED FUEL RODS.
INTACT FUEL ROD	INTACT FUEL ROD is a fuel rod without known or suspected cladding defects greater than a pinhole leak or hairline crack.
LACBWR DAMAGED FUEL ASSEMBLY	Spent nuclear fuel (SNF) that cannot fulfill its fuel-specific or system-related function. SNF is classified as a LACBWR DAMAGED FUEL ASSEMBLY under the following conditions. 1. There is visible deformation of the rods in the SNF assembly. Note: This is not referring to the uniform bowing that occurs in the reactor; this refers to bowing that significantly opens up the lattice spacing. 2. Individual fuel rods are missing from the assembly and the missing rods are not replaced by a solid dummy rod that displaces a volume equal to, or greater than, the original fuel rod. 3. The SNF assembly has missing, displaced or damaged structural components such that either radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch); or the assembly cannot be handled by normal means (i.e., crane and grapple). 4. Any SNF assembly that contains fuel rods for which reactor operating records (or other records or tests) cannot support the conclusion that they do not contain gross breaches. Note: Breached fuel rods with minor cladding defects (i.e., pinhole leaks or hairline cracks that will not permit significant release of particulate matter from the spent fuel rod) are classified as undamaged. 5. The SNF is no longer in the form of an INTACT FUEL ASSEMBLY (e.g., consists of or contains debris such as loose fuel pellets, individual fuel rods or rod segments).

(continued)

RETAINER	A retainer used for the Gulf United Nuclear Fuel (GUNF) lead test assemblies to retain the removable fuel rods within the fuel assembly.
STORAGE OPERATIONS	STORAGE OPERATIONS include all activities that are performed at the ISFSI, while an NAC-MPC SYSTEM containing spent fuel is located on the storage pad within the ISFSI perimeter.
STRUCTURAL DAMAGE	Damage to the fuel assembly that does not prevent handling the fuel assembly by normal means. STRUCTURAL DAMAGE is defined as partially torn, abraded, dented or bent grid straps, end fittings or guide tubes. The damaged grid straps or end fittings must continue to provide support to the fuel rods, as designed, and may not be completely torn or missing. Guide tubes cannot be ruptured and must be continuous between the upper and lower end fittings. Yankee and CY fuel assemblies with STRUCTURAL DAMAGE are considered to be INTACT FUEL ASSEMBLIES provided that they do not have failed or DAMAGED FUEL RODS.
TRANSFER CASK	TRANSFER CASK is a shielded lifting device that holds the CANISTER during LOADING and UNLOADING OPERATIONS and during closure welding, vacuum drying, leak testing, and non-destructive examination of the CANISTER closure welds. The TRANSFER CASK is also used to transfer the CANISTER into and from the CONCRETE CASK and into the transport cask.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all activities involved in transferring a loaded CANISTER from a CONCRETE CASK to another CONCRETE CASK, to a TRANSPORT CASK, or to an appropriate location for unloading.
TRANSPORTABLE STORAGE CANISTER (CANISTER)	TRANSPORTABLE STORAGE CANISTER is a container consisting of a tube and disk fuel basket in a cylindrical canister shell welded to a baseplate. When the shield lid with welded port covers and the structural lid are welded in place, or the closure lid with port covers is welded in place, the CANISTER provides the confinement boundary for the confined spent fuel.

(continued)

TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all activities involved in moving a loaded NAC-MPC CONCRETE CASK and CANISTER to and from the ISFSI. TRANSPORT OPERATIONS begin when the NAC-MPC SYSTEM is positioned on the transporter, and end when the NAC-MPC SYSTEM is at its destination and no longer on the transporter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all activities on a NAC-MPC SYSTEM to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the NAC-MPC SYSTEM is no longer secured on the transporter and end when the last fuel assembly is removed from the NAC-MPC SYSTEM.
VERTICAL CONCRETE CASK (CONCRETE CASK)	VERTICAL CONCRETE CASK is the cask that receives and holds the sealed CANISTER. It provides the gamma and neutron shielding and convective cooling of the spent fuel confined in the CANISTER.
WATER COOLING	Placement of the TRANSFER CASK holding an NAC-MPC CANISTER in the spent fuel pool. The canister must be backfilled with helium. WATER COOLING shall be maintained for a minimum of 24 hours, once initiated.
YANKEE-MPC	YANKEE-MPC is a NAC-MPC SYSTEM having a fuel basket designed to accommodate Yankee Class spent fuel. The YANKEE-MPC meets the requirements designated for the NAC-MPC SYSTEM.
YANKEE-MPC DAMAGED FUEL CAN	A stainless steel container that is similar to an enlarged fuel tube and that confines a Yankee Class INTACT FUEL ASSEMBLY, DAMAGED FUEL ASSEMBLY, RECAGED FUEL ASSEMBLY or a RECONFIGURED FUEL ASSEMBLY. A damaged fuel can is closed on its end by screened openings that allow gaseous and liquid media to escape, but minimize the dispersal of gross particulate. Use of the damaged fuel can requires that four cans be used in the canister in conjunction with a special shield lid machined to accept the cans.

(continued)

Chapter 12
Appendix 12.B

Table B.3-1 List of ASME Code Alternatives for the NAC-MPC SYSTEM (continued)

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification and Compensatory Measures
CANISTER Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The NAC-MPC SYSTEM is marked and identified in accordance with 10 CFR 72 requirements. Code stamping is not required. The QA data package will be in accordance with NAC's approved QA program.
CANISTER Basket Assembly	NG-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials are supplied by vendors approved under the NAC Quality Assurance Program. Materials are procured using ASME material specifications. CMTRs for the material are required in accordance with NG-2000.
CANISTER Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The NAC-MPC SYSTEM will be marked and identified in accordance with 10 CFR 72 requirements. No Code stamping is required. The CANISTER basket data package will be in accordance with NAC's approved QA program.
CANISTER Vessel and Basket Assembly Material	NB-2130/ NG-2130	States requirements for certification of material organizations and materials to NCA-3861 and NCA-3862, respectively.	The NAC-MPC CANISTER and Basket Assembly component materials are procured in accordance with the specifications for materials in ASME Code Section II with Certified Material Test Reports. The component materials will be obtained from NAC approved Suppliers in accordance with NAC's approved QA program.

B 3.4 Site Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the NAC-MPC SYSTEM user are, as a minimum, as follows:

1. The temperature of 75°F is the maximum average yearly temperature. The 3-day average ambient temperature shall be 100°F or less for YANKEE-MPC and CY-MPC, and 105°F or less for MPC-LACBWR.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than –40°F and less than 125°F.

3. a) For Yankee-MPC and CY-MPC, the design basis earthquake horizontal acceleration at the top surface of the ISFSI pad is 0.25g in each orthogonal direction and is (0.25g x 0.667 =) 0.167g in the vertical direction.

   b) For MPC-LACBWR, the design basis earthquake horizontal acceleration at the top surface of the ISFSI pad is 0.45g in each orthogonal direction and is (0.45g x 0.667 =) 0.30g in the vertical direction.

   c) Alternatively, the design basis earthquake motion of the ISFSI pad may be limited so that the acceleration g-load resulting from the collision of two sliding casks remains bounded by the accident condition analyses presented in Chapter 11 of the FSAR.

   Site-specific analysis by the cask user shall demonstrate that a cask does not slide off the ISFSI pad.

4. The analyzed flood condition of 15 fps water velocity and a height of 50 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the fuel tank of the cask handling equipment used to move the loaded CONCRETE CASK onto or from the ISFSI site contains no more than 50 gallons of fuel.

(continued)

Proposed
MPC-LACBWR
Technical Specifications
Appendix B

B 3.4 Site Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the NAC-MPC SYSTEM user are, as a minimum, as follows:

1. The temperature of 75°F is the maximum average yearly temperature. The 3-day average ambient temperature shall be 100°F or less for YANKEE-MPC and CY-MPC, and 105°F or less for MPC-LACBWR.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than –40°F and less than 125°F.

3. a) For Yankee-MPC and CY-MPC, the design basis earthquake horizontal acceleration at the top surface of the ISFSI pad is 0.25g in each orthogonal direction and is (0.25g x 0.667 =) 0.167g in the vertical direction.

   b) For MPC-LACBWR, the design basis earthquake horizontal acceleration at the top surface of the ISFSI pad is 0.45g in each orthogonal direction and is (0.45g x 0.667 =) 0.30g in the vertical direction.

   c) Alternatively, the design basis earthquake motion of the ISFSI pad may be limited so that the acceleration g-load resulting from the collision of two sliding casks remains bounded by the accident condition analyses presented in Chapter 11 of the FSAR.

   Site-specific analysis by the cask user shall demonstrate that a cask does not slide off the ISFSI pad.

4. The analyzed flood condition of 15 fps water velocity and a height of 50 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the fuel tank of the cask handling equipment used to move the loaded CONCRETE CASK onto or from the ISFSI site contains no more than 50 gallons of fuel.

(continued)