

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

Dominion Energy Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
DominionEnergy.com



May 3, 2018

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

10 CFR 50.90
Serial No.: 18-039
NRA/DEA R0
Docket No.: 50-423
License No.: NPF-49

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
LICENSE AMENDMENT REQUEST REGARDING PROPOSED TECHNICAL
SPECIFICATIONS CHANGES FOR SPENT FUEL STORAGE AND NEW FUEL STORAGE

Pursuant to 10 CFR 50.90, Dominion Energy Nuclear Connecticut, Inc. (DENC) is submitting a license amendment request to revise the following Technical Specifications (TS):

- TS 1.40 "Storage Pattern"
- TS 1.41 "3-OUT-OF-4 and 4-OUT-OF-4"
- TS 3/4.9.1.2 "Boron Concentration"
- TS 3/4.9.13 "Spent Fuel Pool – Reactivity"
- TS 3/4.9.14 "Spent Fuel Pool – Storage Pattern"
- Figure 3.9-1 "Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 1 4-OUT-OF-4 Fuel Storage Configuration"
- Figure 3.9-2 "Region 1 3-OUT-OF-4 Storage Fuel Assembly Loading Schematic"
- Figure 3.9-3 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 2 Storage Configuration"
- Figure 3.9-4 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Pre-Uprate (3411 MWt) Cores"
- Figure 3.9-5 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 MWt) Cores"
- TS 5.6.1.1 "Criticality."

Conforming changes to the associated Technical Specifications Bases (TSB) are also included in this license amendment request for information only.

DENC performed a criticality safety evaluation for fuel assembly storage in the Millstone Power Station Unit 3 (MPS3) Spent Fuel Pool (SFP) storage racks and New Fuel Storage Racks (NFSR) to support the proposed TS change using a new methodology described in Attachment 5.

A001
NRK

Attachment 5 contains information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation from Attachment 5, this letter is decontrolled.

This proposed amendment implements the following conditions associated with fuel assembly storage at MPS3:

- Increase the TS minimum SFP soluble boron concentration.
- Revise allowed storage patterns and initial enrichment/burnup/decay time for fuel assemblies in the SFP to meet k_{eff} requirements under normal and accident conditions as follows:
 - Region 1 will have an updated configuration as follows:
 - Each Region 1 fuel storage rack will contain storage locations designated as Region 1A and Region 1B.
 - Region 1A – Region 1A can store fresh fuel with an enrichment ≤ 4.75 wt% U-235 and no integral burnable poison without restriction.
 - Region 1A – Region 1A can store fresh fuel with an enrichment ≤ 5.0 wt% U-235 and with a minimum of twelve (12) Integral Fuel Burnable Absorber (IFBA) rods.
 - Region 1A – Region 1A can store fuel with an enrichment ≤ 5.0 wt% U-235 and with a minimum of 2.0 GWd/MTU burnup.
 - Region 1B – Region 1B can store fresh fuel with an enrichment ≤ 5.0 wt% U-235 and no integral burnable absorber without restriction.
 - Regions 2 and 3 will have new burnup curves. The Region 3 burnup curve will include credit for decay time.
 - The proposed change will permit the storage of any fuel assembly with an enrichment ≤ 5.0 wt% U-235 that contains a Rod Cluster Control Assembly (RCCA) in Region 2 without restriction (e.g., fresh fuel with no integral burnable absorber)
 - For consistency, the criticality analysis for the New Fuel Storage Racks is also being revised using the updated methods for the SFP criticality analysis. This reanalysis will not require a TS change.

Information provided in the attachments to this letter is summarized below:

- Attachment 1 provides an evaluation of the proposed TS changes.
- Attachment 2 provides proposed marked-up TS pages.
- Attachment 3 provides marked-up TS Bases pages (for information only).
- Attachment 4 contains the Criticality Safety Evaluation Checklist.
- Attachment 5 contains the Criticality Safety Evaluation Report (Proprietary).
- Attachment 6 contains the Criticality Safety Evaluation Report (Non-proprietary).
- Attachment 7 contains the MPS3 Boron Dilution Analysis
- Attachment 8 contains the Holtec International, Inc. (Holtec) affidavit for withholding proprietary information from public disclosure.

- Attachment 9 contains the Westinghouse Electric Company, LLC (Westinghouse) application and accompanying affidavit for withholding proprietary information from public disclosure.

Since Attachment 5 contains information proprietary to Holtec and Westinghouse, it is supported by affidavits signed by the owners of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390.

Accordingly, it is respectfully requested that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390.

DENC has evaluated the proposed amendment and determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is included in Attachment 1. DENC has also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite or any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change.

The proposed TS change has been reviewed and approved by the MPS3 Facility Safety Review Committee.

DENC requests approval of the proposed amendment by April 30, 2019. DENC plans to fully implement the revised TS within 90 days after NRC approval of this proposed license amendment.

Should you have any questions in regard to this submittal, please contact Wanda D. Craft at (804) 273-4687.

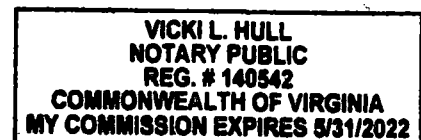
Sincerely,



Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA

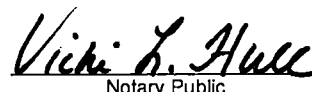
COUNTY OF HENRICO



The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support of Dominion Energy Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 3rd day of May, 2018.

My Commission Expires: 5-31-22


Notary Public

Commitments made in this letter: None

Attachments:

1. Discussion of Change
2. Marked-up Technical Specifications Pages
3. Marked-up Technical Specifications Bases Pages (for information only)
4. Criticality Analysis Checklist
5. Criticality Safety Evaluation Report - (Proprietary)
6. Criticality Safety Evaluation Report - (Non-proprietary)
7. Spent Fuel Pool Boron Dilution Analysis
8. Holtec, International Affidavit
9. Westinghouse Electric Company, LLC Application and Accompanying Affidavit

cc: U.S. Nuclear Regulatory Commission – Region I
Regional Administrator
2100 Renaissance Blvd.
Suite 100
King of Prussia, PA 19406-2713

Richard V. Guzman
Senior NRC Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 C2
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

Director, Radiation Division
Department of Energy and Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 8

HOLTEC, INTERNATIONAL AFFIDAVIT

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Kimberly Manzione, being duly sworn, depose and state as follows:

- (1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.
- (2) The information sought to be withheld is information provided in Letter Serial Number 18-039 from Dominion Energy Nuclear Connecticut, Inc. to the NRC "Dominion Energy Nuclear Connecticut Inc. Millstone Power Station Unit 3 License Amendment Request Regarding Proposed Technical Specifications Changes for Spent Fuel Storage and New Fuel Storage." This letter contains Holtec Proprietary information.
- (3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

AFFIDAVIT PURSUANT TO 10 CFR 2.390

-
- (4) Some examples of categories of information which fit into the definition of proprietary information are:
- a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;
 - b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
 - c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;
 - d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;
 - e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b and 4.e above.

- (5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as

AFFIDAVIT PURSUANT TO 10 CFR 2.390

proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

- (6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.
- (7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.
- (8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

-
- (9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
)
COUNTY OF CAMDEN) ss:

Kimberly Manzione, being duly sworn, deposes and says:

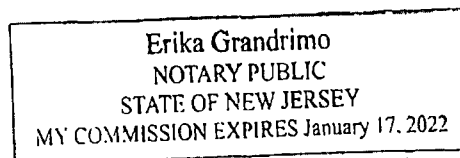
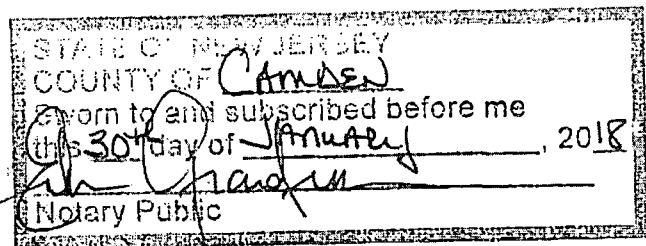
That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Camden, New Jersey, this 30th day of January, 2018.



Kimberly Manzione
Licensing Manager
Holtec International

Subscribed and sworn before me this 30th day of January, 2018.



Attachment 9

**WESTINGHOUSE ELECTRIC COMPANY, LLC
APPLICATION AND ACCOMPANYING AFFIDAVIT**

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**



Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8542
e-mail: greshaja@westinghouse.com

CAW-18-4731

April 4, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Millstone Unit 3 LAR Proprietary Class 2 Marked Pages (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4731 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Dominion.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4731, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2, Suite 259, Cranberry Township, Pennsylvania 16066.

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

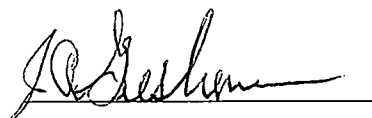
COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 4/4/18

A handwritten signature in black ink, appearing to read "J. A. Gresham", written over a horizontal line.

James A. Gresham, Manager
Regulatory Compliance

- (1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.
- (2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.
- (3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.
- (4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.
 - (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.
 - (ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

- (a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

- (b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).
 - (c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.
 - (d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.
 - (e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.
 - (f) It contains patentable ideas, for which patent protection may be desirable.
- (iii) There are sound policy reasons behind the Westinghouse system which include the following:
- (a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.
 - (b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.
 - (c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

- (d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.
 - (e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.
 - (f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.
- (iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.
- (v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.
- (vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in CE-18-139 Revision 1 –Attachment 1, “Millstone Unit 3 LAR Proprietary Class 2 Marked Pages” (Proprietary), for submittal to the Commission, being transmitted by Dominion letter. The proprietary information as submitted by Westinghouse is that associated spent fuel pool storage information for criticality analyses, and may be used only for that purpose.
- (a) This information is part of that which will enable Westinghouse to perform spent fuel pool analyses.
 - (b) Further, this information has substantial commercial value as follows:
 - (i) Westinghouse plans to sell the use of similar information to its customers for the purpose of spent fuel pool analysis.

- (ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.
- (iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Dominion

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. "Millstone Unit 3 Spent Fuel Pool Analysis Proprietary Marking Identification" (Proprietary)
2. "Millstone Unit 3 Spent Fuel Pool Analysis Proprietary Marking Identification" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-18-4731, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-18-4731 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2, Suite 259, Cranberry Township, Pennsylvania 16066.

Attachment 1

DISCUSSION OF CHANGE

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

Attachment 5 contains information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation from Attachment 5, this letter is decontrolled.

Discussion of Change

Table of Contents

1.0	Summary Description.....	2
2.0	Detailed Description of Change	3
2.1	TS 1.40 – Storage Pattern and TS 1.41 – 3-OUT-OF-4 and 4-OUT-OF-4.....	3
2.2	TS 3/4.9.1.2 - Boron Concentration	3
2.3	TS 3/4.9.13 – Spent Fuel Pool - Reactivity	3
3.0	Discussion	7
3.1	Current MPS3 Spent Fuel Pool Configuration	8
4.0	Technical Evaluation Summary.....	9
4.1	Introduction	9
4.2	Fuel Storage Criticality Analysis - General	10
4.3	New Fuel Storage Racks Criticality Analysis – Normal Storage and Accident Conditions.....	10
4.4	Spent Fuel Pool Criticality Safety Evaluation - Normal Storage Conditions.....	10
4.5	Spent Fuel Pool Criticality Safety Evaluation - Accident Conditions	11
4.6	Boron Dilution	11
4.7	Storage of Non-fuel Components and Non-standard Fuel Assemblies.....	12
4.8	Spent Fuel Pool Storage – Other Items	12
4.9	Implementation Considerations	13
4.10	Conclusions	13
5.0	Regulatory Evaluation.....	14
5.1	Applicable Regulatory Requirements and Criteria	14
5.2	No Significant Hazards Consideration	14
5.3	Precedents.....	18
5.4	Conclusion	19
6.0	Environmental Considerations	20
7.0	References	20

Attachment 5 contains information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation from Attachment 5, this letter is decontrolled.

1.0 Summary Description

Dominion Energy Nuclear Connecticut, Inc. (DENC) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Technical Specifications (TS) of the Millstone Power Station Unit 3 (MPS3). DENC is proposing to change the following TS:

- TS 1.40 "Storage Pattern"
- TS 1.41 "3-OUT-OF 4 and 4-OUT-OF-4"
- TS 3/4.9.1.2 "Boron Concentration"
- TS 3/4.9.13 "Spent Fuel Pool – Reactivity"
- TS 3/4.9.14 "Spent Fuel Pool – Storage Pattern"
- Figure 3.9-1 "Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 1 4-OUT-OF-4 Fuel Storage Configuration"
- Figure 3.9-2 "Region 1 3-OUT-OF-4 Storage Fuel Assembly Loading Schematic"
- Figure 3.9-3 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 2 Storage Configuration"
- Figure 3.9-4 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Pre-Uprate (3411 MWt) Cores"
- Figure 3.9-5 "Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 MWt) Cores"
- TS 5.6.1.1 "Criticality."

The associated Bases for TS 3/4.9.1.2, 3/4.9.13, and 3/4.9.14 are also being modified to address the proposed changes and are provided for information only. Changes to the TS Bases are controlled in accordance with the MPS TS Bases Control Program.

The Region 3 Spent Fuel Pool (SFP) storage racks contain Boraflex which is currently not credited as a neutron absorber. This proposed change will not credit Boraflex. Regions 1 and 2 racks contain BORAL[®] as a neutron absorber which will continue to be credited.

To meet the SFP criticality requirements, the following changes are being proposed:

- Revise allowed storage patterns for fuel assemblies in the SFP to meet k_{eff} requirements under normal and accident conditions:
 - Region 1 will no longer require the use of cell blocking devices.
- Update the surveillance requirements for storing fuel assemblies in Region 1, Region 2, and Region 3.
- Include new burnup curves for Regions 2 and 3. The Region 3 burnup curve will include credit for decay time.

The proposed change has been reviewed and confirmed to accommodate all fuel currently in the SFP and New Fuel Storage Racks (NFSR), and fuel assembly designs anticipated in the future.

2.0 Detailed Description of Change

2.1 TS 1.40 – Storage Pattern and TS 1.41 – 3-OUT-OF-4 and 4-OUT-OF-4

These definitions are being deleted. They apply to the required fuel storage configuration imposed by cell blocking devices in designated Region 1 storage locations. Since the proposed change will remove the requirement for cell blocking devices, these definitions will no longer apply.

2.2 TS 3/4.9.1.2 - Boron Concentration

The proposed change to TS 3/4.9.1.2 will increase the licensed minimum SFP boron concentration from 800 ppm to 2600 ppm.

LCO 3.9.1.2 and Surveillance Requirement 4.9.1.2

LCO 3.9.1.2 is updated to increase the licensed minimum SFP boron concentration from 800 ppm to 2600 ppm. The updated boron concentration will maintain $k_{\text{eff}} \leq 0.95$ under the bounding postulated accident conditions, including allowance for biases and uncertainties.

2.3 TS 3/4.9.13 – Spent Fuel Pool - Reactivity

The TS title is being changed to "Spent Fuel Pool – Storage" to address the new SFP storage configurations and requirements. The criticality safety evaluation demonstrates that the SFP will maintain $k_{\text{eff}} < 1.0$ in an unborated water environment assuming each SFP rack storage location contains a fuel assembly with the highest reactivity allowed for that location. Also, the criticality safety evaluation demonstrates that the SFP will maintain $k_{\text{eff}} \leq 0.95$ at all times if the SFP soluble boron concentration is ≥ 2600 ppm; thus, existing TS 3.9.13.a, which regards borating the SFP, will be deleted because the SFP boron concentration requirements will now be solely governed by TS 3/4.9.1.2.

Existing TS 3.9.13.b refers to four (4) burnup curves. These curves are being replaced with two (2) burnup curves, one for Region 2 and one for Region 3. The Region 3 burnup curve includes credit for decay time.

For simplicity, the proposed new burnup curves will now be designated as Figures 3.9-2 and 3.9-3. Figure 3.9-1 will become the new Fuel Assembly Loading Schematic for Region 1 spent fuel racks (this figure designates which storage locations are Region 1A and Region 1B). TS 3/4.9.14, which specifies installation and removal of cell blocking devices, is being deleted since the proposed change will no longer require usage of cell blocking devices.

Two items need to be defined regarding enrichment and burnup. Initial enrichment, when used to compare to fuel storage requirements, is the "maximum initial planar volume averaged as-built U-235 enrichment" in the assembly. If the assembly has axial blankets the lower enriched fuel is not credited in determining the enrichment.

Also, fuel burnup when used to compare to fuel storage requirements is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

Each Region 1 rack will designate each storage location as a Region 1A or Region 1B storage location. Region 1B storage locations will be permitted to store fresh fuel with an initial enrichment ≤ 5.0 wt% U-235 and with no burnable absorber. Region 1B is defined in the proposed Fuel Assembly Loading Schematic in Figure 3.9-1. In Region 1, the two rows adjacent to the west SFP wall are designated Region 1B, and the remaining locations designated as Region 1A.

Fuel may be stored in a Region 1A storage location if it meets one of the following criteria:

- Initial enrichment ≤ 4.75 wt% U-235, or
- Initial enrichment ≤ 5.0 wt% U-235 and fuel burnup (measured) ≥ 2.0 GWD/MTU, or
- Initial enrichment ≤ 5.0 wt% U-235 and contains a minimum of twelve (12) Integral Fuel Burnable Absorber (IFBA) rods.

There will no longer be a burnup curve specifically for pre-uprate fuel assemblies. The proposed change accommodates a rated thermal power greater than any historical MPS3 cycle. These changes will also be reflected in the proposed changes to the TS Surveillance Requirements.

Changes to the TS are as follows:

LCO 3.9.13

Action a. will now only refer to Surveillance Requirement 4.9.13.1.1, Action b. will refer to Figure 3.9.2 (Region 2 burnup curve), and Action c. will refer to Figure 3.9.3 (Region 3 burnup curve), and will require immediate action if an assembly does not meet the relevant requirements. In Region 2, this LCO only applies to fuel assemblies that do not contain a RCCA. Action a. in the existing TS will be deleted since the proposed change to TS 3/4.9.1.2 will provide the requirement to borate the SFP should boron concentration fall below 2600 ppm.

Region 1A requirements include:

- Fuel assemblies with enrichment ≤ 4.75 wt% U-235 may be stored in Region 1A.
- A fuel assembly with enrichment > 4.75 wt % and ≤ 5.0 wt% U-235 may be stored in a Region 1A storage location if fuel burnup is ≥ 2.0 GWD/MTU, or if it contains at least 12 IFBA rods.

Region 1B requirements include:

- Fuel assemblies with enrichment ≤ 5.0 wt% U-235 may be stored in Region 1B.

Region 2 requirements include:

- Fuel assemblies with a combination of initial enrichment and burnup in the "Acceptable" domain of the new burnup curve (Figure 3.9-2) may be stored in Region 2.
- A fuel assembly with enrichment ≤ 5.0 wt% U-235 that contains a RCCA may be stored in Region 2.

Region 3 requirements include:

- Fuel assemblies with a combination of initial enrichment, burnup, and decay time in the "Acceptable" domain of the new burnup curve (Figure 3.9-3) may be stored in Region 3.
- Decay Time is defined as the time elapsed since a fuel assembly was last used at power in a reactor core. A burnup curve is applicable to a fuel assembly if the fuel assembly decay time is greater than or equal to the burnup curve decay time.

Each burnup curve in Figures 3.9-2 and 3.9-3 is defined by a set of polynomial fit coefficients with the following format:

$$BU [GWD/MTU] = a_4 * wt\%^4 + a_3 * wt\%^3 + a_2 * wt\%^2 + a_1 * wt\%^1 + a_0$$

wt%: maximum initial planar volume averaged, as-built U-235 enrichment.

SR 4.9.13

Surveillance Requirements (SR) 4.9.13.1.1, 4.9.13.1.2, 4.9.13.1.3, and 4.9.13.1.4 are updated to instruct personnel to confirm that (1) initial enrichment, burnup, IFBA loading (Region 1), decay time (Region 3), and location of assemblies are acceptable based on the requirements in new Figures 3.9.1, 3.9.2, 3.9.3, and the Region 1A restrictions in Surveillance Requirement 4.9.13.1.1 or (2) that the assembly contains a RCCA for storage in Region 2.

TS Figure 3.9-1

TS Figure 3.9-1 is replaced. The new figure shows the bounding Region 1 fuel storage loading schematic, including which storage locations are designated as Region 1A and 1B.

TS Figure 3.9-2

TS Figure 3.9-2 is replaced. This new figure shows the minimum required fuel assembly burnup as a function of initial enrichment to permit storage in Region 2 (for assemblies that do not contain a RCCA).

TS Figure 3.9-3

TS Figure 3.9-3 is replaced. This new figure shows the minimum required fuel assembly burnup and decay time as a function of initial enrichment to permit storage in Region 3.

TS Figure 3.9.4

TS Figure 3.9-4 is deleted.

TS Figure 3.9-5

TS Figure 3.9-5 is deleted.

TS 3/4.9.14

TS 3/4.9.14 is deleted. Cell blocking devices will no longer be required.

TS 5.6.1.1

The proposed change to TS 5.6.1.1 will update the summary of the fuel storage configuration and burnup/decay time requirements for each region consistent with the proposed TS changes above. This update will also indicate that k_{eff} will remain less than 1.0 with no soluble boron in the SFP water and ≤ 0.95 with credit for soluble boron. TS 5.6.1.1 currently indicates that k_{eff} will remain less than ≤ 0.95 with no soluble boron in the SFP water. A description of the NFSR will be added to TS 5.6.1.1, including that the NFSR will maintain $k_{\text{eff}} \leq 0.95$ under flooded conditions and ≤ 0.98 under optimum moderator conditions.

3.0 Discussion

DENC proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the MPS3 TS. A summary of the proposed changes is provided in Section 1.0 above. The supporting criticality safety evaluation report is included as Attachment 5 (Proprietary) and Attachment 6 (Non-proprietary).

Three unique regions have been analyzed for the SFP, each with a somewhat different rack design.

- Region 1 racks have a flux trap design and contain BORAL[®] neutron absorber panels. Region 1 racks are sub-divided into Region 1A and Region 1B. Region 1B occupies two rows of Region 1 rack storage cells adjacent to the west SFP wall. Region 1B credits neutron leakage at the interface with the Region 1 wall.
- Region 2 racks contain BORAL[®] neutron absorber panels, but do not use a flux trap design. A fuel assembly must meet the requirements of the Region 2 burnup curve to be stored in this region. However, any fuel assembly with initial enrichment ≤ 5.0 wt% U-235 that contains a RCCA may also be stored in Region 2 without restriction.
- Region 3 racks have a flux trap design and contain Boraflex neutron absorber material. Boraflex is not credited in the criticality safety analysis and is modeled as SFP water. A fuel assembly must meet the requirements of the Region 3 burnup curve, which credits decay time, to be stored in Region 3.

The rack design and the modeling of the racks are discussed in the criticality safety evaluation.

There are 1779 storage locations currently in the MPS3 SFP as follows:

- Region 1 (contains BORAL[®] as a poison)
 - Five 7 x 10 racks
- Region 2 (contains BORAL[®] as a poison)
 - Three 7 x 9 racks
 - One 7 x 10 rack
 - One 9 x 10 rack
 - Four 9 x 9 racks
 - There is one 9 x 9 rack that is licensed per Reference 1 but has not been installed. The proposed change continues to also assume that this rack is installed.
- Region 3 (contains Boraflex, which is not credited)
 - Twenty-one 6 x 6 racks.

The criticality safety evaluation was performed for each region independently. The results verified that no adverse boundary effects occur at region interfaces. The analysis includes soluble boron credit. The present TS SFP boron concentration requirement of the SFP is 800 ppm of soluble boron. The proposed change will increase this requirement to 2600 ppm.

3.1 *Current MPS3 Spent Fuel Pool Configuration*

The MPS3 SFP contains 350 Region 1 storage locations, 673 Region 2 storage locations, and 756 Region 3 storage locations, for a total of 1779 fuel storage locations. An additional Region 2 rack with 81 storage locations is licensed to be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations. The total SFP storage capacity is limited to no more than 1860 fuel assemblies. The proposed license amendment will not make any changes to the licensed number of spent fuel racks or storage locations (1860).

Region 1 storage currently includes a 3-out-of-4 configuration which employs a cell blocker on the required empty cells. This configuration and the requirement for the use of cell blockers is eliminated in the proposed configuration.

4.0 Technical Evaluation Summary

4.1 Introduction

The analysis and results in this section are summarized from the criticality safety evaluation report (Attachment 5 [Proprietary]/Attachment 6 [Non-proprietary]). The topics presented here have the most significant impact with regard to the proposed license amendment.

The proposed license amendment does not result in any equipment modifications to the plant or any changes regarding how equipment is operated and maintained. There are no changes in how fuel assemblies are handled and moved, nor are there any changes in how they are inserted into or removed from a SFP or NFSR storage location. There are no changes to how RCCAs are handled and moved, nor are there any changes in how they are inserted into or removed from a fuel assembly. There is no change to how personnel qualify and verify fuel assembly storage in either the SFP or the NFSR. The existing four burnup curves are being replaced with two burnup curves, one for Region 2 and one for Region 3. The Region 3 burnup curve will credit decay time. The burnup curves no longer distinguish between pre-uprate and post-uprate fuel and apply to all fuel currently in the SFP and NFSR, as well as anticipated future fuel designs. The proposed change is analyzed for a maximum power level of 3725 megawatts thermal.

Each Region 1 rack will have storage locations designated as Region 1A and Region 1B. A proposed new Fuel Assembly Storage Schematic (TS Figure 3.9-1) specifies which storage locations are Region 1A and which are Region 1B.

The administrative process for verifying proper fuel assembly loading using the new Region 1 storage configuration and new Region 2 and 3 burnup curves will be the same as the process used for the present burnup curves and storage configuration. Also, the response to a fuel assembly misloading event remains the same. There are no changes regarding how fuel assemblies are handled and moved, or in the administrative means used to ensure that fuel assemblies are not dropped or misloaded.

The TS requirement for SFP soluble boron will increase from 800 ppm to 2600 ppm. The process of controlling and measuring boron concentration, and responding should the concentration be found to be below the requirement, will remain the same.

4.2 Fuel Storage Criticality Analysis - General

For each normal operational and postulated accident scenario (both in the NFSR and the SFP) the criticality safety evaluation conservatively assumes that each fuel assembly is of a design to maximize fuel reactivity. The limiting design is a bounding composite of current and historical fuel designs that is expected to remain bounding for future fuel designs. The analyses assume bounding conservative depletion conditions for depleted fuel and account for biases and uncertainties as described in the criticality safety evaluation.

4.3 New Fuel Storage Racks Criticality Analysis – Normal Storage and Accident Conditions

Results of the criticality safety evaluation show that the NFSR maintain $k_{\text{eff}} \leq 0.95$ under normal storage conditions, accounting for biases and uncertainties.

The following postulated accident conditions were analyzed:

- NFSR area fully flooded (with water)
- NFSR area with optimum moderation (fully flooded with foam).

The results also show that the NFSR maintain $k_{\text{eff}} \leq 0.95$ for the area fully flooded scenario, and maintain $k_{\text{eff}} \leq 0.98$ for the optimum moderation scenario.

4.4 Spent Fuel Pool Criticality Safety Evaluation - Normal Storage Conditions

Results of the SFP criticality safety evaluation show that the proposed SFP storage configuration will maintain $k_{\text{eff}} < 1.0$ with 0 ppm of soluble boron in the SFP water, and $k_{\text{eff}} \leq 0.95$ with the SFP filled with 600 ppm soluble boron for normal fuel assembly storage conditions (including biases and uncertainties). Each Region 1 rack has storage locations designated as Region 1A or Region 1B per the configuration specified in new Figure 3.9-1.

Fresh fuel with enrichment ≤ 5.0 wt% may be stored in Region 1B. In order to remain bounded by the criticality safety evaluation, fuel assemblies must meet one of the following requirements for Region 1A storage:

- Enrichment ≤ 4.75 wt% U-235, or
- Enrichment ≤ 5.0 wt% U-235 with fuel burnup ≥ 2.0 GWD/MTU, or
- Enrichment ≤ 5.0 wt% U-235 with at least 12 IFBA rods

Fuel assemblies may be stored in Region 2 if they are in the acceptable burnup domain of the initial enrichment/burnup curve in Figure 3.9-2, or contain a RCCA (no burnup restriction).

Fuel assemblies may be stored in Region 3 if they are in the acceptable burnup/decay time domain of the initial enrichment/burnup/decay time curve in Figure 3.9-3.

4.5 Spent Fuel Pool Criticality Safety Evaluation - Accident Conditions

The SFP criticality safety evaluation has analyzed the following postulated accident conditions:

- single fuel assembly misload in the spent fuel racks
- multiple fuel assembly misload in the spent fuel racks
- loss of SFP cooling including partial voiding
- dropped assembly into the racks with grid damage (optimum fuel pin pitch)
- misload of an assembly between fuel racks
- fuel handling error (two fresh fuel assemblies out of rack in close proximity)
- seismic event.

The bounding accident is the multiple fuel assembly misload in Region 2 spent fuel racks. This scenario assumes a 6 x 4 storage array of the limiting fresh fuel assembly (5.0 wt% U-235 enrichment with no burnable poison) surrounded by fresh fuel assemblies with 5.0 wt% U-235 enrichment and 32 IFBA rods in a 20x20 storage cell array.

The criticality safety evaluation shows that 2600 ppm of soluble boron maintains $k_{\text{eff}} \leq 0.95$ for this bounding scenario, accounting for biases and uncertainties.

The proposed change will increase the minimum SFP boron concentration to 2600 ppm.

4.6 Boron Dilution

The proposed change will increase the SFP minimum soluble boron concentration requirement from 800 ppm to 2600 ppm. No equipment that could contribute to or mitigate a boron dilution event will be changed as part of this proposed change. Thus, no new avenues for a boron dilution event will be created. There are no proposed changes regarding boron concentration maintenance or response to a boron dilution event.

DENC performed a SFP boron dilution analysis that assumes a dilution from 2300 ppm to 700 ppm soluble boron (summarized in Attachment 7). The systems that could dilute SFP boron, either by direct connection to the spent fuel pool or by a potential pipe crack/break, were analyzed via a bleed and feed methodology. The analysis demonstrates that sufficient time is available to detect and mitigate a boron dilution event prior to reaching a concentration of 700 ppm.

4.7 Storage of Non-fuel Components and Non-standard Fuel Assemblies

The criticality safety evaluation concludes that non-fuel components may be placed in any spent fuel rack storage location since these components are less reactive than fuel. Non-fuel components can also be placed in the guide tubes of any fuel assembly because the fuel lattice is under moderated.

The criticality safety evaluation evaluated each existing non-standard fuel assembly currently stored in the MPS3 SFP using the same methods used for standard fuel. The reactivity of non-standard fuel assemblies in the SFP is bounded by the limiting assembly design. The criticality safety evaluation provides a list of the non-standard fuel assemblies analyzed.

4.8 Spent Fuel Pool Storage – Other Items

Seismic Response:

The criticality safety evaluation analyzed the reactivity impact of a postulated seismic event under the proposed storage requirements, and found that the spent fuel racks maintain $k_{eff} \leq 0.95$. The evaluation considered fuel assembly and spent fuel rack motion during the event. The reactivity impact of the seismic event is bounded by the multiple fuel misload.

The MPS3 SFP is currently licensed to store 1860 fuel assemblies per MPS3 TS 5.6.3 (this includes the capacity of one Region 2 rack that has not been placed into the SFP). The proposed change will not change the number of fuel storage locations or fuel assemblies, or physically change any of the spent fuel racks. Thus, the SFP seismic/structural loading requirements for the proposed change are bounded by the existing TS.

Radiological and Thermal Impact:

DENC has considered the impact of the proposed change on the MPS3 licensing basis fuel handling accident dose consequence assessment and the SFP heat load. The proposed change does not alter the existing limits on enrichment, burnup limits, peaking factors, or gap fractions in the MPS3 core. Therefore, there is no impact on the radiological assessment or the SFP heat load.

Safety Analysis Limits:

Other than the proposed changes to the TS which specify SFP Region 1 storage configuration, the SFP soluble boron concentration, and the reactivity credit taken for assemblies containing RCCAs in Region 2, there are no changes to any TS LCO or operating or safety-related setpoints associated with the proposed change.

4.9 Implementation Considerations

DENC plans to fully implement the revised TS within 90 days after NRC approval of this proposed license amendment.

4.10 Conclusions

Implementation of the proposed license amendment is safe and will not negatively affect plant operation. The proposed change will make no modifications to plant equipment or how equipment is operated or maintained. In particular, there are no changes to how fuel is handled, including how fuel is moved, inserted into, and removed from SFP and NFSR storage locations. There are no changes to how RCCAs are handled, including how they are moved, inserted into, and removed from a fuel assembly. There are no changes to qualifying and verifying fuel storage in the SFP. There are no changes to the required response to a fuel misloading or drop event. Also, since the proposed license amendment does not modify plant equipment or its operation and maintenance, including equipment used to maintain SFP soluble boron levels, the proposed license amendment will not impact a boron dilution event or plant response to it.

SFP fuel storage requirements will continue to be maintained by administrative means to ensure compliance with the proposed Region 1 storage configuration and fresh fuel enrichment requirements, the proposed burnup curves for Regions 2 and 3, and the allowance to credit RCCAs contained in fuel assemblies for Region 2. The consequences of, or plant response to, a fuel misload event are not changed.

The criticality safety evaluation shows that the NFSR will maintain $k_{\text{eff}} \leq 0.95$ in the fully flooded condition and $k_{\text{eff}} \leq 0.98$ in the optimum moderation condition. Furthermore, the criticality safety evaluation shows that the SFP will maintain $k_{\text{eff}} \leq 0.95$ under normal and postulated accident conditions with credit for soluble boron. The SFP will also maintain $k_{\text{eff}} < 1.0$ with no soluble boron under normal conditions under the most reactive fuel storage configuration allowed by the proposed change.

5.0 Regulatory Evaluation

5.1 Applicable Regulatory Requirements and Criteria

Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), General Design Criterion (GDC) 62, "Prevention of criticality in fuel storage and handling," states that "criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." The NRC has established a 5% subcriticality margin (i.e., k_{eff}) less than or equal to 0.95) for nuclear power plant licensees to comply with GDC 62.

10 CFR 50.68 subpart (b), regarding New Fuel Storage Racks, specifies that "(2) The estimated ratio of neutron production to neutron absorption and leakage (k -effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k -effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Also, Subpart (b)(4) of 10 CFR 50.68, "Criticality accident requirements," specifies, "if credit is taken for soluble boron, the k -effective of the SFP storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k -effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

5.2 No Significant Hazards Consideration

DENC has performed the significant hazards consideration for the proposed license amendment by addressing the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: *No.*

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will not affect plant equipment or structure, including the SFP, NFSR, or fuel handling equipment, including how the equipment is operated and maintained. There are no changes to the equipment for fuel handling or how fuel assemblies are handled, including how fuel assemblies are inserted into and removed from SFP and NFSR storage locations. There is no change to administrative means to verify correct fuel assembly storage in the SFP, or the required response to a fuel assembly misload or drop event. There are no changes to how RCCAs will be handled, including how RCCAs are inserted into or removed from a fuel assembly or other location such as a SFP storage location. Also, since the proposed change does not modify plant equipment or its operation and maintenance, including equipment used to maintain SFP soluble boron levels, the proposed change will not impact a boron dilution event or plant response to it.

The criticality safety evaluation concluded that the NFSR limiting accident is the fully flooded condition with each storage location loaded with a maximum reactivity fuel assembly. The NFSR maintains $k_{eff} \leq 0.95$ for this postulated scenario including uncertainties and biases. The NFSR also maintains $k_{eff} \leq 0.98$ for the optimum moderation scenario including uncertainties and biases. Thus, the consequences of a previously evaluated NFSR related accident is not significantly increased. There is no change to the plant equipment or its operation and maintenance due to the proposed change. Thus, the probability of a flooding accident that could impact the NFSR is not significantly increased.

Regarding the SFP, the Region 1 storage configuration will change. The Region 2 and 3 burnup curves will be updated and reduced in number. The process of choosing fuel assembly storage locations will not change, except that the Region 1 storage configuration and Region 2 and 3 burnup requirements will be updated, and fuel assemblies containing RCCAs may be stored in Region 2 without consideration of the burnup curve. The physical handling, insertion, removal, and storage of fuel assemblies in SFP racks will not change. The MPS3 program for choosing fuel assembly storage locations, for fuel handling, and for assuring that the fuel assemblies are placed into correct locations will remain in place. Thus, the probability of a fuel assembly misloading or a fuel assembly drop in the SFP will not significantly increase due to the proposed change.

Several postulated accidents for the SFP were reviewed for the proposed change which included postulated fuel assembly misload and drop scenarios. The criticality safety evaluation for the SFP concluded that the limiting accident, which bounds the other scenarios, is a multiple misload of fuel assemblies into each Region 2 fuel storage location. The criticality safety evaluation concluded that a SFP soluble boron concentration of 2600 ppm will maintain $k_{\text{eff}} \leq 0.95$, including uncertainties and biases, for this postulated scenario. The minimum TS soluble boron concentration will be increased from 800 ppm to 2600 ppm. MPS3 has maintained SFP soluble boron concentration greater than 2600 ppm for many years, so the proposed change will not affect the routine maintaining of the boron concentration.

There are no changes to plant equipment, including its operation and maintenance, as a result of the proposed change, including equipment associated with maintaining SFP soluble boron concentration or possible flow paths that could contribute to a boron dilution event. Thus, no new avenues for a boron dilution event will be created. There is no change regarding how the plant maintains boron concentration or responds to a boron dilution event. The criticality safety evaluation for the postulated boron dilution event shows the SFP maintains $k_{\text{eff}} \leq 0.95$ at 600 ppm soluble boron. Thus, there is no significant increase in the probability or consequences of a boron dilution accident.

The MPS3 SFP is currently licensed to store 1860 fuel assemblies which include a Region 2 rack that has not been placed in the SFP (TS 5.6.3). Thus, the SFP seismic/structural loading requirements for the proposed change are bounded by the existing TS. The criticality safety evaluation shows that k_{eff} will be maintained ≤ 0.95 during a postulated seismic event. Thus, there is no increase in the consequences of a seismic event.

In each of the above scenarios the proposed change does not significantly increase the probability of an accident previously evaluated, and maintains required k_{eff} margin. Therefore, it is concluded that the probability or consequences of a previously evaluated accident do not significantly increase.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: *No.*

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

There is no change to any plant equipment, including how equipment is operated and maintained. There will be no changes to equipment used to handle fuel assemblies (or any heavy load) over the NFSR or the SFP. There is no change regarding how the fuel assemblies are stored, inserted

into, and removed from fuel storage locations. There is no change to how RCCAs will be inserted into or removed from a fuel assembly or other location, or otherwise how RCCAs are handled. Thus, there are no new accidents created over and above the existing postulated accidents of a fuel misload or a fuel assembly drop in the SFP or a flooding event in the NFSR area.

Also, since there is no change to the plant equipment or how equipment is operated and maintained, the probability of a new type of accident that could impact the SFP or NFSR is not significantly increased.

The criticality safety evaluation for the first time at MPS3 specifically analyzes a boron dilution event. However, the overall accident analyzed is the potential for a SFP criticality, and the boron dilution event is another potential initiator of the postulated SFP criticality accident. Also, the possibility of a SFP boron dilution event has always existed at MPS3 and the proposed change does not newly create or change the possibility of such an event occurring.

The criticality safety evaluation for the first time at MPS3 specifically analyzes a multiple fuel misload event. As with the postulated boron dilution event, the possibility of a multiple fuel assembly misload has always existed at MPS3 and the proposed change does not newly create or change the possibility of such an event occurring. Also, this postulated event was analyzed for the MPS2 spent fuel pool criticality LAR which the NRC approved in June 2016 (Reference 3).

Since the proposed change will not change fuel/RCCA handling equipment or how fuel assemblies and RCCAs are handled and stored, nor will it change any other plant equipment, there is no mechanism for creating a new or different kind of accident not previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: *No.*

The proposed change does not involve a significant reduction in a margin of safety.

The licensing requirement for the SFP is that k_{eff} remain ≤ 0.95 under normal and postulated accident conditions with credit for soluble boron. The criticality safety evaluation concluded that this requirement is met for the bounding postulated accident of a multiple misload of fuel assemblies into each Region 2 fuel storage location. The analyses apply to all of the fuel assemblies currently stored in the MPS3 SFP and to future anticipated fuel designs.

In addition, the criticality safety evaluation concludes that the SFP will maintain $k_{eff} < 1.0$ with 0 ppm soluble boron in the SFP under normal conditions with the maximum allowed reactivity fuel assembly stored in each fuel storage location.

The criticality safety evaluation also allows the following storage configurations. In each case the storage configuration does not increase reactivity assuring that k_{eff} margin is maintained:

- Storing non-fuel components in any spent fuel rack storage location where fuel assemblies are allowed
- Storing non-fuel components in the guide tubes of any fuel assembly.

The criticality safety evaluation evaluated non-standard fuel stored in the MPS3 to determine where they can be stored in the SFP. This information is used to maintain k_{eff} margin when storing non-standard fuel assemblies.

The licensing requirement for the NFSR is that k_{eff} remain ≤ 0.95 for the fully flooded scenario, and ≤ 0.98 for the optimum moderation scenario. The criticality safety evaluation concludes that these requirements are met assuming each fuel storage location is loaded with a maximum reactivity fuel assembly (5.0 wt% U-235 enrichment with no burnable poisons).

Therefore, all the margins of safety are maintained, and the proposed change does not involve a significant reduction in a margin of safety

Based on the above information, DENC concludes that the proposed license amendment involves no significant hazards consideration under the criteria set forth in 10 CFR 50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

5.3 Precedents

The proposed changes to the MPS3 technical specifications are similar in fundamental aspects to those referenced in this NRC Safety Evaluation Report:

1. NRC License Amendment and associated SER under cover letter "Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Technical Specification Changes for Spent Fuel Storage (TAC NO. MF0435)," from R. V. Guzman (NRC) to D. A. Heacock (Dominion Nuclear Connecticut, Inc.), June 23, 2016.

The current MPS3 application includes unique aspects that reflect DENC's understanding of NRC staff expectations for the content and supporting analyses of spent fuel criticality submittals. The enclosed submittal content was developed based on insights and discussion between DENC and NRC staff which occurred during a pre-submittal meeting:

1. Summary of April 26, 2016, Pre-Application Teleconference with Virginia Electric and Power Company for Increase in Maximum Fuel Enrichment for New Fuel Storage Racks and Spent Fuel Pool (CAC Nos. MF7432 and MF7433).

5.4 Conclusion

Based on the considerations discussed above, there is reasonable assurance that (1) the health and safety of the public will not be endangered by the proposed changes, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the requested license amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Considerations

DENC has reviewed the proposed license amendment for environmental considerations. The proposed license amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

1. NRC Amendment and associated SER under cover letter "Millstone Nuclear Power Station, Unit No. 3 – Issuance of Amendment Re: Increasing Spent Fuel Storage Capacity (TAC No. MA5137)," letter from V. Nerses (NRC) to R. G. Lizotte (Northeast Nuclear Energy Co.), November 28, 2000.
2. NRC Amendment and associated SER under cover letter "Millstone Power Station, Unit No.3 - Issuance of Amendment Re: Spent Fuel Pool Criticality (TAC NO. MD8251)," letter from C. J. Sanders (NRC) to D. A. Heacock (Dominion Nuclear Connecticut, Inc.), March 26, 2010.
3. NRC Amendment and associated SER under cover letter "Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Technical Specification Changes for Spent Fuel Storage (TAC NO. MF0435)," from R. V. Guzman (NRC) to D. A. Heacock (Dominion Nuclear Connecticut, Inc.), June 23, 2016.

Attachment 2

MARKED-UP TECHNICAL SPECIFICATIONS PAGES

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

DEFINITIONS

VENTING

1.40 Deleted

1.41 Deleted

1.39 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 AND 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

- (a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.
- (b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

1.43 Deleted

1.44 Deleted

REFUELING OPERATIONSBORON CONCENTRATIONLIMITING CONDITION FOR OPERATION

- 3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be greater than or equal to ~~800~~ ppm.

2600

APPLICABILITY:

Whenever fuel assemblies are in the spent fuel pool.

ACTION:

- a. With the boron concentration less than ~~800~~ ppm, initiate action to bring the boron concentration in the fuel pool to at least 800 ppm within 72 hours, and
- b. With the boron concentration less than ~~800~~ ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

- 4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to ~~800~~ ppm at the frequency specified in the Surveillance Frequency Control Program.

REFUELING OPERATIONS

STORAGE

3/4.9.13 SPENT FUEL POOL - REACTIVITYInsert # 1 on
next pageLIMITING CONDITION FOR OPERATION

3.9.13 The Reactivity Condition of the Spent Fuel Pool shall be such that k_{eff} is less than or equal to 0.95 at all times.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

- a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and
- b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3, 3.9-4, or 3.9-5 to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-3 by checking the fuel assembly's design, decay time, and burn-up documentation.

4.9.13.1.3. Ensure that all fuel assemblies used exclusively in pre-uprate (3411 Mwt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel assembly's design, decay time, and burn-up documentation. Ensure that all fuel assemblies used in post-uprate (3650 Mwt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burn-up limits of Figure 3.9-5 by checking the fuel assembly's design, decay time, and burn-up documentation.

Insert # 1 to TS 3.9.13 Spent Fuel Pool – Storage Limiting Condition for Operation

3.9.13 The spent fuel storage requirements necessary to maintain k_{eff} within limits shall be met.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION:

- a. For a fuel assembly stored in Region 1A - initiate immediate action to move any assembly which does not meet Surveillance Requirement 4.9.13.1.1. to Region 1B.
- b. For a fuel assembly stored in Region 2 that does not contain a Rod Cluster Control Assembly - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-2 to a location for which that fuel assembly is allowed.
- c. For a fuel assembly stored in Region 3 - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-3 to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

-----NOTE-----

The Region 1 Fuel Storage Loading Schematic (Figure 3.9-1) designates each storage location as either Region 1A or Region 1B.

Regarding fuel assemblies that contain a Rod Cluster Control Assembly for storage in Region 2 - if the enrichment and burnup of a given assembly is not in the "Acceptable" domain of Figure 3.9-2 (e.g., the assembly requires a Rod Cluster Control Assembly to be stored in Region 2), then the assembly must be located in an acceptable Region 1 storage location before its Rod Cluster Control Assembly can be inserted or removed.

Initial enrichment is the maximum initial planar volume averaged as-built U-235 enrichment in the assembly. If the assembly has axial blankets the lower enriched fuel is not credited in determining the enrichment. Also, fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates.

Insert # 1 to TS 3/4.9.13 Spent Fuel Pool – Storage Limiting Condition for Operation and Surveillance Requirements (continued)

- 4.9.13.1.1. Ensure that all fuel assemblies to be placed into a Region 1A storage location, with an initial enrichment greater than 4.75 w/o U-235, have achieved a fuel burnup greater than or equal to 2.0 GWD/MTU or contain a minimum of twelve (12) Integral Fuel Burnable Absorber (IFBA) Rods by checking the fuel assembly's location, design, and burnup documentation. Fuel assemblies with an initial enrichment less than or equal to 4.75 w/o U-235 may be stored in Region 1A without restriction.
- 4.9.13.1.2. Ensure that all fuel assemblies to be placed into Region 1B are stored consistent with the Fuel Storage Loading Schematic specified in Figure 3.9-1 by checking the fuel assembly's storage location. All fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 may be stored in Region 1B without restriction.
- 4.9.13.1.3. Ensure that all fuel assemblies to be stored in Region 2 - that do not contain a Rod Cluster Control Assembly - are within the enrichment and burnup limits of Figure 3.9-2 by checking the fuel assembly's design and burnup documentation. A fuel assembly that contains a Rod Cluster Control Assembly may be stored in Region 2 without restriction.
- 4.9.13.1.4. Ensure that all fuel assemblies to be stored in Region 3 are within the enrichment, burnup, and decay time limits of Figure 3.9-3 by checking the fuel assembly's design, burnup, and decay time documentation.

REFUELING OPERATIONS

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

~~3.9.14 Each STORAGE PATTERN of the Region 1 spent fuel pool racks shall require that:~~

- ~~a. Prior to storing fuel assemblies in the STORAGE PATTERN per Figure 3.9-2, the cell blocking device for the cell location must be installed.~~
- ~~b. Prior to removal of a cell blocking device from the cell location per Figure 3.9-2, the STORAGE PATTERN must be vacant of all stored fuel assemblies~~

~~APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.~~

~~ACTION: Take immediate action to comply with 3.9.14(a), (b).~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.14 Verify that 3.9.14 is satisfied with no fuel assemblies stored in the STORAGE PATTERN prior to installing and removing a cell blocking device in the spent fuel racks.~~

TS 3.9.14 is DELETED

Figure 3.9-1 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment
for Region 1 4-OUT-OF-4 Fuel Storage Configuration

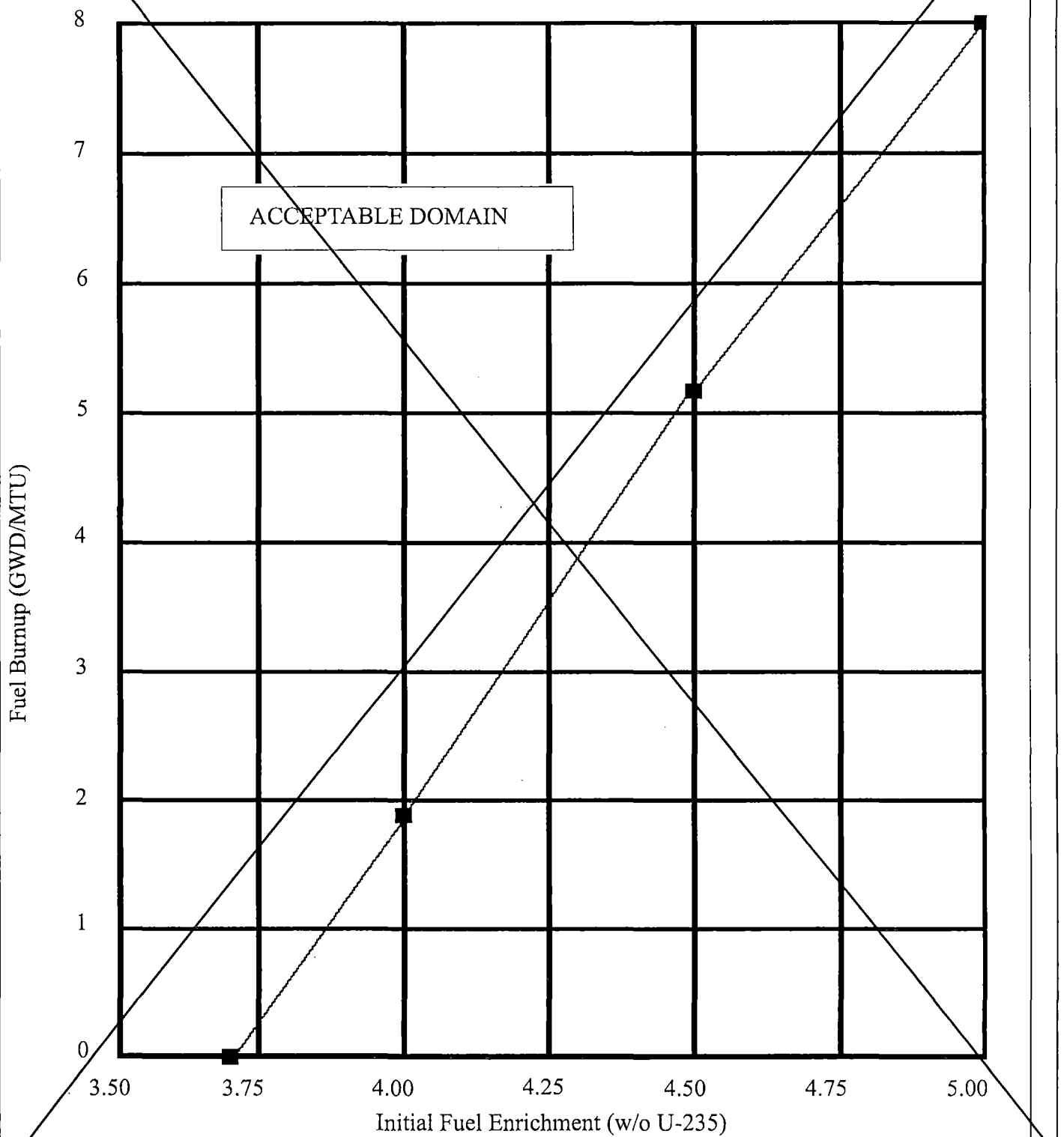


Figure 3.9-1 Region 1 Fuel Storage Loading Schematic

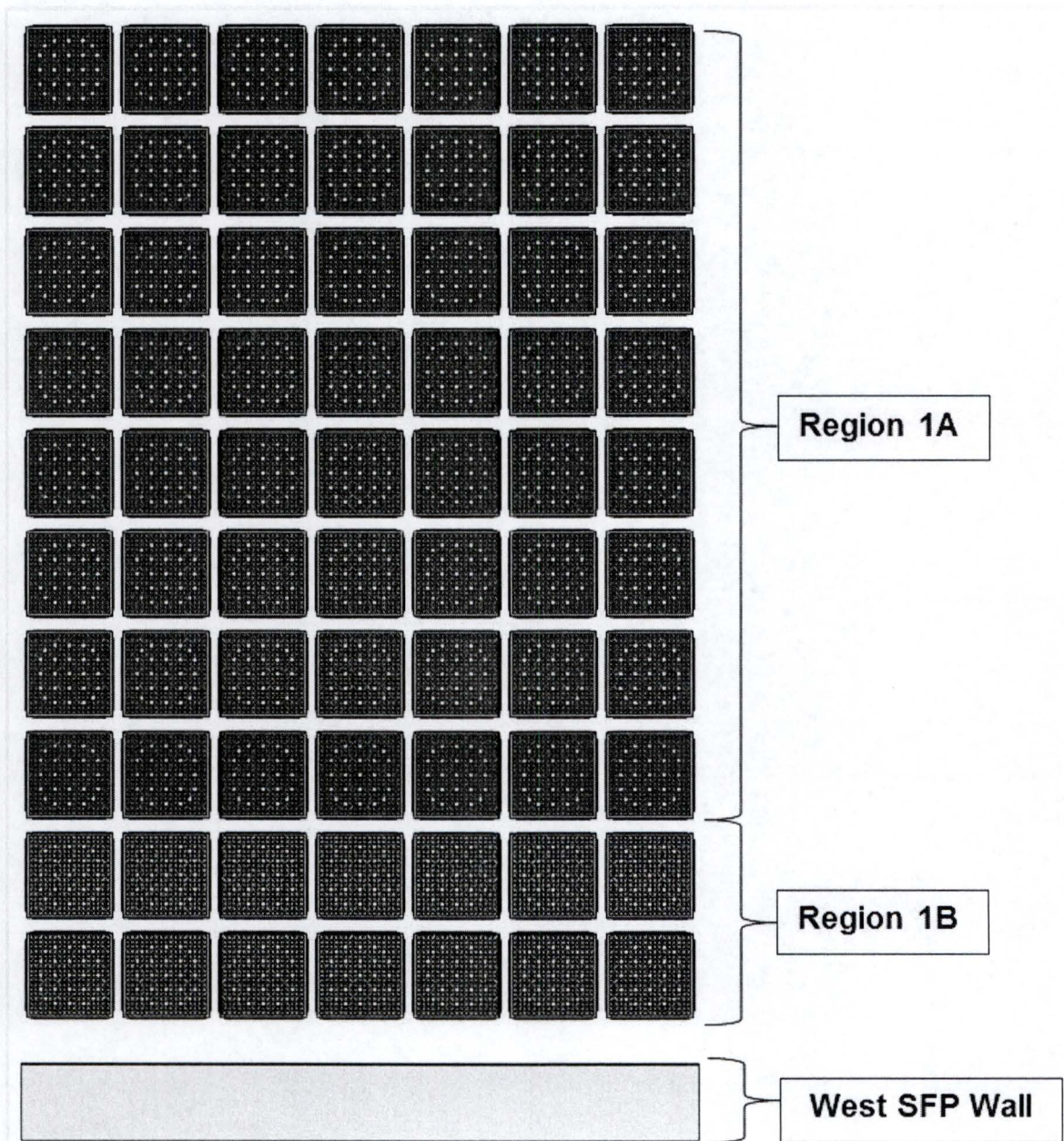


Figure 3.9-2 Region 1 3-OUT-OF-4 Storage Fuel Assembly Loading Schematic

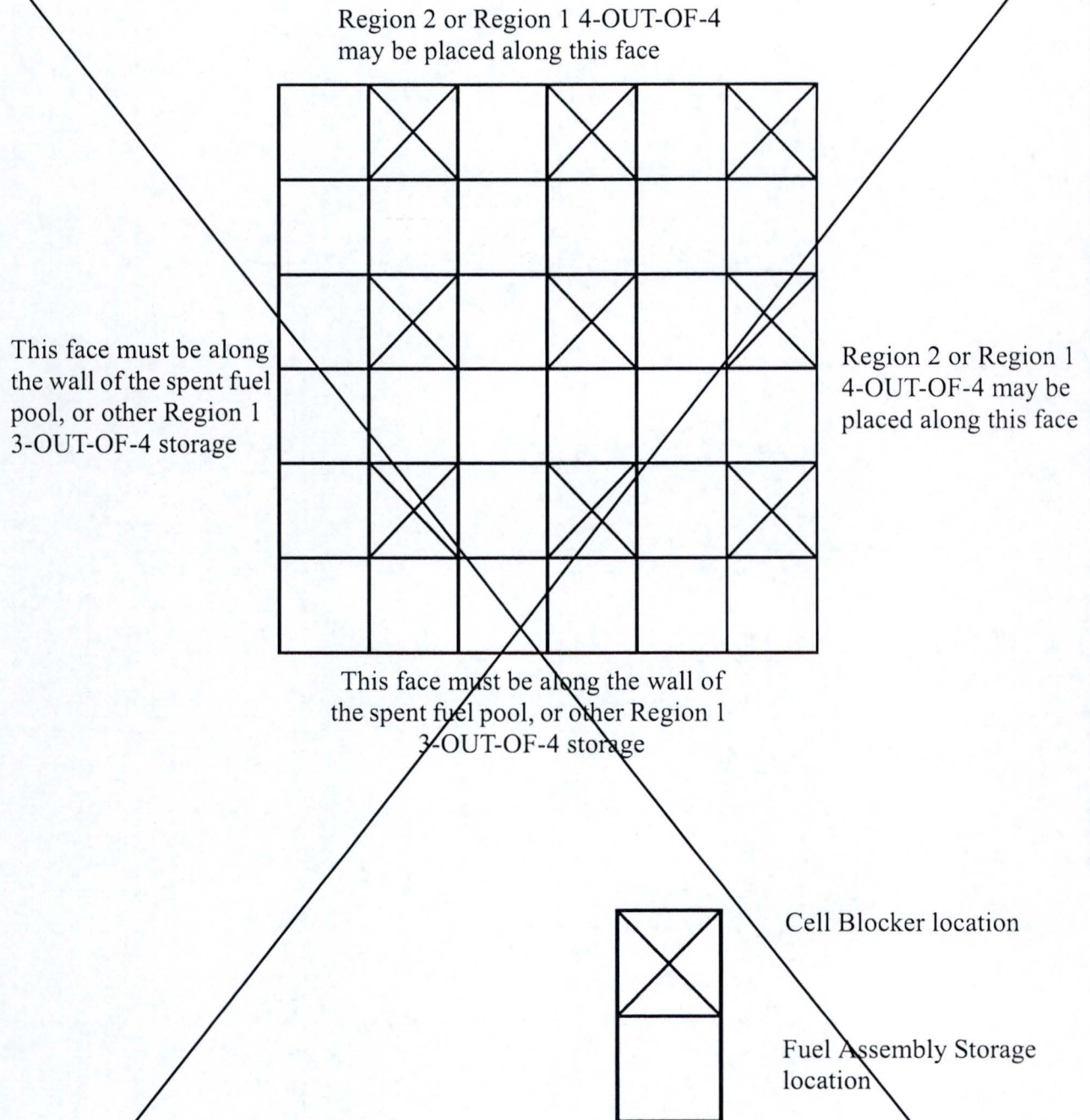


FIGURE 3.9-2 Minimum Fuel Assembly Burnup versus Initial Enrichment for Region 2
Storage Configuration
(Fuel Assemblies without Rod Cluster Control Assemblies)

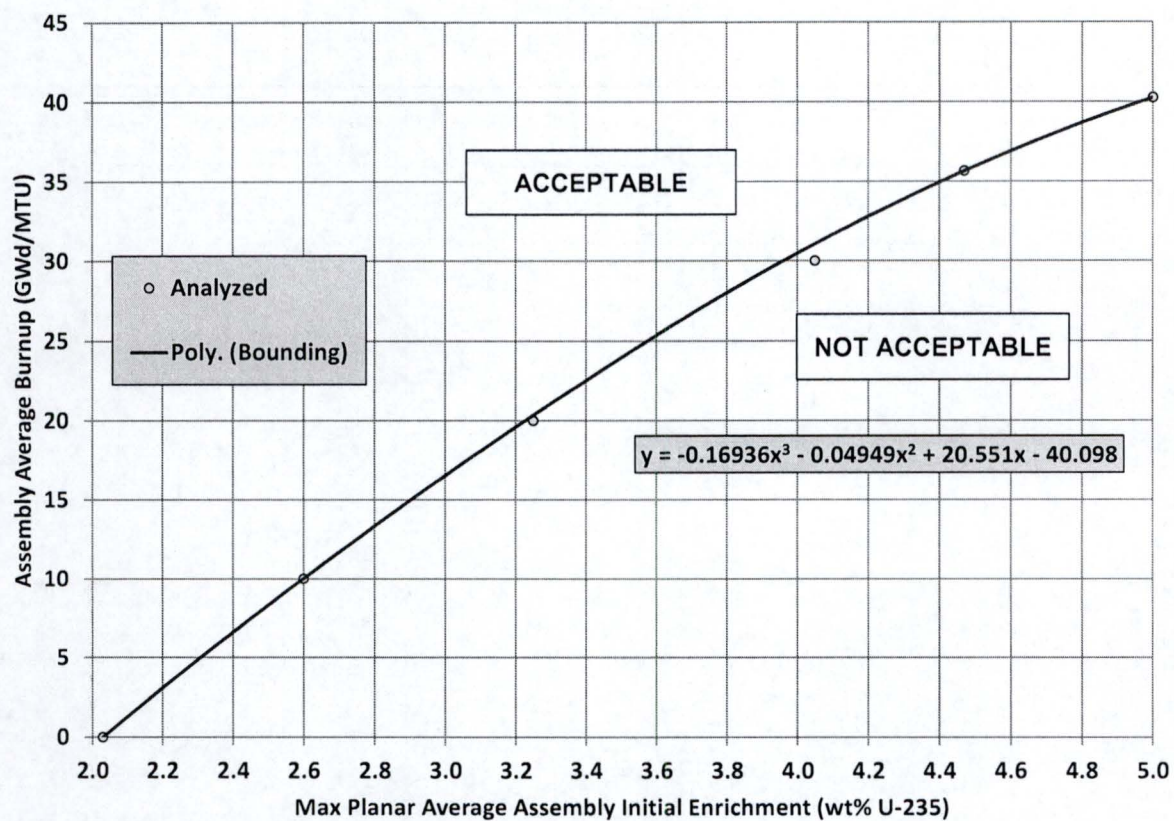
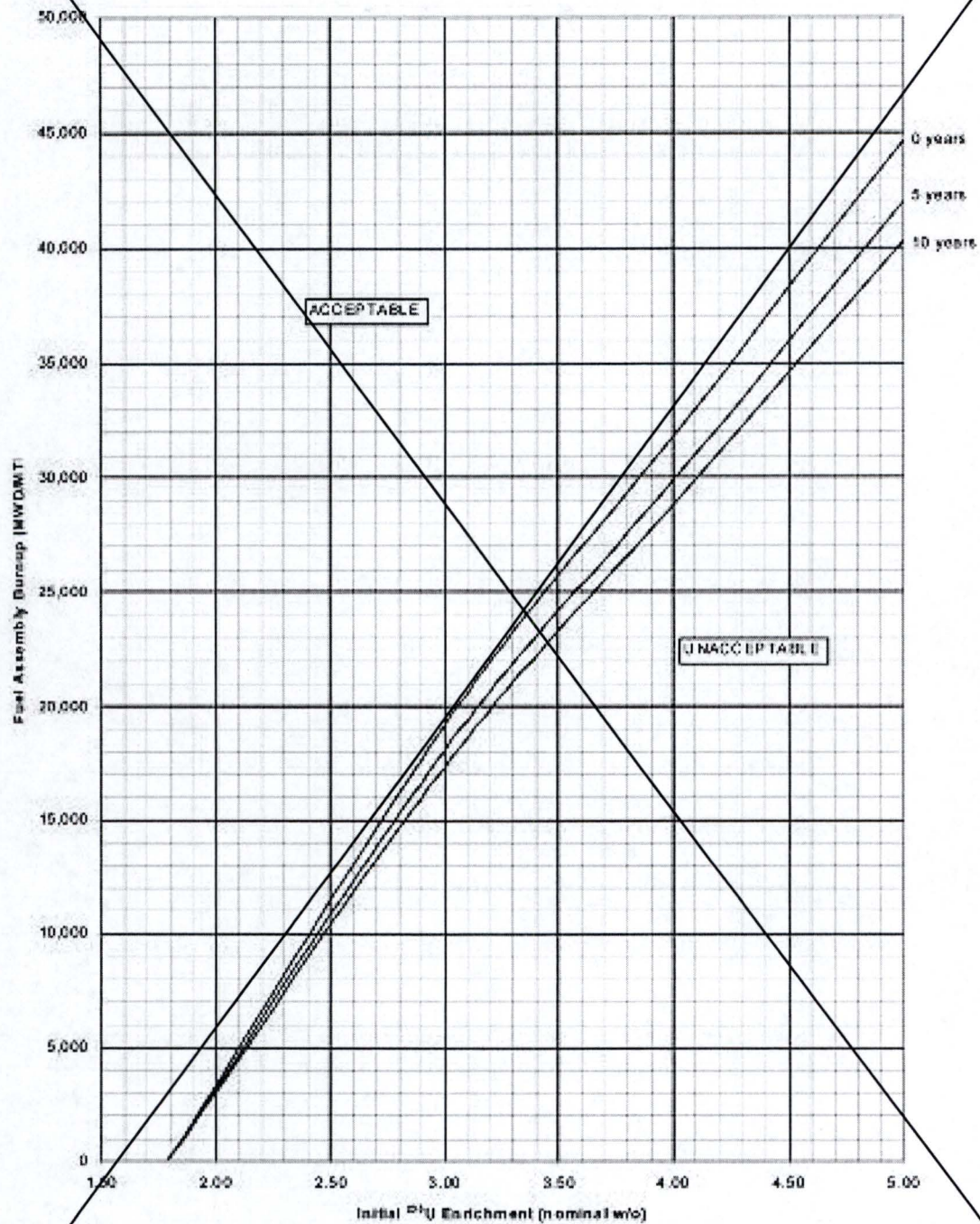
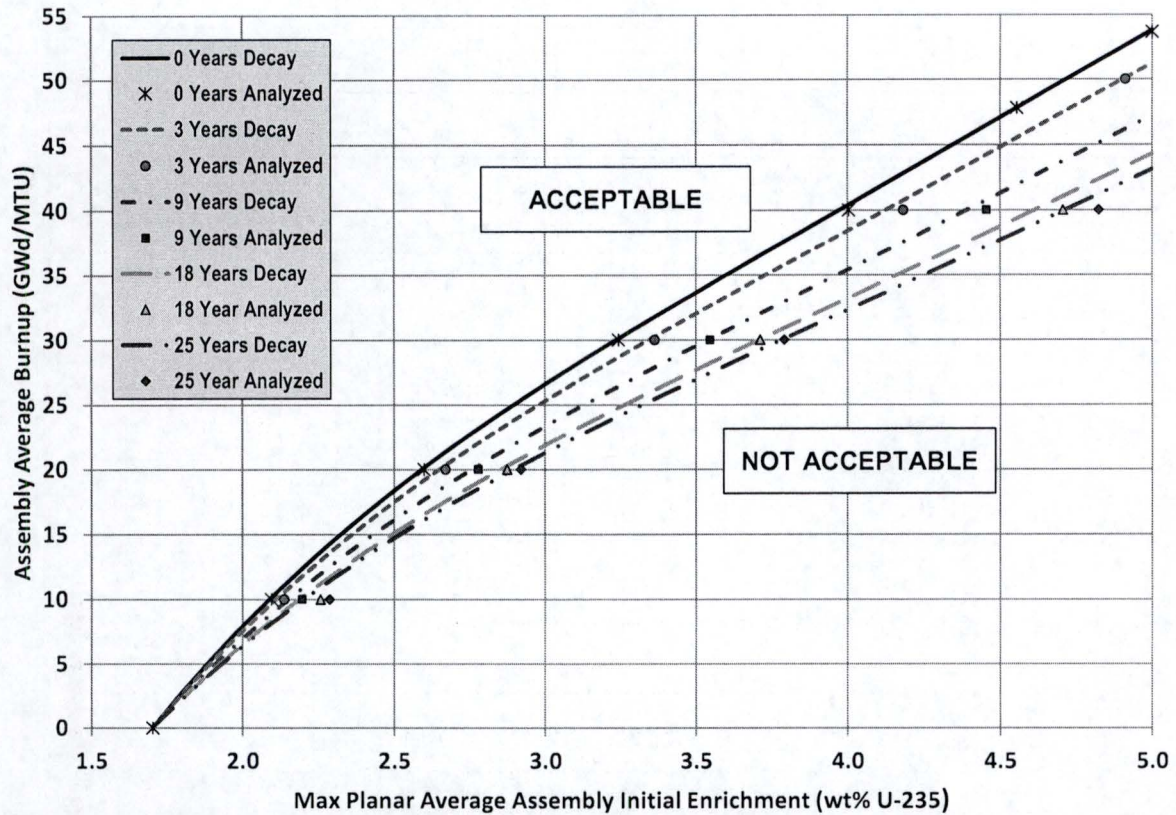


Figure 3.9-3 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 2 Storage Configuration



NOTE: For assemblies from Post-Uprate (3650 Mwt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel. Fuel batches A, B, C, and D may not be stored in Region 2.

FIGURE 3.9-3 Minimum Fuel Assembly Burnup and Decay Time versus Initial Enrichment for Region 3 Storage Configuration



The burnup curve equations have the following polynomial format (bounding):

$$BU [GWD/MTU] = a_4 * wt\%^4 + a_3 * wt\%^3 + a_2 * wt\%^2 + a_1 * wt\%^1 + a_0$$

Burnup Credit Curve Polynomial Coefficients

Region	Decay Time (Years)	a_4	a_3	a_2	a_1	a_0
3	No Credit	-0.2459	4.208	-26.80	88.70	-92.00
3	3 Years	-0.2338	4.001	-25.48	84.34	-87.47
3	9 Years	-0.2153	3.684	-23.46	77.66	-80.54
3	18 Years	-0.2020	3.458	-22.02	72.88	-75.59
3	25 Years	-0.1964	3.361	-21.40	70.84	-73.47

Figure 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Pre-Uprate (3411 Mwt) Cores

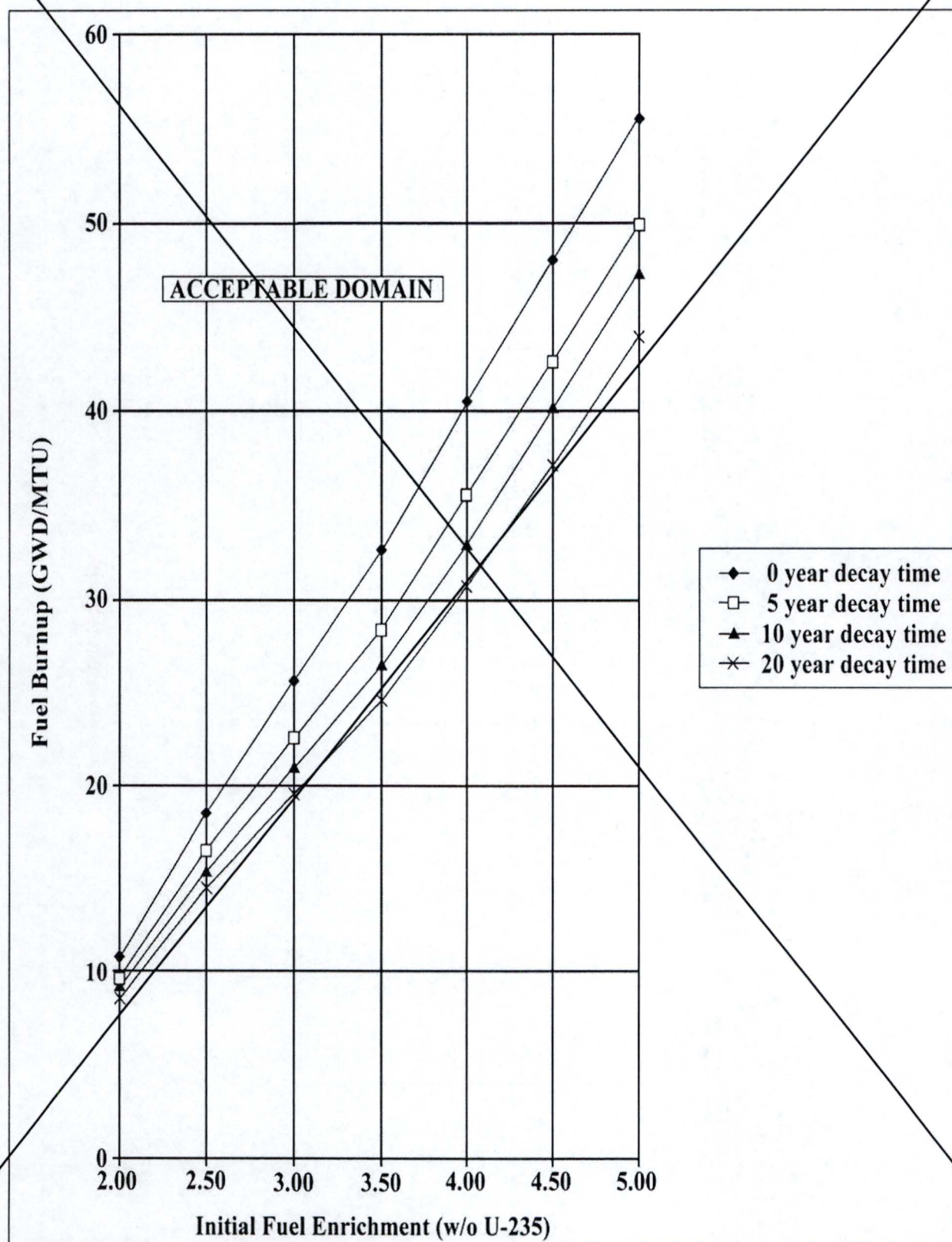
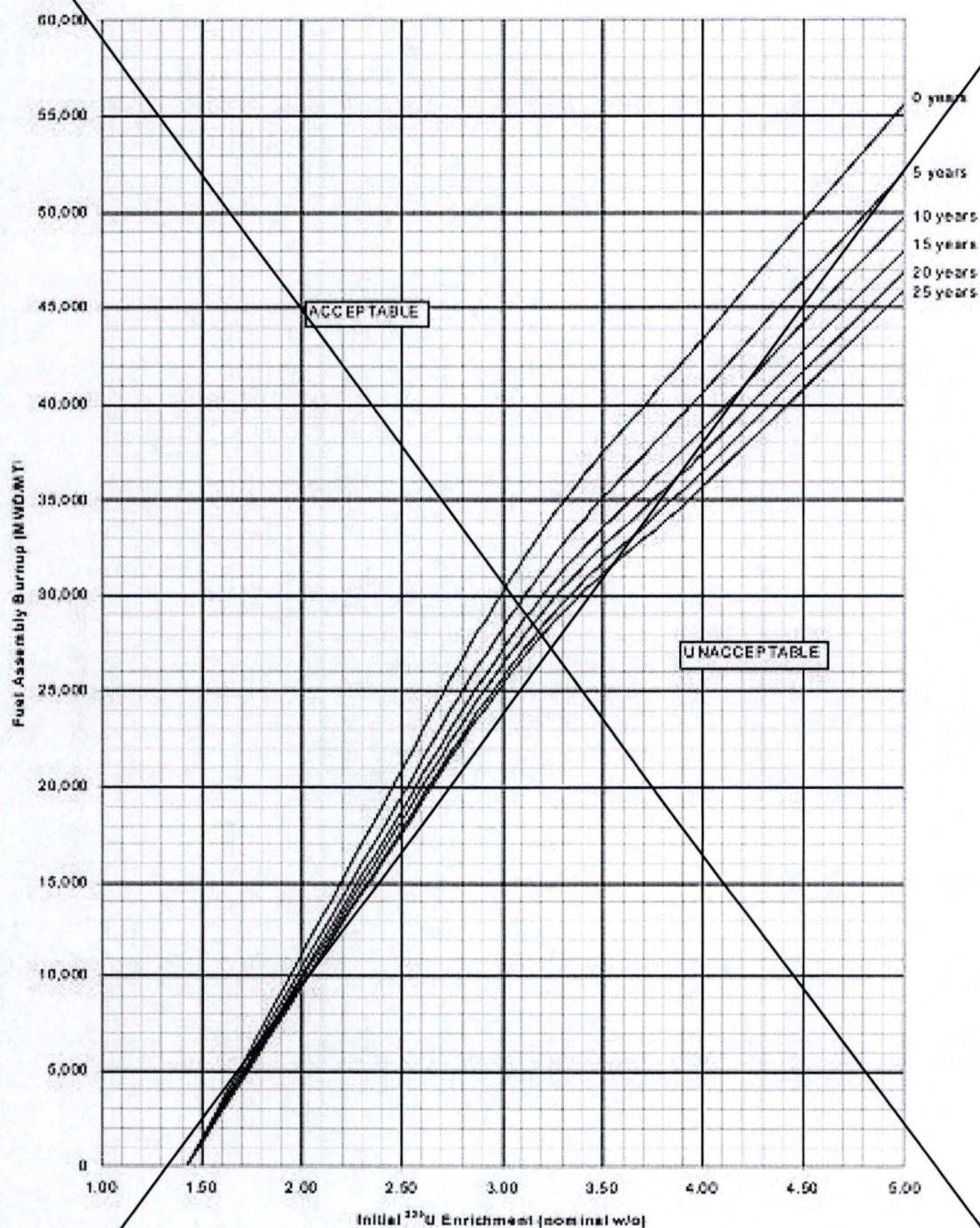


Figure 3.9-5 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 Mwt) Cores



NOTE: For assemblies from Post-Uprate (3650 Mwt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel.

DESIGN FEATURES5.6 FUEL STORAGE

Insert # 2 on next page

CRITICALITY

- 5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The storage rack Regions are:
- a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:
 - (1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.
 - (2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.
 - b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup and fuel decay time as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.
 - c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4 for assemblies used exclusively in pre-uprate (3411 Mwt) cores or Figure 3.9-5 for assemblies used in post-uprate (3650 Mwt) cores, fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

Insert # 2 to TS 5.6.1.1 Fuel Storage – Criticality

- 5.6.1.1 The New Fuel Storage Racks, a nominal 22.125 inch center to center distance, credit a fixed neutron absorber (BORAL) within the rack and are designed and shall be maintained with:
- K_{eff} less than or equal to 0.95 with the storage racks fully loaded with the highest reactivity fuel and flooded with potential moderators,
 - K_{eff} less than or equal to 0.98 with the storage racks fully loaded with the highest reactivity fuel and optimum moderation of the racks.

The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than 1.0 when flooded with unborated water, and K_{eff} less than or equal to 0.95 with 600 ppm soluble boron in the spent fuel pool water. The storage rack regions are as follows:

- Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack. Each Region 1 fuel storage rack contains two storage sub-regions - Region 1A and Region 1B:
 - (1) Region 1A – Fuel assemblies meeting one of the following three criteria may be stored in Region 1A storage locations:
 - initial enrichment less than or equal to 4.75 w/o U-235, or
 - initial enrichment less than or equal to 5.0 w/o U-235 with a fuel burnup greater than or equal to 2.0 GWD/MTU, or
 - initial enrichment less than or equal to 5.0 w/o U-235 that contain a minimum of 12 Integral Fuel Burnable Absorber (IFBA) rods.
 - (2) Region 1B – Fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 shall be stored per the Fuel Storage Loading Schematic shown in Figure 3.9-1 (the two rows against the spent fuel pool west wall are designated Region 1B).
- Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack and either fuel burnup as shown in Figure 3.9-2 or takes credit for containing a Rod Cluster Control Assembly.
- Region 3, a nominal 10.35 inch center to center distance, credits fuel burnup and decay time as shown in Figure 3.9-3. These racks contain Boraflex which is not credited.

Attachment 3

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES

(For Information Only)

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

March 9, 2010

3/4.9 REFUELING OPERATIONS

and the use of Rod Cluster Control
Assemblies for certain assemblies in
Region 2.

BASES3/4.9.1.2 BORON CONCENTRATION IN SPENT FUEL POOL

During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than ~~or equal to 0.95~~ ^{1.0} in an unborated water environment. This is accomplished in Region 1, 2, and 3 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in some fuel storage regions, the limits on fuel burnup, fuel enrichment, and minimum fuel decay time, ~~and the use of blocking devices in certain fuel storage locations.~~

The boron requirement in the spent fuel pool specified in 3.9.1.2 ensures that in the event of a fuel assembly ~~handling accident involving either a single dropped or misplaced fuel assembly,~~ the K_{eff} of the spent fuel storage racks will remain less than or equal to 0.95.

3/4.9.2 INSTRUMENTATION

~~misload accident, which involves the misloading of multiple fuel assemblies,~~

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range-neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

There are two sets of source range neutron flux monitors:

- (1) Westinghouse source range neutron flux monitors, and
- (2) Gamma-Metrics source range neutron flux monitors.

The Westinghouse monitors are the normal source range monitors used during refueling activities. Gamma-Metrics source range neutron flux monitors are an acceptable equivalent control room indication for the Westinghouse source range neutron flux Monitors in MODE 6, including CORE Alterations, as follows:

with the core in place within the reactor vessel or,

with the Gamma Metrics source range neutron flux monitor(s) coupled to the core.
Reactor Engineering shall determine whether each monitor is coupled to the core.

This limiting condition for operation requires two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication in the control room. In addition, at least one of the two monitors must provide an OPERABLE audible count rate function in the control room and containment.

July 12, 2007

REFUELING OPERATIONSBASES3/4.9.13 SPENT FUEL POOL - REACTIVITY

← STORAGE

Insert # B1 on next page

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 3-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-3.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4 for assemblies used exclusively in the pre-uprate (3411 Mwt) cores and Figure 3.9-5 for assemblies used in the post-uprate (3650 Mwt) cores. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3, 3.9-4, and 3.9-5 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3, 3.9-4, and 3.9-5 as well as restrictions specified in the Note on Figures 3.9-3 and 3.9-5 are complied with.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN ← DELETED

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region 1 3-OUT-OF-4 storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 Boundary to other Storage Regions and minimize the number of boundaries where a fuel misplacement incident can occur.

Insert # B1 to TSB 3/4.9.13 Spent Fuel Pool – Storage

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than 1.0 in an unborated water environment, and less than or equal to 0.95 with 600 ppm soluble boron in the spent fuel pool water.

Maintaining K_{eff} less than or equal to 0.95 is accomplished in Region 1A storage rack locations by the combination of geometry of the rack spacing, the use of fixed neutron absorbers (BORAL) in the racks, and prohibiting storage of fuel assemblies with an enrichment greater than 4.75 w/o U-235 unless its fuel burnup is greater than or equal to 2.0 GWD/MTU or it contains twelve (12) or more IFBA rods.

Maintaining K_{eff} less than or equal to 0.95 is accomplished in Region 1B storage rack locations by the combination of geometry of the rack spacing, the use of fixed neutron absorbers (BORAL) in the racks, a maximum 5.0 weight percent initial fuel enrichment, and specifying which storage locations are designated as Region 1B.

Maintaining K_{eff} less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers (BORAL) in the racks, and the limits on initial fuel enrichment/fuel burnup specified in Figure 3.9-2. As an alternative, maintaining K_{eff} less than or equal to 0.95 can also be accomplished in Region 2 storage rack locations if the assembly has a maximum initial enrichment less than or equal to 5.0 weight percent and contains a Rod Cluster Control Assembly.

Maintaining K_{eff} less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing and the limits on initial fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-3. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3, the burnup/IFBA requirement in Region 1A, and the use of Rod Cluster Control Assemblies in Region 2 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool is conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the initial fuel enrichment, fuel burnup, and fuel decay times specified in Figures 3.9-1, 3.9-2, 3.9-3, the burnup/IFBA requirement in Region 1A, and the presence of a Rod Cluster Control Assembly (Region 2) are complied with.

Initial enrichment, when used to compare to fuel storage requirements, is the maximum initial planar volume averaged as-built U-235 enrichment in the assembly. If the assembly has axial blankets the lower enriched fuel is not credited in determining the enrichment. Fuel burnup when used to compare to fuel storage requirements is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

ATTACHMENT 4

CRITICALITY ANALYSIS CHECKLIST

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
1.0 Introduction		
Purpose of submittal	YES	Identify and add margin, MUR uprate, boron credit, control rod credit, SFP wall credit, update methods and codes, provide more core design flexibility.
License changes requested	YES	Burnup curves, storage pattern, minimum soluble boron, remove axial blanket restrictions, remove cell blocker requirement.
Summary of physical changes	YES	Remove cell blockers.
Summary of analytical scope	YES	SFP and new fuel storage criticality safety analysis including normal storage and fuel handling, abnormal conditions, and boron dilution analysis.
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	YES	
Requirements documents referenced	YES	Multiple.
Guidance documents referenced	YES	DSS-ISG-2010-01, NEI 12-16 draft
Acceptance criteria described	YES	
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	YES	Bounds historical and anticipated.
Describe all fuel in pool	YES	
Geometric dimensions (Nominal and Tolerances)	YES	
Schematic of guide tube patterns	YES	Only one pattern.
Material compositions	YES	
Describe future fuel to be covered	NO	None proposed. Composite bounding design used.
Geometric dimensions (Nominal and Tolerances)	NO	None proposed.
Schematic of guide tube patterns	NO	None proposed.
Material compositions	NO	None proposed.
Describe all fuel inserts	YES	
Geometric dimensions (Nominal and Tolerances)	YES	BPRA, WABA, RCCA, source rods, in-core thimble (tolerances for in-SFP components).
Schematic (axial/cross-section)	NO	Commonly used inserts.
Material compositions	YES	
Describe non-standard fuel	YES	

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
Geometric dimensions	YES	Fuel rod storage canister and fuel reconstitution considered.
Describe non-fuel items in fuel cells	NO	Non-fuel items can only be stored in cells that are allowed to store fuel assemblies. Therefore, non-fuel items displace fuel.
Nominal and tolerance dimensions	NO	
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & Storage rack description	YES	
Nominal and tolerance dimensions	YES	
Schematic (axial/cross-section)	YES	Rack diagrams
Material compositions	YES	
Spent fuel pool, Storage rack description	YES	Region 1: BORAL, flux trap Region 2: BORAL, non-flux trap Region 3: Boraflex (not credited), flux trap
Nominal and tolerance dimensions	YES	
Schematic (axial/cross-section)	YES	Rack diagrams
Material compositions	YES	
Other Reactivity Control Devices (Inserts)	NO	No installed rack inserts (optional RCCA credit).
Nominal and tolerance dimensions	NO	
Schematic (axial/cross-section)	NO	
Material compositions	NO	
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES	BORAL installed in New Fuel Storage racks.
Storage geometries	NO	All cells qualified for new fuel storage.
Bounding assembly design(s)	YES	
Integral absorber credit	NO	
Accident analysis	YES	
Spent fuel storage rack analysis description	YES	
Storage geometries	YES	Region 1A has all-cell storage. Region 1B restricted to two rows adjacent to SFP wall (credits leakage). Regions 2 and 3 have all-cell fuel storage.
Bounding assembly design(s)	YES	Composite bounding assembly.
Soluble boron credit	YES	
Boron dilution analysis	YES	

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
Burnup credit	YES	
Decay time credit	YES	Region 3.
Integral absorber credit	YES	Minimum historical integral absorber loading used for some fuel in the multiple misload accident analysis (Region 2 limiting). Optional IFBA credit in Region 1A.
Other credit	YES	SFP wall credited for Region 1B, control rod credit for Region 2, and decay time credit for Region 3.
Fixed neutron absorbers	YES	BORAL in Regions 1 and 2, Boraflex in Region 3 (not credited).
Aging management program	YES	Coupon monitoring (BORAL).
Accident analysis	YES	
Temperature increase	YES	
Assembly drop	YES	Into racks (broken grids pitch change) and between racks.
Single assembly misload	YES	
Multiple misload	YES	Defines a maximum reactivity batch of 24 un-poisoned fresh 5.0 wt % assemblies clustered together surrounded by minimally poisoned fuel.
Boron dilution	YES	
Other	YES	Fuel Handling Accident
Fuel out of rack analysis	YES	
Handling	YES	
Movement	YES	
Inspection	YES	
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/Modules Used for Calculation of k_{eff}	YES	SCALE6.0/CSAS5 – KENO V.a
Cross section library	YES	ENDF/B-VII 238 Group
List all the isotopes used	YES	All TRITON isotopes.
Convergence checks	YES	1000 generations skipped, specific convergence checks on key cases.
Code/Module Used for Depletion Calculation	YES	SCALE6.0/T5-depl – KENO V.a
Cross section library	YES	ENDF/B-VII 238 Group
List all the isotopes used	YES	All SCALE 6.0 (T5-DEPL addnux=3)

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
Convergence checks	YES	Same input (step sizes, neutron histories) as was confirmed to achieve convergence in prior LAR with same fuel design. Additional convergence demonstration cases provided.
Validation of Code and Library	YES	
Major Actinides and Structural Materials	YES	
Minor Actinides and Fission Products	YES	1.5% bias (NUREG/CR-7109)
Absorbers Credited	YES	B (BORAL, IFBA, soluble boron), Cd (RCCA).
7.0 Criticality Safety Analysis of the New Fuel Rack		
Rack model	YES	Full storage area with structure and concrete.
Boundary conditions	YES	Void.
Source distribution	NO	Uniform in fissile material.
Geometry restrictions	NO	
Limiting fuel design	YES	
Fuel density	YES	Bounding high.
Burnable Poisons	NO	No credit.
Fuel dimensions	YES	Composite bounding design.
Axial blankets	NO	No credit.
Limiting rack model		
Storage vault dimensions and materials	YES	
Temperature	YES	
Multiple regions/configurations	NO	
Flooded	YES	
Low density moderator	YES	
Eccentric fuel placement	YES	
Tolerances		
Fuel geometry	YES	
Fuel pin pitch	YES	
Fuel pellet OD	YES	
Fuel clad OD	YES	
Fuel content		
Enrichment	YES	
Density	YES	
Integral absorber	NO	No credit.
Rack geometry		
Rack pitch	YES	
Cell wall thickness	YES	

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
Storage vault dimensions / materials	YES	
Code uncertainty	YES	
Biases		
Temperature	YES	
Code bias	YES	
Moderator Conditions		
Fully flooded and optimum density moderator	YES	
8.0 Depletion Modeling and Burnup Effects		
Depletion Model Considerations		
Time step verification	YES	Verified (similar to prior LARs).
Convergence verification	YES	Verified (similar to prior LARs).
Simplifications	YES	Bounding grid volume homogenized in fuel lattice water, constant depletion conditions (except reduced power for final 40 days).
Non-uniform enrichments	NO	
Post Depletion Nuclide Adjustment	YES	Reduced volatile fission product content (same as prior LARs). Remove very low concentration isotopes.
Cooling time	YES	Time of peak reactivity (5 days) except for decay time credit cases.
Limiting depletion parameters		
Burnable Absorbers	YES	Maximum WABA. IFBA and BPRA have been used at MP3. Maximum WABA bounds maximum IFBA and allows for possible WABA use. Unbounded BPRA from early cycles dispositioned.
Integral absorbers	NO	Maximum IFBA plus source rods bounded by maximum WABA.
Soluble Boron	YES	Bounding
Fuel and Moderator Temperature	YES	Calculated using bounding high fuel assembly power history and bounding high core moderator exit temperature
Power	YES	Bounding high power reduced near end of depletion to maximize depleted fuel reactivity.
Control rod insertion	NO	Bounded by WABA depletion, including maximum IFBA plus control rod insertion. Unbounded CR insertion from early cycles are dispositioned
Atypical Cycle Operating History	YES	Bounded or justified.

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
9.0 Spent Fuel Rack Analysis		
Rack model	YES	Three rack models (Regions 1-3), two configurations in Region 1
Boundary conditions	YES	Periodic X-Y.
Source distribution	YES	Uniform in fissile material
Geometry restrictions	YES	Region 1B must be 2 rows adjacent to SFP wall.
Design Basis Fuel Description	YES	Composite bounding design.
Fuel density	YES	Bounds all fuel assemblies.
Burnable Poisons	YES	Optional credit in Region 1A. Minimum IFBA credit in some fuel in multiple misload analysis.
Fuel assembly inserts	YES	Optional control rod credit, evaluation of depleted BPRA storage.
Fuel dimensions	YES	Present and past MP3 designs bounded.
Axial blankets	NO	No credit taken.
Configurations considered	YES	Evaluated for each rack type, with and without boron.
Borated	YES	
Unborated	YES	
Multiple rack designs	YES	Three rack types.
Alternate storage geometry	YES	Region 1B.
Reactivity Control Devices	YES	
Fuel Assembly Inserts	YES	Optional RCCA credit in Region 2.
Storage Cell Inserts	NO	
Storage Cell Blocking Devices	NO	
Axial burnup shapes		
Uniform/Distributed	YES	NUREG/CR-6801 and uniform, justified with MP3 shapes.
Nodalization	YES	NUREG/CR-6801 18 nodes.
Blankets modeled	NO	No credit except in disposition of unbounded discharged fuel.
Tolerances/Uncertainties		
Fuel geometry		
Fuel rod pin pitch	YES	
Fuel pellet OD	YES	
Cladding OD	YES	
Axial fuel position	YES	
Fuel content		
Enrichment	YES	

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
Density	YES	
Assembly insert dimensions and materials)	NO	Conservative RCCA model used.
Rack geometry		
Flux-trap size (width)	YES	
Rack cell pitch	YES	
Rack wall thickness	YES	
Neutron Absorber Dimensions	YES	
Rack insert dimensions and materials	NO	None present.
Code validation uncertainty	YES	
Criticality case uncertainty	YES	
Depletion Uncertainty	YES	Use ISG 2010-01 (5% burnup worth).
Burnup Uncertainty	YES	4% of measured burnup (developed from measurement uncertainty).
Biases		
Design Basis Fuel design	YES	Built into Base Model
Minor actinides and fission product worth	YES	1.5% of worth (NUREG/CR-7109).
Code bias	YES	
Temperature	YES	Built into Base Model
Eccentric fuel placement	YES	Built into Base Model
Incore thimble depletion effect	YES	Incore thimble included in fuel depletion model.
NRC administrative margin	YES	
Grid growth	YES	Fuel clad creep also included.
Modeling simplifications		
Identified and described	YES	
10.0 Interface Analysis		
Interface configurations analyzed	YES	
Between dissimilar racks	YES	Interface region bias and uncertainty determined based on major components.
Between storage configurations within a rack	YES	Region 1A, 1B.
Interface restrictions	NO	No restrictions needed.
11.0 Normal Conditions		
Fuel handling equipment	YES	Bounding analysis.
Administrative controls	YES	Described.
Fuel inspection equipment or processes	YES	Bounding analysis.
Fuel reconstitution	YES	Evaluated with limitations.

Criticality Analysis Checklist Millstone Unit 3 New Fuel Storage Rack and Spent Fuel Pool		
Subject	Included	Justification / Explanation
12.0 Accident Analysis		
Boron dilution	YES	0 ppm $k_{eff} < 1.0$ including biases and uncertainties.
Normal conditions	YES	$k_{eff} < 0.95$ with minimum dilution analysis boron.
Accident conditions	YES	$k_{eff} < 0.95$ with TS minimum SFP boron.
Single assembly misload	YES	Bounded by multiple misload.
Fuel assembly misplacement	YES	Bounded by multiple misload.
Neutron Absorber Insert Misload	NO	Bounded by multiple misload.
Multiple fuel misload	YES	Maximum anticipated reactivity full fresh fuel batch misload.
Dropped assembly	YES	Between and in racks, bounded by multiple misload.
Temperature	YES	Partial voiding due to boiling considered. Bounded by multiple misload.
Seismic event/ other natural phenomena	NO	No interface effect from shifting racks due to Region design.
13.0 Summary and Conclusions		
Summary of results	YES	
Burnup curve interpolation	YES	Bounding polynomial coefficients.
Intermediate Decay time treatment	NO	Pass / fail.
New administrative controls	NO	
Technical Specification markups	YES	
14.0 References	YES	
Appendix A Computer Code Validation:		
Code validation methodology and bases	YES	NUREG 6698 Method.
New Fuel	YES	
Depleted Fuel	YES	
MOX	YES	
HTC	YES	
Convergence	YES	
Trends	YES	
Bias and uncertainty	YES	
Range of applicability	YES	
Analysis of Area of Applicability coverage	YES	

ATTACHMENT 6

CRITICALITY SAFETY EVALUATION REPORT

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3**

Table of Contents

1	Introduction.....	17
2	Acceptance Criteria and Regulatory Guidance	18
3	Reactor and Fuel Design Description	24
3.1	Fuel Description	24
3.2	Fuel Inserts Description.....	27
3.2.1	Burnable Absorbers	27
3.2.2	Control Rods.....	29
3.2.3	Sources and Other Inserts.....	30
3.3	Non-Standard Fuel in the Pool	31
4	Storage Rack Description	33
4.1	New Fuel Storage Area	33
4.2	Spent Fuel Pool	36
4.2.1	Region 1	37
4.2.2	Region 2	41
4.2.3	Region 3	45
5	Overview of the Method of Analysis	50
5.1	New Fuel Storage Area	50
5.2	Spent Fuel Storage Racks	51
5.2.1	Storage Geometry	51
5.2.2	Bounding Fuel Design.....	53
5.2.3	Soluble Boron Credit.....	54
5.2.4	Burnup Credit	54
5.2.5	Other Credit	57
5.2.6	Neutron Absorbers.....	57
5.2.7	Accident Analysis.....	57
5.2.8	Normal Operations	58
6	Cross Sections, Computer Codes, and Validation.....	59
6.1	Cross Sections and Computer Codes	59
6.1.1	CSAS5	59
6.1.2	Isotopes Used.....	60

6.1.3	TRITON	63
6.2	Uncertainty in Depleted Fuel Isotopic Content	64
6.3	Validation of Criticality Analysis	65
6.3.1	Major Actinides and Structural Materials.....	65
6.3.2	Minor Actinides and Fission Products.....	69
6.3.3	Temperature Dependence.....	69
6.3.4	Absorbers Credited	70
7	Criticality Safety Analysis of the New Fuel Storage Area.....	71
7.1	New Fuel Storage Area KENO Model	71
7.2	Limiting Fuel Design	74
7.2.1	Fuel Dimensions and Materials	74
7.2.2	Fuel Density, Burnable Poisons, and Axial Blankets.....	75
7.3	Limiting Rack Model	77
7.3.1	NFSR Materials and Dimensions.....	77
7.3.2	Temperature and Flooding	79
7.3.3	Low Density Moderator	80
7.3.4	Asymmetric Fuel Placement.....	82
7.3.5	Summary of the Base Case for the NFSA Analysis	83
7.4	Biases and Uncertainties for the New Fuel Storage Area Analysis	84
7.5	Accident Conditions	88
7.5.1	Optimum Moderation	88
7.5.2	Dropped/Misplaced Assembly.....	88
7.5.3	Seismic Event	88
8	Depletion Modeling and Burnup Effects	89
8.1	Depletion Method Overview	89
8.2	Bounding Fuel Assembly Depletion Power.....	90
8.3	Bounding Depletion Boron.....	93
8.4	Bounding RCS Temperature.....	94
8.5	Bounding Fuel Temperature.....	95
8.6	Bounding Axial Burnup Profiles	97
8.7	Burnable Absorbers.....	98
8.8	Control Rod History	100
8.9	In-core Thimble	102
8.10	Reduced Power before Storage	103

8.11	Grid Growth and Clad Creep Depletion Effects.....	104
8.12	Grids.....	104
8.13	Instrument and Guide Tube Design	105
8.14	Comparison of CASMO and TRITON Depletion Reactivity.....	106
8.15	TRITON Depletion Model Summary	108
9	Spent Fuel Rack Analysis	110
9.1	Region 1 Analysis	110
9.1.1	Region 1 Fuel Storage.....	110
9.1.2	Region 1 Modeling Assumptions	110
9.2	Region 1 Infinite Lattice KENO Model.....	112
9.2.1	BORAL Blisters	114
9.2.2	SFP Normal Operation Water Temperature and Density	114
9.2.3	Simplified TRITON Depletion Input for Burnup Credit	115
9.2.4	IFBA Credit.....	115
9.2.5	Asymmetric Fuel Placement.....	115
9.2.6	Code Validation Bias and Uncertainty	117
9.2.7	Measured Burnup Uncertainty.....	117
9.2.8	Summary of Bias and Uncertainty.....	119
9.2.9	Region 1 Infinite Lattice k-eff and Margin Calculation	121
9.3	Region 1 KENO SFP Wall Credit Model	124
9.4	Region 1 Analysis Summary	127
9.5	Region 2 Analysis	128
9.5.1	Region 2 Fuel Storage.....	128
9.5.2	Region 2 Modeling Assumptions	128
9.6	Region 2 Infinite Lattice Model.....	129
9.6.1	BORAL Blisters	132
9.6.2	SFP Normal Operation Water Temperature and Density	132
9.6.3	TRITON Depletion Input for Burnup Credit.....	132
9.6.4	Asymmetric Fuel Placement.....	132
9.6.5	Fuel Geometry Changes with Burnup.....	132
9.6.6	Horizontal Burnup Tilt.....	137
9.6.7	Code Validation Bias and Uncertainty	141
9.6.8	Summary of Bias and Uncertainty.....	142
9.6.9	Unbounded Historical Fuel	144
9.6.10	Region 2 k-eff and Margin Calculation – Fresh fuel	145

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 5 of 300

9.6.11	Region 2 Control Rod Credit.....	147
9.6.12	Region 2 k-eff and Margin Calculation – Burnup credit.....	150
9.7	Region 2 Analysis Summary	155
9.8	Region 3 Analysis	156
9.8.1	Region 3 Fuel Storage.....	156
9.8.2	Region 3 Modeling Assumptions	156
9.9	Region 3 Infinite Lattice Model.....	157
9.9.1	SFP Normal Operation Water Temperature and Density	160
9.9.2	TRITON Depletion Input for Burnup Credit.....	160
9.9.3	Asymmetric Fuel Placement.....	160
9.9.4	Fuel Geometry Changes with Burnup.....	160
9.9.5	Horizontal Burnup Tilt.....	161
9.9.6	Code Validation Bias and Uncertainty	163
9.9.7	Summary of Bias and Uncertainty.....	163
9.9.8	Unbounded Historical Fuel	165
9.9.9	Region 3 k-eff and Margin Calculation Without Decay Time Credit	168
9.9.10	Region 3 k-eff and Margin Calculation With Decay Time Credit	173
9.10	Region 3 Analysis Summary	182
10	Interface Analysis	183
10.1	Interfaces Between Storage Configurations Within a Rack	183
10.2	Interfaces Between Dissimilar Racks	183
10.2.1	MPS3 Region Interfaces.....	184
10.2.2	Interface Analysis Method	184
10.3	Region 1-2 Interface	185
10.3.1	Region 1-2 Interface with Region 2 Fresh Fuel.....	189
10.3.2	Region 1-2 Interface with Region 2 Depleted Fuel	189
10.4	Region 2-3 Interface	193
10.4.1	Region 2-3 Interface (Fresh / Fresh).....	195
10.4.2	Region 2-3 Interface (Fresh / Depleted)	195
10.4.3	Region 2-3 Interface (Depleted / Fresh)	198
10.4.4	Region 2-3 Interface (Depleted / Depleted)	202
10.5	Interface Analysis Summary	205
11	Normal Conditions	206
11.1	Fuel Handling	206

11.2	Fuel Inspection	208
11.3	Non-standard Fuel.....	208
11.3.1	Fuel Rod Storage Canister.....	208
11.3.2	Reconstituted Fuel	210
11.4	Normal Condition Boron Credit.....	215
12	Accident Analysis.....	221
12.1	Accident Condition Soluble Boron Requirement.....	221
12.1.1	Loss of Cooling	221
12.1.2	Single Mis-placement.....	222
12.1.3	Multiple Mis-load	226
12.1.4	Dropped Assembly	232
12.1.5	Handling Error	233
12.1.6	Seismic Event	234
12.2	Accident Analysis Summary	234
13	Summary and Conclusions.....	235
13.1	New Fuel Storage Area	235
13.2	Spent Fuel Pool	236
13.3	Bounding Fuel Design Values	236
13.4	Bounding Depletion Condition Input	236
13.5	Summary of Loading Constraints	237
14	References	243
Appendix A: Validation for Criticality Analysis Using Laboratory Critical Experiments		247
A.1.	Overview.....	247
A.2.	Definition of the Range of Parameters to Be Validated	248
A.3.	Selection of the Critical Benchmark Experiments.....	248
A.3.1	Selection of the Fresh UO ₂ Critical Benchmark Experiments	248
A.3.2	Selection of MOX Critical Experiments.....	258
A.4.	Modeling and Calculating k of the Critical Experiments.....	260
A.5.	Statistical Analysis of the Data.....	271
A.5.1	Statistical Analysis of the UO ₂ Critical Experiments.....	273
A.5.2	Statistical Analysis of MOX Critical Experiments	287
A.5.3	Subcritical Margin	292

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 7 of 300

A.6. **Area of Applicability (Benchmark Applicability)293**

A.7. **Summary and Recommendations294**

A.8. **Temperature Bias296**

A.9. **Appendix References300**

List of Tables

Table 2.1: Use of DSS-ISG-2010-01.....	21
Table 3.1: Fuel Design Dimensions.....	24
Table 3.2: Fuel Design Tolerances	25
Table 3.3: Feed Fuel Design and Enrichment History for Millstone Unit 3	26
Table 3.4: Description of Burnable Poisons	28
Table 3.5 Description of Control Rods	29
Table 3.6: Description of In-core Thimble	30
Table 4.1: New Fuel Storage Rack Dimensions	34
Table 4.2: Axial Heights of the New Fuel Storage Area KENO Model (Inches).....	35
Table 4.3: Region 1 Rack Dimensions and Tolerances.....	40
Table 4.4: Region 1 Rack Axial Alignment.....	41
Table 4.5: Region 2 Racks.....	42
Table 4.6: Region 2 Rack Dimensions and Tolerances.....	44
Table 4.7: Region 2 Rack Axial Alignment.....	44
Table 4.8: Region 3 Rack Model X-Y Dimensions (cm)	47
Table 4.9: Region 3 Rack Model Axial Dimensions	49
Table 5.1 – Storage Geometries.....	51
Table 6.1: Isotopes Used in the Criticality Analysis	60
Table 6.2: Fission Product Gases and Volatiles	61
Table 6.3: Fission Gas Release Fractions.....	62
Table 6.4	NOT USED
Table 6.5	NOT USED
Table 6.6: Summary of Validation Bias and Uncertainty From Major Actinides and Structural Materials	68
Table 7.1: EPRI Dry Concrete Composition	77
Table 7.2: k-eff of NFSR vs. Concrete Composition	78
Table 7.3: Axial Location of Active Fuel for Various Fuel Design Types	78
Table 7.4: k-eff of NFSR vs. Active Fuel Axial Location.....	79
Table 7.5: NFSR Reactivity Sensitivity to Water Density	79
Table 7.6: NFSR Reactivity Sensitivity to Water Density	80
Table 7.7: k-eff of NFSR vs. Assembly Asymmetric Position	83

Table 7.8: MPS3 NFSR Bias and Uncertainty Cases	84
Table 7.9: MPS3 NFSR Maximum k-eff, Full Density Water	85
Table 7.10: Maximum k-eff for MPS3 NFSR, Optimum Moderation	87
Table 8.1: Bounding Burnup Averages Relative Assembly Power versus Burnup.....	92
Table 8.2: RCS Thermal Hydraulic History for MPS3	95
Table 8.3: Example of Nodal Depletion Conditions (20 GWd/MTU).....	96
Table 8.4	NOT USED
Table 8.5	NOT USED
Table 8.6: Depletion with Different Burnable Absorbers (5.0 wt%, 54 GWD/T).....	98
Table 8.7: Depletion with IFBA and WABA Simultaneously (Node 16)	99
Table 8.8: Control Rod Dimensions.....	100
Table 8.9: Control Rod Use Sensitivity Cases	102
Table 8.10: In-Core Thimble and Instrument Tube Dimensions	102
Table 8.11: Low Power EOC Effect on SFP Reactivity (5.0 w/o, 54 GWD/T)	103
Table 8.12: SFP Grid Modeling Effect (Node 16, Fresh Fuel, 0 ppm soluble boron)	105
Table 8.13: Effect of Depleting with Grids (Node 16, 0 ppm soluble boron).....	105
Table 8.14: Instrument and Guide Tube Dimensions	105
Table 8.15: Comparison of CASMO and TRITON Depletion Worth	107
Table 8.16: Depletion Parameters for TRITON Depletion Model.....	108
Table 9.1: SFP Normal Operation Water Temperature and Density	114
Table 9.2: Validation Bias and Uncertainty ($EALF \leq 0.35$ eV, Appendix A).....	117
Table 9.3: Summary of Region 1 Biases, Uncertainties, and Conservatism	120
Table 9.4: Selected Tolerance Results for Region 1 (Fresh 4.75 wt% fuel)	121
Table 9.5: Worth of Burnup and Minor Actinides and Fission Products	122
Table 9.6: Region 1 Total Bias, Uncertainty, and Margin (0 ppm soluble boron).....	123
Table 9.7: Region 1 Wall Model Base and Sensitivity Cases	126
Table 9.8	NOT USED
Table 9.9	NOT USED
Table 9.10: Horizontal Burnup Tilt Bias	141
Table 9.11: Region 2 Validation Bias and Uncertainty	142
Table 9.12: Summary of Region 2 Biases, Uncertainties, and Conservatism.....	143
Table 9.13: Selected Tolerance Results for Region 2 (Fresh fuel)	145
Table 9.14: Region 2 Total Bias, Uncertainty, and Margin (0 ppm soluble boron)	146

Table 9.15: MPS3 Control Rod Data	148
Table 9.16: Region 2 KENO Control Rod Credit Cases (5.0 w/o fresh fuel)	149
Table 9.17: SFP Region 2 Limiting Axial Shapes.....	150
Table 9.18: Enrichment Tolerance Calculation Comparison	152
Table 9.19: Region 2 Base Cases (0 ppm boron).....	152
Table 9.20: Region 2 Total Bias, Uncertainty, and Margin	153
Table 9.21	NOT USED
Table 9.22: Region 3 Validation Bias and Uncertainty	163
Table 9.23: Summary of Region 3 Biases, Uncertainties, and Conservatism.....	164
Table 9.24: Depletion Conditions Comparison for Assembly B24	166
Table 9.25: Modeling Assembly B24 with Measured Design Inputs.....	167
Table 9.26: Temperature and Burnup Shape Test Cases.....	168
Table 9.27: SFP Region 3 Limiting Axial Shapes.....	169
Table 9.28: Region 3 Total Bias, Uncertainty, and Margin (0 years decay).....	171
Table 9.29: Uncertainty Cases for Region 3 Decay Time Burnup Curves.....	175
Table 9.30: Region 3 Total Bias, Uncertainty, and Margin (3 years decay).....	177
Table 9.31: Region 3 Total Bias, Uncertainty, and Margin (9 years decay).....	178
Table 9.32: Region 3 Total Bias, Uncertainty, and Margin (18 years decay).....	179
Table 9.33: Region 3 Total Bias, Uncertainty, and Margin (25 years decay).....	180
Table 9.34: Region 3 Burnup Credit Curve Coefficients	181
Table 10.1: Region 1-2 Interface with Fresh Fuel	189
Table 10.2: Region 1-2 Interface (Region 1 Fresh Fuel and Region 2 Depleted Fuel)	189
Table 10.3: Region 1-2 Interface Model Selected Tolerances (Fresh/Depleted)	190
Table 10.4: Region 1-2 Interface Model Tolerance Comparison (Fresh / Depleted)	191
Table 10.5: Region 1-2 Interface Model Margin (Fresh / Depleted).....	192
Table 10.6: Region 2-3 Interface (Fresh / Fresh).....	195
Table 10.7: Region 2-3 Interface (Fresh / Depleted)	196
Table 10.8: Region 2-3 Interface Model Selected Tolerances (Fresh / Depleted)	196
Table 10.9: Region 2-3 Interface Model Tolerance Comparison (Fresh / Depleted)	197
Table 10.10: Region 2-3 Interface Model Margin (Fresh / Depleted).....	198
Table 10.11: Region 2-3 Interface (Depleted / Fresh).....	199
Table 10.12: Region 2-3 Interface Model Selected Tolerances (Depleted / Fresh)	199

Table 10.13: Region 2-3 Interface Model Tolerance Comparison (Depleted / Fresh)	200
Table 10.14: Region 2-3 Interface Model Margin (Depleted / Fresh).....	201
Table 10.15: Region 2-3 Interface (Depleted / Depleted)	202
Table 10.16: Region 2-3 Interface Model Selected Tolerances (Depleted / Depleted)	203
Table 10.17: Region 2-3 Interface Model Tolerance Comparison (Depleted / Depleted) ..	203
Table 10.18: Region 2-3 Interface Model Margin (Depleted / Depleted).....	204
Table 11.1: Region 2 and 3 Fuel Rod Storage Canister Model Results.....	210
Table 11.2: Fuel Reconstitution Assembly Isolation	212
Table 11.3: Fuel Reconstitution Fuel Rod Replacement	213
Table 11.4: Region 1-3 Normal Condition Boron Requirement Cases (600 ppm).....	217
Table 11.5: Region 1 Bias, Uncertainty, and Margin Comparison	218
Table 11.6: Region 2 Bias, Uncertainty, and Margin Comparison (fresh fuel).....	219
Table 11.7: Region 2 Bias, Uncertainty, and Margin Comparison (depleted fuel).....	220
Table 12.1: Loss of Cooling Event.....	222
Table 12.2: Single Fuel Assembly Mis-placement	225
Table 12.3: Millstone Unit 3 Fuel Batch History	227
Table 12.4: Multiple Mis-load Base Case Results	229
Table 12.5: Multiple Mis-load Tolerance Results (2550 ppm boron).....	230
Table 12.6: Multiple Mis-load Bias, Uncertainty and Margin	231
Table 12.7: Dropped Assembly Results	233
Table 12.8: Dropped Assembly Results	233
Table 13.1: Bounding Fuel Design Values	235
Table 13.2: Bounding Fuel Design Values	236
Table 13.3: Bounding Depletion Conditions for Burnup Credit.....	237
Table 13.4: MPS3 SFP Storage Constraint Summary	238
Table 13.5: Burnup Credit Curve Polynomial Coefficients	242
Table A.3.1: Selection Review of OECD/NEA Criticality Benchmarks.....	250
Table A.4.1: UO2 Critical Experiment Results with SCALE 6.0 and ENDF/B-VII.....	261
Table A.4.2: HTC Critical Experiment Results with SCALE 6.0 and ENDF/B-VII.....	267
Table A.4.3: Results of Low Enriched MOX Critical Experiments Calculated with SCALE	270

Table A.5.1: Bias and Uncertainty Based on the EALF Trend from UO2 Critical Experiments	279
Table A.5.2: UO2 Critical Experiment With Soluble Boron Results with SCALE 6.0 and ENDF/B-VII.....	284
Table A.5.3: Bias and Uncertainty Based on the EALF Trend from UO2 Critical Experiments	286
Table A.5.4: Bias and Uncertainty Based on the EALF Trend from MOX Critical Experiments	292
Table A.6.1: Area of Applicability (Benchmark Applicability)	293
Table A.7.1: Summary of the Trend Analysis.....	295
Table A.7.2: Final Bias and Uncertainty	296
Table A.8.1: LCT-046 with Full Thermal Expansion Calculated with SCALE 6.0 and ENDF/B-VII.....	297

List of Figures

Figure 3.1: 17x17 Fuel Design.....	25
Figure 3.2: Fuel Rod Storage Container Schematic	32
Figure 4.1: Millstone Unit 3 New Fuel Storage Area	33
Figure 4.2: Axial Drawing of a Millstone Unit 3 New Fuel Storage Rack Module	35
Figure 4.3: MPS3 SFP Rack Orientation.....	36
Figure 4.4: MPS3 Region 1 Rack Dimensions (X-Y)	38
Figure 4.5: MPS3 Region 1 Rack Dimensions (X-Z)	39
Figure 4.6: MPS3 Region 2 Rack Dimensions (X-Y)	42
Figure 4.7: MPS3 Region 2 Rack Dimensions (X-Z)	43
Figure 4.8: Location of Region 3 Racks in the SFP	45
Figure 4.9: X-Y Drawing of a Region 3 Rack Cell.....	46
Figure 4.10: X-Y Drawing of a Region 3 Rack Wrapper.....	46
Figure 4.11: Axial Drawing of a Region 3 Rack Cell	48
Figure 5.1: KENO Region 1A/1B Wall Credit Model.....	52
Figure 7.1: Cutaway View of the New Fuel Storage Area Model.....	72
Figure 7.2: Top View of Part of the New Fuel Storage Area Model.....	73
Figure 7.3: Full New Fuel Storage Area Model.....	73
Figure 7.4: MPS3 Fuel Assembly Stack Density	75
Figure 7.5: k-eff of NFSR vs. Water Density.....	81
Figure 7.6: Asymmetric Skewing of Assemblies	82
Figure 8.1: Bounding Burnup Averaged Relative Assembly Power.....	92
Figure 8.2: MPS3 Bounding Cycle Average Soluble Boron	93
Figure 8.3	NOT USED
Figure 8.4	NOT USED
Figure 8.5: Cycle Average Control Rod Insertion.....	101
Figure 9.1: X-Y Representation of Region 1 KENO Model	112
Figure 9.2: 3D Representation of Region 1 KENO Model (full model and cutaway).....	113
Figure 9.3: Asymmetric Fuel Placement (4x4 in a 6x6 model).....	116
Figure 9.4: KENO Region 1 Wall Credit Model.....	125
Figure 9.5	NOT USED
Figure 9.6	NOT USED

Figure 9.7: X-Y Representation of Region 2 2x2 KENO Model	130
Figure 9.8: 3D Representation of Region 2 2x2 KENO Model.....	131
Figure 9.9: Zircaloy-4 Grid Growth [38].....	135
Figure 9.10: ZIRLO Grid Growth Data [42]	136
Figure 9.11: Horizontal Burnup Tilt KENO Model #1	138
Figure 9.12: Horizontal Burnup Tilt KENO Model #2	139
Figure 9.13: Quadrant Tilt vs. Burnup	140
Figure 9.14: Region 2 Bounding Burnup Credit Curve	154
Figure 9.15: X-Y Representation of Region 3 KENO Model	158
Figure 9.16: Axial Representation of Region 3 KENO Model.....	159
Figure 9.17: Region 3 Horizontal Burnup Tilt KENO Model #1	162
Figure 9.18: Region 3 Horizontal Burnup Tilt KENO Model #2	162
Figure 9.19: Region 3 Bounding Burnup Credit Curve (0 Years Decay)	172
Figure 9.20: Region 3 Bounding Burnup Credit Curves	181
Figure 10.1: X-Y Representation of the Region 1-2 Interface Model	186
Figure 10.2: X-Y Representation of the Region 1-2 Interface Showing Flux Traps	188
Figure 10.3: X-Y Representation of the Region 2-3 Interface Model	193
Figure 11.1: k-eff of Two Fresh 5 wt% Fuel Assemblies in Unborated Water	207
Figure 11.2: Location of MPS3 Fuel Handling Equipment	207
Figure 11.3: Region 2 Fuel Rod Storage Canister Model	209
Figure 11.4: Region 3 Fuel Rod Storage Canister Model	209
Figure 11.5: Region 2 Reconstitution Model.....	211
Figure 11.6: Region 3 Reconstitution Model.....	212
Figure 11.7: Region 3 Fuel Assembly MR71 Model	214
Figure 12.1: Region 1-2 Interface Model for Between-rack Mis-placement	223
Figure 12.2: Region 2-3 Interface Model for Between-rack Mis-placement	224
Figure 12.3.....	NOT USED
Figure 12.4: Region 2 KENO Multiple Mis-load Model (partial X-Y view)	228
Figure 12.5: Region 2 Dropped Assembly Model	232
Figure 13.1: Region 1A/1B Orientation.....	239
Figure 13.2: Region 2 Burnup Credit Curve	240
Figure 13.3: Region 3 Burnup Credit Curves	241

Figure A.5.1: Distribution of the Calculated k's Around the Mean for All the UO2 Benchmarks	275
Figure A.5.2: Q-Q Normality Plot for the UO2 Benchmarks	275
Figure A.5.3: Calculated k for the UO2 Critical Benchmarks as a Function of EALF	280
Figure A.5.4: Calculated k for the UO2 Critical Benchmarks as a Function of Pin Diameter	281
Figure A.5.5: Calculated k for the UO2 Critical Benchmarks as a Function of Fuel Pin Pitch.....	282
Figure A.5.6: Calculated k for the UO2 Critical Benchmarks as a Function of Enrichment	283
Figure A.5.7: Calculated k for the UO2 Critical Benchmarks as a Function of Soluble Boron.....	285
Figure A.5.8: Calculated k for the Critical Benchmarks as a Function of Plutonium Content.....	288
Figure A.5.9a: Distribution of Calculated ks for the MOX Critical Benchmarks	289
Figure A.5.9b: Q-Q Normality Plot for the MOX Benchmarks	290
Figure A.5.10: Calculated MOX Critical k as a Function of EALF	291
Figure A.8.1: LCT-046 Corrected Calculated k per Case.....	298
Figure A.8.2: LCT-046 k versus Temperature	299

Annotation of proprietary information herein corresponds to the specific reason(s) for claiming the information as proprietary as delineated in the respective Affidavit executed by the owners of the information. The annotations used in Attachment 5 are provided as follows:

- 1) Holtec proprietary information - denoted with "H" superscripts:
- 2) Westinghouse proprietary information - denoted with "a,c" superscripts:

List of Acronyms and Abbreviations

BPRA	Burnable Poison Rod Assembly is a group of boron containing rodlets held together by a plate that rests on the assembly top nozzle.
DOE	Department of Energy (USA)
EALF	Energy of the Average Lethargy of a neutron causing Fission
EOL	End of Life
FA	Fuel Assembly
FRSC	Fuel Rod Storage Container
FP	Fission Product
GWd/MTU	Gigawatt*day per Metric Ton (Tonne) of Uranium. A unit of burnup.
HTC	Haut Taux de Combustion. This is a set of critical experiments done in France that uses fuel that represents the uranium and plutonium content of 4.5 wt% fuel burned to 37.5 GWd/MTU
ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber which is a ZrB ₂ coating placed on the outside of the fuel pellet. A Westinghouse product.
ISG	Interim Staff Guidance from the NRC
MOX	Mixed Oxide fuel. Contains both UO ₂ and PuO ₂ .
MPS3	Millstone Power Station Unit 3
MW/MTU	Megawatt per Metric Ton (Tonne) of Uranium. A unit of specific power.
MWd/MTU	Megawatt*day per Metric Ton (Tonne) of Uranium. A unit of burnup.
NCS	Nuclear Criticality Safety
NFSA	New Fuel Storage Area. Referring to the concrete pit and steel racks used to hold the new fuel. Sometimes used interchangeably with NSFR.
NSFR	New Fuel Storage Racks. Referring to the racks inside the New Fuel Storage Area. Sometimes used interchangeably with NSFA.
OD	Outer Diameter
pcm	0.00001 in k (acronym from percent mille)
ppm	Parts per million by weight
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly (control rod)
RCS	Reactor Coolant System
RFA	Robust Fuel Assembly is a fuel design manufactured by Westinghouse.
RSS	square Root of the Sum of the Squares
RTP	Rated Thermal Power
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SNF	Spent Nuclear Fuel
SS	Stainless Steel
VF	Volume Fraction
WABA	Wet Annular Burnable Absorbers. This is a Westinghouse removable burnable absorber product.

1 Introduction

A SFP criticality analysis is performed for Millstone Unit 3 (MPS3) to allow for a measurement uncertainty recapture power uprate, to incorporate spent fuel pool (SFP) soluble boron credit, to update computer codes and analysis methods, to eliminate the need for empty storage cells and cell blockers, and to increase identified margin to k-effective (k-eff) limits. A new fuel storage area (NFSA) criticality analysis is performed for MPS3 to update computer codes and analysis methods and to increase identified margin to k-eff limits. The current enrichment limit for the MPS3 NFSA and SFP is 5.0 wt% U-235.

Criticality analysis scope includes normal fuel storage, normal fuel handling, abnormal fuel storage conditions, abnormal fuel handling events, and storage of non-standard fuel-bearing components. A boron dilution analysis is provided that identifies potential SFP dilution events, dilution water source flow rates and volumes, means of detection, time required for detection and mitigation, and minimum dilution event soluble boron concentration.

Changes to the operation and administration of the SFP are summarized as follows:

- Modify storage requirements in Technical Specifications 1.40, 1.41, 3/4.9.13 and 3/4.9.14, including Figures detailing storage patterns and minimum fuel burnup versus fuel enrichment.
- Eliminate the requirement for cell blockers in Region 1.
- Modify SFP soluble boron requirements in Technical Specifications 3/4.9.1.2.
- Modify the description of SFP criticality control in Technical Specification 5.6.1.1.

No changes to the operation and administration of the NFSA are proposed.

Fuel burnup when used to compare to fuel storage requirements is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty. Enrichment when used to compare to fuel storage requirements is the maximum planar volume averaged as-built initial enrichment in the assembly. If the assembly has axial blankets, the lower enriched fuel is not credited in determining the enrichment.

2 Acceptance Criteria and Regulatory Guidance

The Code of Federal Regulations Title 10 Part 50 Section 68 (b).4 states:

"If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

This analysis shows at a 95 percent probability and 95 percent confidence level that if the fuel loaded in the SFP meets the Technical Specification requirement for enrichment, burnup, and decay time, the SFP k-eff will be less than 0.95 crediting soluble boron and less than 1.0 with unborated water as specified in 10CFR50.68. [1]

Further, Title 10 Part 50 Section 68 (b) paragraphs 2 and 3 state:

"(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used."

"(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

This analysis of the fresh fuel storage racks shows that the storage racks meet the requirements of paragraphs 2 and 3 when fully loaded with fresh 5.0 wt% U-235 fuel assemblies.

Meeting 10CFR50.68 also satisfies 10CFR50, Appendix A, General Design Criterion 62 [2] which states:

"Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 19 of 300

Guidance for the regulatory review is found in the Standard Review Plan, NUREG-0800 Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling." [3] Below are the eleven specific areas of review from this guidance with where the information is found in this report. (Note that there are actually 13 areas of review but the last two do not apply to License Amendment Requests.)

1. *Fuel assembly design to verify that appropriate fuel assembly data were used.*

Fuel assembly design data is found in Section 3.

2. *Fuel storage rack design to verify that appropriate fuel storage rack data were used.*

Storage rack design data is found in Section 4.

3. *Evaluation of performance effectiveness of the neutron absorbing materials in the fresh and spent fuel racks.*

BORAL is credited in the NFSA and in Region 1 and 2 SFP racks. A BORAL coupon monitoring program description has been provided to the NRC. [5, 6]

4. *Computational methods and related data to verify that acceptable computational methods and data were used.*

Computational methods are described in Sections 5 and 6.

5. *Computational method validation to verify that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and abnormal conditions models and to verify that the neutron distribution coefficient (K_{eff}) bias and bias uncertainty values are conservatively determined.*

Validation is summarized in Section 6 and the details are provided in Appendix A.

6. *Identification of normal conditions to verify that the scope of specified normal conditions is comprehensive.*

Range of normal conditions is identified in Section 11

7. *Normal-conditions models to verify that normal conditions are modeled conservatively and that all modeling approximations and assumptions are appropriate.*

Normal conditions models and the tolerances and uncertainties in these models are described in Sections 7 - 11.

8. *Identification of abnormal conditions to verify that the scope of considered abnormal conditions is comprehensive.*

Abnormal conditions are described in Section 7 and 12.

9. Abnormal-conditions models to verify that abnormal conditions are modeled conservatively and that all modeling approximations and assumptions are appropriate.

Abnormal condition models are described in Section 12.

10. Analysis of normal and credible abnormal conditions to verify that the analysis is complete and logically sound and that assumptions, limits, and controls are clearly stated.

The analysis is contained in Sections 7 through 12. The limitations of the analysis are listed in Section 13.

11. Analysis conclusions to verify the applicant's conclusions regarding maintaining subcriticality for all normal and credible abnormal conditions.

The conclusion of the analysis is in Section 13.

Guidance for spent fuel pool criticality analysis is given in DSS-ISG-2010-01. [4]

Table 2.1: Use of DSS-ISG-2010-01

Guidance from DSS-ISG-2010-01	Implementation	Section in this Report
1. Fuel Assembly Selection The NCS analysis must adequately bound all designs and variations within a design.	MPS3 fuel designs and variations are described. Variations are bounded by the analysis for both depletions and criticality calculations by use of a hybrid assembly design and biases as needed. Zr was used to conservatively represent all Zr based grid and clad alloys.	Sections 3, 7.2, 8.7, 8.9, 8.11, 8.12, and 8.13
2. Depletion Analysis a.i. Depletion uncertainty (5%) covers only isotopic concentration uncertainty.	Used 5% depletion uncertainty only for isotopic concentration uncertainty. Code validation covers major actinide cross section uncertainty. Bias and uncertainty in the worth of fission products and minor actinides is covered by bias of 1.5% of the fission production and minor actinides worth.	Sections 6.2, 6.3, 9.2.8, 9.6.8, and 9.9.9
2. Depletion Analysis a.ii. Reactivity decrement should not include the worth of the burnable absorbers.	Followed.	Section 6.2, 9.2.7, 9.6.8, 9.9.9
2. Depletion Analysis b.i. Bounding values should be used.	Bounding values are identified and used	Section 8
2. Depletion Analysis b.ii. Use the more limiting bounding parameter when a conflict occurs.	Fuel and moderator temperatures are maximized based on high specific power. Depletion at reduced power is performed near EOL without changing moderator and fuel temperatures.	Section 8
2. Depletion Analysis b.iii. Non-bounding values are outside scope of ISG.	Bounding values were used for all parameters or justification provided.	Section 8.15
2. Depletion Analysis c.i. All removable burnable absorbers must be considered.	All removable burnable absorbers are identified and the most limiting burnable absorbers are used. Justification provided for early cycle BPRA.	Section 8.7
2. Depletion Analysis c.ii. Limiting integral burnable absorbers should be used.	Reference depletions use maximum WABA, which bounds the effect of maximum IFBA. Maximum IFBA plus some WABA is also considered.	Section 8.7
2. Depletion Analysis c.iii. Model the burnable absorbers appropriately.	WABA are conservatively modeled in the depletion analysis as full length with no cutback region and contain the maximum design B-10 loading.	Section 8.7
2. Depletion Analysis c.iv. Consider competing effects	The depletion model correctly accounts for competing effects.	Section 8.15
2. Depletion Analysis d.i. Spectrum hardening from rodged operation should be considered.	Rodded operation is bounded by the standard depletion. Early cycle exceptions are justified.	Sections 8.8 and 9.9.8

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 22 of 300

Guidance from DSS-ISG-2010-01	Implementation	Section in this Report
2. Depletion Analysis d.ii. Effect of control rods on the axial burnup profile should be considered	NUREG/CR-6801 axial burnup profiles were used, which include rodged operation effects.	Section 8.6
3. Criticality Analysis a. Axial Burnup Profile	NUREG/CR-6801 axial burnup profiles were used and justified by comparison to MPS3 profiles. Analysis was also done with uniform burnup and the most limiting of the two profiles was used.	Section 8.6
3. Criticality Analysis b. Rack Model i. Model inputs should be traceable.	The rack dimensions and materials are taken from the manufacturer's drawings.	Section 4
3. Criticality Analysis b. Rack Model ii. Efficiency of the neutron absorber should be established.	BORAL is modeled with minimum certified areal density. BORAL coupon program is used to verify conformity of the rack absorber and the analysis. RCCA credit assumes maximum anticipated RCCA depletion over the portion of absorber inserted during plant operation.	Sections 4.2, 5.2.6, and 9.6.11
3. Criticality Analysis b. Rack Model iii. Conservative degradation should be used.	RCCA credit assumes maximum anticipated RCCA depletion over the portion of absorber inserted during plant operation. Conservative (low) BORAL B-10 content used in analysis is confirmed by coupon monitoring program.	Sections 4.2, 5.2.6, and 9.6.11
3. Criticality Analysis c. Interfaces - If not determining interface model biases and uncertainties, use the maximum uncertainties from either side.	Either the maximum uncertainty from either side is used or the biases and uncertainties for the interface model are calculated.	Section 10
3. Criticality Analysis d. Normal Conditions - All normal conditions such as movement of fuel and inspections should be considered.	Normal conditions are considered.	Section 11
3. Criticality Analysis e. Accident Conditions	Normal initial conditions are considered as base conditions for the accident analysis.	Section 12
4. Criticality Code Validation NUREG/CR-6698 endorsed	NUREG/CR-6698 was followed for the validation.	Appendix A
4. Criticality Code Validation a. Area of Applicability i. Include the HTC criticals	The HTC critical experiments are included in the analysis.	Appendix A
4. Criticality Code Validation a. Area of Applicability ii. Use appropriate criticals	Appropriate critical experiments are used.	Appendix A

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 23 of 300

Guidance from DSS-ISG-2010-01	Implementation	Section in this Report
4. Criticality Code Validation a. Area of Applicability iii. Sufficient criticals for analysis and appropriate grouping.	A large sample of critical experiments is used.	Appendix A
4. Criticality Code Validation a. Area of Applicability iv. Be sure the set is not highly correlated.	Correlation avoided by use of experiments from multiple critical facilities.	Appendix A
4. Criticality Code Validation b. Trend Analysis Adequate, appropriate, not rejected.	Trend analysis is performed on the major parameters. The trend analysis finds the best linear fit. No trends are rejected to be conservative. The most limiting bias and uncertainty for the area of applicability is applied assuming both that all trends are real and there are no trends.	Appendix A
4. Criticality Code Validation c. Statistical Treatment i. Use the variance of the population about the mean	The statistical approach recommended in NUREG/CR-6698 is used. The variance of the population about the mean rather than the variance of the mean is used.	Appendix A
4. Criticality Code Validation c. Statistical Treatment ii. Use correct confidence factors.	The statistical approach recommended in NUREG/CR-6698 is used. The correct confidence factors were used.	Appendix A
4. Criticality Code Validation c. Statistical Treatment iii. Consider Normality	Normality testing was performed and the assumption of normality justified.	Appendix A
4. Criticality Code Validation d. Lumped Fission Products	Lumped Fission Products are not used.	Section 6.1
4. Criticality Code Validation e. Code-to-Code Comparisons	5% of the delta k of depletion is used to cover isotopic uncertainty. The analysis in the North Anna Power Station LAR dated May 2, 2017 confirmed this is appropriate for SCALE/TRITON by comparisons to CASMO-4 and 5.	Sections 6.2 & 8.14
5. Miscellaneous a. Precedence b. References c. Assumptions	Precedence is not quoted as a licensing basis. References used were carefully chosen to be applicable to the point being made. Assumptions are identified and justified.	

3 Reactor and Fuel Design Description

3.1 Fuel Description

MPS3 uses a 17x17 lattice fuel with a center instrument tube and 24 guide tubes. Four fuel designs have been used but all the designs are similar for the criticality analysis. The initial fuel design, which is designated "Standard," used all Inconel grids. The currently used fuel design is the Westinghouse RFA-2 design. Table 3.1 provides fuel dimensions.

Table 3.1: Fuel Design Dimensions
(Dimensions in inches)

	Standard	V5H	RFA/RFA-2	NGF
Pellet Diameter	0.3225	0.3225	0.3225	0.3225
Clad Inner Diameter	0.329	0.329	0.329	0.329
Clad Outer Diameter	0.374	0.374	0.374	0.374
Clad Material	Zircaloy-4	Zircaloy-4 ZIRLO	ZIRLO Opt. ZIRLO	Opt. ZIRLO
Rod Pitch	0.496	0.496	0.496	0.496
Guide Tube and Instrument Tube				
Inner Diameter	0.450	0.442	0.442	0.442
Outer Diameter	0.482	0.474	0.482	0.482
Grid Volume (cubic inches)¹	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}

¹The grid volume is the volume of the grids plus sleeves in the active fuel ignoring the bottom grid. The grids are a zirconium alloy except for the Standard fuel which used all Inconel grids.

The fuel pellet is dished and chamfered. The fuel batch stack density (density of the pellet reduced by the dishing and chamfering and takes no credit for annular pellet blankets) has ranged from []^{a,c} of the UO₂ theoretical density. Figure 7.4 shows that the maximum assembly stack density is []^{a,c}. Two standard deviation variation of fuel assembly stack density within a batch is 0.5%. Manufacturing tolerances for the fuel are found on Table 3.2. MPS3 fuel has had reduced enrichment axial blankets since Cycle 3 and annular pellet axial blankets since Cycle 5. Axial blanket length is nominally 6 inches on each end of the fuel pins. The annular pellets have the same outer diameter as standard pellets but have a void center with a diameter of 0.155 inches.

The active length of the fuel is 144 inches []^{a,c}. The distance from the bottom of the fuel assembly to the bottom of the active fuel has varied by design from []

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 25 of 300

^{a,c}. Axial blanket enrichments of 0.74 wt% (Batches 6 and 7) and 2.6 wt% have been used. All fuel pins in an assembly have the same central zone enrichment and the same blanket enrichment.

The assembly pitch in the core is 8.466 inches (21.50 cm). The location of the guide tubes in the assembly is given on Figure 3.1. Table 3.3 lists the fuel type and enrichment of each fuel Batch.

Table 3.2: Fuel Design Tolerances

Pellet Diameter	[] ^{a,c}
Clad Inner Diameter	[] ^{a,c}
Clad Outer Diameter	[] ^{a,c}
Rod Pitch	[] ^{a,c}
Guide Tube and Instrument Tube Inner Diameter	[] ^{a,c}
Guide Tube and Instrument Tube Outer Diameter	[] ^{a,c}
Stack Density	[] ^{a,c}

Figure 3.1: 17x17 Fuel Design

F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	F	F	G	F	F	G	F	F	G	F	F	F	F	F
F	F	F	G	F	F	F	F	F	F	F	F	F	G	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	G	F	F	G	F	F	G	F	F	G	F	F	G	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	G	F	F	G	F	F	I	F	F	G	F	F	G	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	G	F	F	G	F	F	G	F	F	G	F	F	G	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	G	F	F	F	F	F	F	F	F	F	G	F	F	F
F	F	F	F	F	G	F	F	G	F	F	G	F	F	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F
F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F	F

(F = Fuel, G = Guide Tube, I = Instrumentation Tube)

Table 3.3: Feed Fuel Design and Enrichment History for MPS3

Cycle	Assembly Type	Enrichment (wt% U-235)	Blanket Enrichment
1	Standard	2.42, 2.9, 3.4	N/A
2	Standard	3.5, 3.8	N/A
3	Standard	4.1, 4.5	0.74
4	V5H	4.2, 4.5	0.74
5	V5H	4.4	0.74
6	V5H	4.6	0.74
7	RFA	4.4, 4.8	2.6
8	RFA	4.4, 4.8	2.6
9	RFA	4.2, 4.7	2.6
10	RFA-2, NGF	4.7, 4.95	2.6
11	RFA-2	4.0, 4.95	2.6
12	RFA-2	4.7, 4.95	2.6
13	RFA-2	4.1, 4.9	2.6
14	RFA-2	4.1, 4.95	2.6
15	RFA-2	4.1, 4.95	2.6
16	RFA-2	4.1, 4.95	2.6
17	RFA-2	4.1, 4.95	2.6

3.2 Fuel Inserts Description

3.2.1 Burnable Absorbers

Two types of burnable poison have been used at MPS3. Pyrex BP was used in Cycles 1 and 2. Integral Fuel Burnable Absorber (IFBA) has been used for the remaining cycles. WABA will also be considered in this analysis to allow possible future use of WABA.

Pyrex burnable absorbers consist of an annulus of borosilicate glass with an air filled central hole and an inner and outer SS clad. The WABA design is also annular, but has RCS water in the inner hole and Zircaloy-4 clad. Pyrex and WABA assemblies are composed of rods inserted in the fuel assembly guide tubes suspended from a baseplate that rests on the fuel assembly top nozzle. The rods are often referred to as "fingers." There are up to 24 fingers in a Pyrex or WABA assembly. Table 3.4 provides the information needed to model Pyrex and WABA burnable absorbers.

An IFBA rod is a fuel rod with a thin ZrB_2 coating on the surface of some of the fuel pellets. The pellets at the top and bottom 6 inches of an IFBA rod are annular. MPS3 fuel assemblies with IFBA have contained 32 to 128 IFBA rods. Table 3.4 provides IFBA modeling data. BP and IFBA have not been used in the same fuel assembly at MPS3.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 28 of 300

Table 3.4: Description of Burnable Poisons

Dimension	Pyrex BP	IFBA	WABA
Cycles Used	1 & 2	3 -18	None
Inner Clad ID (cm)	0.428	N/A	0.572
Inner Clad OD (cm)	0.460	N/A	0.6782
BP ID (cm)	0.483	0.8192	0.706
BP OD (cm)	0.853	0.8198	0.808
Outer Clad ID (cm)	0.874	N/A	0.8357
Outer Clad OD (cm)	0.9677	N/A	0.968
Clad Material	SS-304	N/A	Zirc-4
Poison Loading (mg B-10/cm)	[] ^{a,c} *	[] ^{a,c} [] ^{a,c}	[] ^{a,c}
BP Material	B ₂ O ₃ in Pyrex glass	ZrB ₂	B ₄ C in Al ₂ O ₃
BP Material Density (g/cc)	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}
Poison axial length (in)	144**	108 (Batch 5) 120 (Batch 6-16) 122 (Batch 17+)	144**
Poison lower bound axial location above bottom of fuel stack (in.)	0**	18 (Batch 5) 12 (Batch 6-16) 11 (Batch 17+)	0**

* Calculated using a B₂O₃ loading of 12.5 w/o and natural B-10/B isotope abundance.

**Conservatively assumed to be full length for fuel depletion. WABA has not been used at MPS3. North Anna WABA has 128 inches of absorber (~8 inches of un-poisoned fuel at top and bottom)

3.2.2 Control Rods

There have been three control rod designs used in MPS3. Table 3.5 summarizes the relevant design features. The first generation of MPS3 control rods were Westinghouse Hf-Zr type. A transition to the second generation (Westinghouse Ag-In-Cd) began in Cycle 3. A transition to the third generation of control rods (AREVA Ag-In-Cd) began in Cycle 13.

A control rod assembly has 24 fingers. When fully inserted, the control rod neutron absorber extends above the top of the active fuel, but can leave a maximum of 5.25 inches of the bottom of the active fuel un-poisoned. At the bottom of the control rod is a SS end plug.

Table 3.5 Description of Control Rods

Component	Design Characteristic*
Control Rod	
Absorber Diameter	0.866 cm **
Clad Inner Diameter	0.874 cm
Clad Outer Diameter	0.968 cm
Absorber Material (Cycles 1-4)	Hf (95.3 – 95.4 w/o)-Zr(4.5 w/o)
Absorber Material (Cycle 5 - present)	Ag(80 w/o)- In(15 w/o)- Cd(5 w/o)
Maximum Distance from the bottom end of the absorber to the bottom end of the active fuel	5.25 inches

*Material and dimensional tolerances are not provided because of excess margin in control rod credit cases.

**Lower 12 inches of absorber in the AREVA design has a slightly smaller absorber diameter.

3.2.3 Sources and Other Inserts

Primary source rods (a single rod displacing a Pyrex BP rod in a 24 BP assembly) and secondary source rods (4 rod assembly) were used in Cycle 1. Because source rods do not contain a strong absorber material the impact of a source rod is less than a BPRA rod.

Secondary source assemblies (SSA) have been used throughout MPS3's history, though not in every cycle. SSA with 4 pins were used in assemblies concurrently with 6 finger Pyrex BP assemblies during Cycle 2. SSA with 4 pins and with 6 pins have been used in combination with IFBA. Secondary source pins consist of SS clad containing Sb-Be source material and inert spacers and can be modeled during fuel depletion as solid SS rods with an outer diameter of 0.381 inches. SSA are weaker absorbers and displace less water than 24 finger WABA. Therefore, depletion effects of 24 finger WABA bound those of SSA alone. Depletion effects of SSA and IFBA in the same fuel assembly are evaluated in the determination of the bounding fuel assembly depletion conditions.

Prior to operation, thimbles for the in-core flux monitoring are inserted into the instrument tubes of about one third of the assemblies. Since these thimbles displace water they have a small effect on fuel depletion. The dimensions of the thimbles are given in Table 3.6.

Table 3.6: Description of In-core Thimble

In-Core Instrument Thimble	
Thimble Inner Diameter	0.51 cm
Thimble Outer Diameter	0.76 cm
Thimble Material	Stainless Steel 316
Fraction of assemblies with in-core thimble	30% (58 of 193)

3.3 *Non-Standard Fuel in the Pool*

There are a few items in the spent fuel pool that are not standard fuel assemblies. If the item does not have fuel in it, it can be placed in any cell that is allowed to have fuel. The following items in the MPS3 SFP contain fuel but are not standard fuel assemblies:

- Fuel rod storage container (FRSC)
- Reconstituted fuel assemblies
 - With SS replacement pins
 - Without SS replacement pins (MR71 has two open lattice locations)

The key design features of the FRSC are:

- Square lattice
- 0.937 inch storage tube pitch
- 52 SS fuel rod storage tubes
- Storage tube OD 0.625 inch (48 tubes) and 0.750 inch (4 tubes)
- Storage tube wall thickness 0.035 inch (48 tubes) and 0.049 inch (4 tubes)

The storage canister currently contains one rod from fuel assembly G81, two from L74, and three from M71. After reconstitution, assembly M71 was renamed MR71. Removed rods were replaced with stainless steel pins except in MR71, which has two remaining empty fuel rod locations. Missing rods in MR71 are locations H01 and J01 (peripheral locations near the middle of one face of the assembly). Figure 3.2 is a schematic view of the FRSC.

Figure 3.2: Fuel Rod Storage Container Schematic

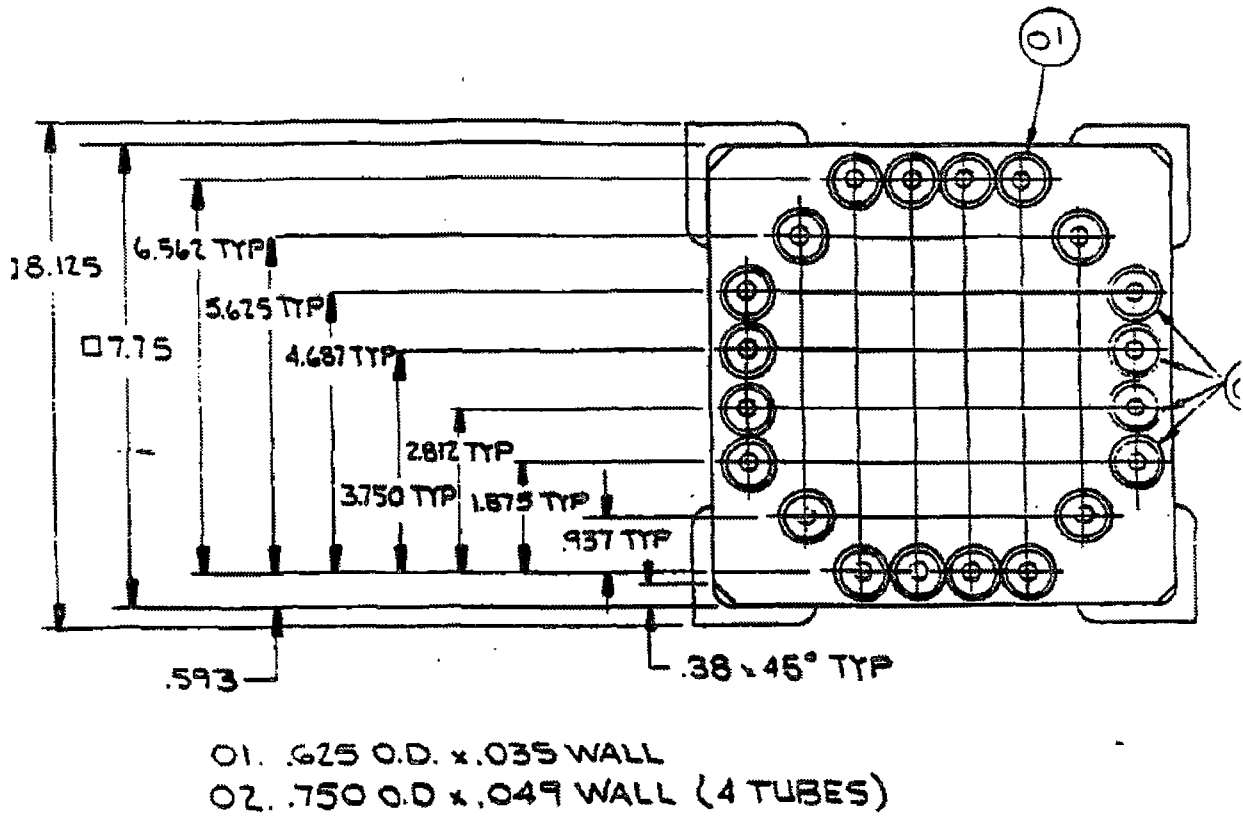
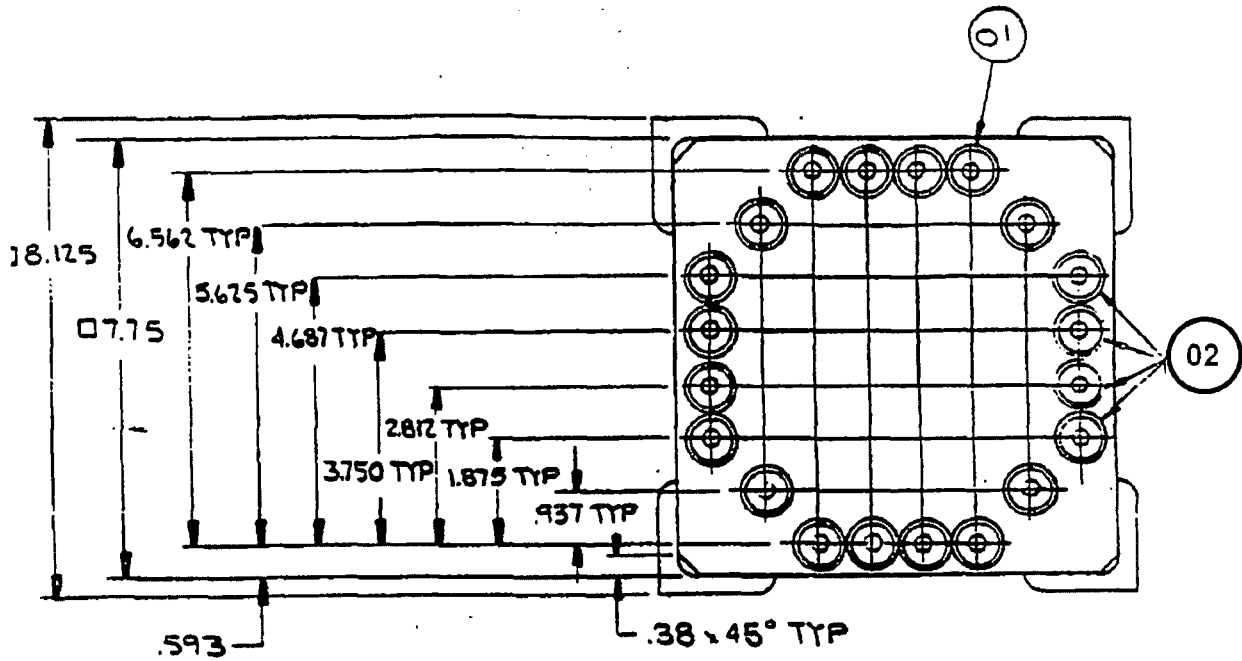


Figure 3.2: Fuel Rod Storage Container Schematic



- 01. .625 O.D. x .035 WALL
- 02. .750 O.D x .049 WALL (4 TUBES)
- 01. .625 O.D. x .035 WALL
- 02. .750 O.D x .049 WALL (4 TUBES)

4 Storage Rack Description

4.1 New Fuel Storage Area

The New Fuel Storage Rack (NFSR) is designed with four 6x4 storage cell rack modules. BORAL neutron absorber panels are aligned parallel in a south to north orientation and are in between both the 2nd and 3rd cells and 4th and 5th cells of each rack module. The BORAL has a minimum B-10 areal density of 0.005 g/cm^2 . The X-Y dimensions of the NFSR modules are shown below in Table 4.1. Figure 4.1 gives an overview of the area. Rack cells are depicted in the SE corner of the SW module inside the circle.

Figure 4.1: MPS3 New Fuel Storage Area

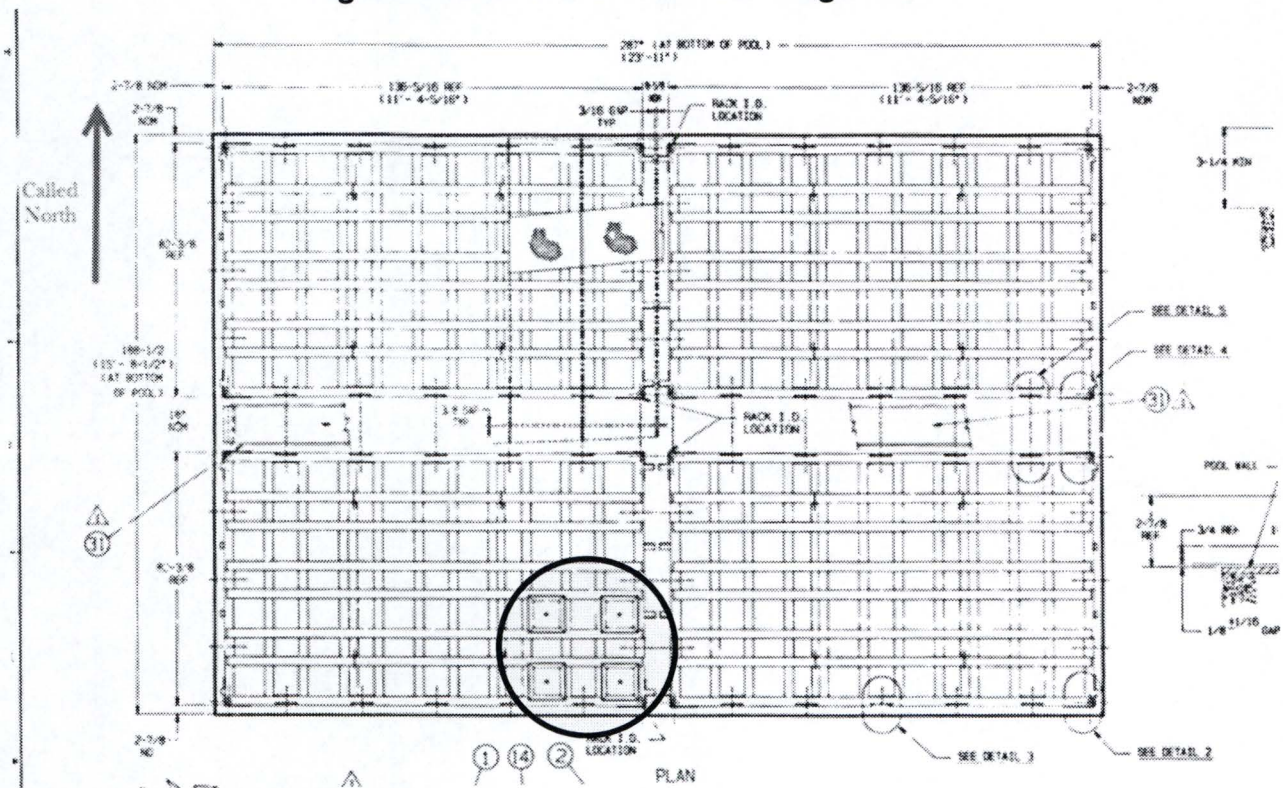


Table 4.1: New Fuel Storage Rack Dimensions

Measurement Description	Dimension (inches)	Uncertainty (inches)
Cell ID	8 15/16	[] ^{a,c}
Cell Wall Thickness	0.090	[] ^{a,c}
Cell Pitch, North -South	22 1/8	[] ^{a,c}
Cell Pitch, East-West	24 1/16	[] ^{a,c}
Gap Between Cell Center and Rack Module Edge	8 1/4	[] ^{a,c}
Gap Between Rack Modules, North-South	18	[] ^{a,c}
Gap Between Rack Modules, East-West	8 5/8	[] ^{a,c}
BORAL Sheet Thickness	0.080	N/A ¹
BORAL Sheet Width	76	[] ^{a,c}
SS Wall Liner Thickness	1/4	[] ^{a,c}

¹BORAL thickness is unimportant because it does not affect B-10 areal density. Thickness variation of the very thin BORAL panels would very slightly change the thickness of the moderator between the fuel cell and the BORAL. For full flooding, the moderator region thickness (~ 7 inches on each side) is too large for a slight change to have an impact. For a low density moderator, the effect of a slight change in moderator thickness would be offset by a change to the optimum moderator density.

The axial dimensions of the New Fuel Storage Racks are shown below in Table 4.2. Figure 4.2 gives an overview of the area. Figure 4.2 does not show the BORAL sheets positions between the cells, but it does appear the sheets are centered between the 2nd and 3rd assembly cell and between the 4th and 5th assembly cell. A sensitivity case in Section 7.4 (Table 7.8 "BORAL Radial Position") with BORAL sheets off center by 1" decreased reactivity. Therefore, centering the BORAL sheets between storage cells is conservative.

There is no BORAL monitoring program for the NFSR because normal operation of these racks is dry with no fuel or with fresh fuel. The BORAL is not subjected to radiation, heat, or a corrosive environment. There is no physical interaction of the BORAL with any other objects during fuel storage activities or when cells are empty.

Figure 4.2: Axial Drawing of a MPS3 New Fuel Storage Rack Module

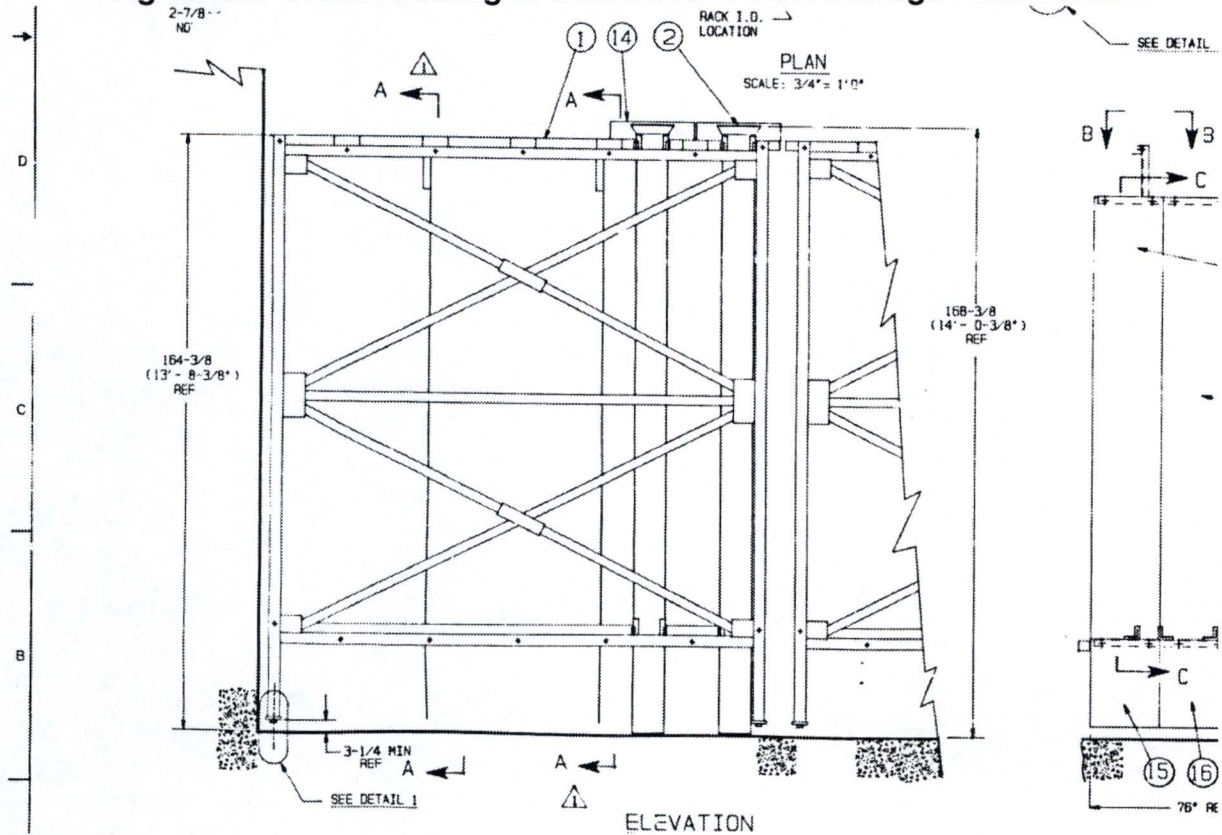


Table 4.2: Axial Heights of the New Fuel Storage Area KENO Model (Inches)

Parameter	Height ¹	Uncertainty
Bottom of SS Pool Floor	-1/4	[] ^{a,c}
Bottom of Cell Base Plate	0	[] ^{a,c}
Bottom of Fuel Assembly	3/8	N/A
Bottom of BORAL	2 7/8	N/A
Bottom of Active Fuel ²	3.278	N/A
Top of Active Fuel	147.278	N/A
Top of BORAL	148 7/8	N/A
Top of Cell	168 1/8	N/A

¹All heights are relative to the bottom of the cell base plate.

²Varies from 3.28 to 3.97 inch depending on design.

4.2 *Spent Fuel Pool*

There are three rack designs in the MPS3 SFP grouped into three areas designated Region 1, Region 2, and Region 3. Figure 4.3 shows the location of each Region in the SFP. Region 2 rack module A5 has not been installed but is included in this analysis.

Figure 4.3: MPS3 SFP Rack Orientation



4.2.1 Region 1

Region 1 racks are BORAL flux trap design. Each storage cell consists of a square stainless steel tube with a BORAL neutron absorber panel enclosed inside a stainless steel wrapper on the outer side of each of the four faces of the cell. There are 5 Region 1 7x10 racks with 350 total storage cells. Figure 4.4 shows the rack geometry in the X-Y plane. The storage cell pitch is larger in the Y dimension than the X dimension. Figure 4.5 shows the rack geometry in the X-Z plane.

The rack design in the X direction (the direction of adjacent Region 1 racks) provides more flux trap space between adjacent storage cells (1.5 inch) than the nominal interior cell spacing (1.0 inch nominal). This eliminates the need to consider the effect of thicker BORAL wrappers on the outer cells which could otherwise modestly reduce the flux trap water region between racks (0.1 inch). An infinite lattice model conservatively represents the actual installed Region 1 rack spacing. Further, minimum rack-to-rack spacing is limited by the protrusion of the rack baseplate beyond the storage cell envelope. Therefore, seismic activity cannot result in Region 1 storage cells being closer together than in the infinite lattice model. Region 1 rack dimensions and tolerances are given in Table 4.3. Table 4.4 provides the axial alignment of key parts of the Region 1 racks. Fuel stack alignment is based on the Westinghouse RFA/NGF fuel design, however, the 148 inch BORAL envelopes the fuel stack in all MPS3 fuel designs.

A BORAL coupon monitoring program is used to confirm that the material condition of the BORAL in service remains bounded by the criticality analysis. The monitoring program description has been provided to the NRC [5,6]. Because Regions 1 and 2 racks are of the same age, manufacturer, and BORAL specification, the MPS3 coupon program includes both Regions. Region 1 and 2 analyses use the minimum B-10 areal density ($[\quad]^H$ Ref. 5). Minimum as-built BORAL coupon areal density (average of measurements taken at several areas of the coupon) is $[\quad]^H$.

Figure 4.4: MPS3 Region 1 Rack Dimensions (X-Y)



Figure 4.5: MPS3 Region 1 Rack Dimensions (X-Z)

H

Table 4.3: Region 1 Rack Dimensions and Tolerances

Item	Description	Dimension	Tolerance
1	BORAL length	148 in. min.	[] ^H
2	BORAL width	[] ^H	[] ^H
3	BORAL thickness	[] ^H	[] ^H
4	BORAL position above top of rack baseplate	[] ^H	[] ^H
5	Minimum as-built BORAL B-10 AD (coupons)	[] ^H	[] ^H
6	Modeled BORAL B-10 AD	[] ^H	N/A***
7	BORAL sheathing thickness	[] ^H	*
8	Storage can height above top of baseplate	[] ^H	[] ^H
9	Baseplate thickness	0.75 in.	N/A
10	Foot height	[] ^H	[] ^H
11	Cell ID	8.80 in.	[] ^H
12	Cell pitch "y"	10.455 in.	[] ^H
13	Cell pitch "x"	10.0 in.	[] ^H
14	Cell wall thickness	[] ^H	[] ^H
15	Inner wrapper thickness	[] ^H	[] ^H
16	Outer wrapper thickness	[] ^H	[] ^H
17	Wrapper space	[] ^H	[] ^H
18	Flux trap width "y"***	[] ^H	[] ^H
19	Flux trap width "x"***	[] ^H	[] ^H
20	Rack gap	[] ^H	[] ^H
21	Fuel Stack position above top of rack baseplate	[] ^H	[] ^H
22	Fuel stack height	144 in. nominal	[] ^{a,c}
23	BORAL extent above fuel	[] ^H	[] ^H
24	BORAL extent below fuel	[] ^H	[] ^H

*No tolerance available. Minimum tolerance is bounded by replacing sheathing on one side with water. Maximum tolerance is bounded by replacing water in the wrapper space with aluminum.

**The flux trap tolerance is equal to the sum of the pitch and cell ID tolerances and is captured in the modeling of the individual components.

***Conservative low value. See Generic Letter responses for more details (Ref. 5 and 6)

Table 4.4: Region 1 Rack Axial Alignment

Item	Axial Position (in.)
Top of rack baseplate	[] ^H
Bottom of BORAL	[] ^H
Bottom of fuel stack	[] ^H
Top of fuel stack	144.00
Top of BORAL	[] ^H
Top of rack cell	[] ^H

4.2.2 Region 2

Region 2 racks are BORAL non-flux trap design. There are two types of storage cells in the Region 2 racks. Every other cell is an actual cell (a square stainless steel tube). Actual cells are welded together at the cell corners to form a checkerboard arrangement of actual and resultant cells. Resultant cells are formed by the checkerboard spaces between actual cells. BORAL neutron absorber panels are enclosed on the outer faces of each actual storage cell by a thin stainless steel wrapper. Stainless steel plates with enclosed BORAL are used to enclose the open side of resultant cells on each rack face.

There are 10 Region 2 racks with 754 total storage cells (including the uninstalled rack module). Table 4.5 shows the Region 2 rack sizes. Figure 4.6 shows the rack geometry in the X-Y plane. Figure 4.7 shows the rack geometry in the X-Z plane.

The rack design includes a baseplate that extends a minimum of []^H beyond the storage cell envelope. Storage cells at the edge of each rack that adjoins another Region 2 rack create a flux trap with the two exterior BORAL panels separated by the rack-to-rack spacing. Nominal installed rack-to-rack spacing is 1 inch. The flux trap water gap cannot be less than allowed by baseplate contact []^H, which also results in an edge assembly cell pitch at the rack-to-rack boundary greater than the interior rack cell pitch. This eliminates the need to consider the effect of thicker BORAL wrappers on the outer cells and ensures that an infinite lattice model conservatively represents the actual installed Region 2 racks. It also ensures a seismic event cannot result in Region 2 storage cells being closer together than in the infinite lattice model. Region 2 rack dimensions and tolerances are given in Table 4.6. Table 4.7 provides the axial alignment of key parts of the Region 2 racks. Fuel stack alignment is based on the

Westinghouse RFA/NGF fuel design, however, the 148 inch BORAL envelopes the fuel stack in all MPS3 fuel designs.

Table 4.5: Region 2 Racks

Number of racks	Rack size	Rack ID
5	9x9	A1-A5
1	9x10	B3
1	7x10	C1
3	7x9	E1-E3

Figure 4.6: MPS3 Region 2 Rack Dimensions (X-Y)

H

Figure 4.7: MPS3 Region 2 Rack Dimensions (X-Z)



Table 4.6: Region 2 Rack Dimensions and Tolerances

Item	Description	Value	Tolerance
1	BORAL length	148 in.	[] ^H
2	BORAL width	[] ^H	[] ^H
3	BORAL thickness	[] ^H	[] ^H
4	BORAL position above top of baseplate	[] ^H	[] ^H
5	Minimum as-built BORAL B-10 AD (coupons)	[] ^H	[] ^H
6	Modeled BORAL B-10 AD	[] ^H	N/A**
7	BORAL clad thickness	[] ^H	[] ^H *
8	Storage can height above top of baseplate	[] ^H	[] ^H
9	Baseplate	0.75 in.	N/A
10	Foot	[] ^H	[] ^H
11	Cell ID	8.80 in.	[] ^H
12	Cell pitch	9.017 in.	[] ^H
13	Cell wall thickness	[] ^H	[] ^H
14	Inner wrapper thickness	[] ^H	[] ^H
15	Outer wrapper thickness	[] ^H	[] ^H
16	Wrapper space (between inside of wrapper and outside of storage cell)	[] ^H	[] ^H

*Based on filling the wrapper space with thicker than nominal cladding

**Conservative low value. See Generic Letter responses for more details (Ref. 5 and 6)

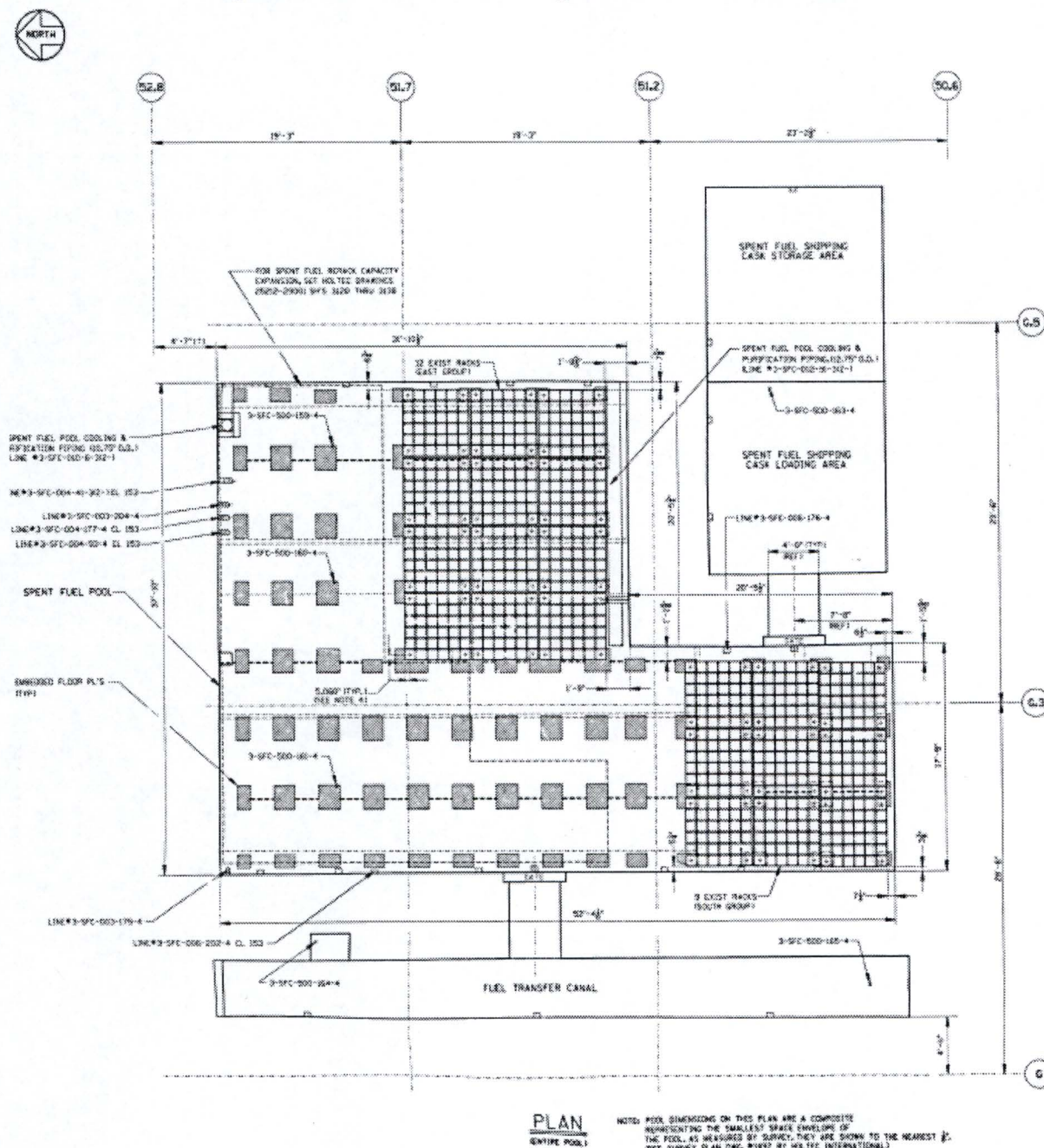
Table 4.7: Region 2 Rack Axial Alignment

Item	Axial Position (in.)
Top of rack baseplate	[] ^H
Bottom of BORAL	[] ^H
Bottom of fuel stack	[] ^H
Top of fuel stack	144.00
Top of BORAL	[] ^H
Top of rack cell	[] ^H

4.2.3 Region 3

Region 3 racks are Boraflex flux trap design. Each Region 3 rack module is made up of a 6x6 array of cells surrounded by a quarter inch stainless steel panel (referred to as a rack wall or rack shear wall). Twenty one racks with a total of 756 fuel storage locations are installed in the SFP. Figure 4.8 shows the orientation of the Region 3 racks in the SFP.

Figure 4.8: Location of Region 3 Racks in the SFP



There are Boraflex panels with wrappers on all cell walls that face another cell. No credit is taken for the Boraflex, which is modeled as water. Figure 4.9 is an X-Y plane drawing of a Region 3 cell. Figure 4.10 shows the X-Y profile of the Boraflex wrapper. Rack dimensions are shown in Table 4.8. Figure 4.11 is an axial drawing of the Region 3 spent fuel racks. Table 4.9 gives the KENO dimensions for the axial model.

Figure 4.9: X-Y Drawing of a Region 3 Rack Cell

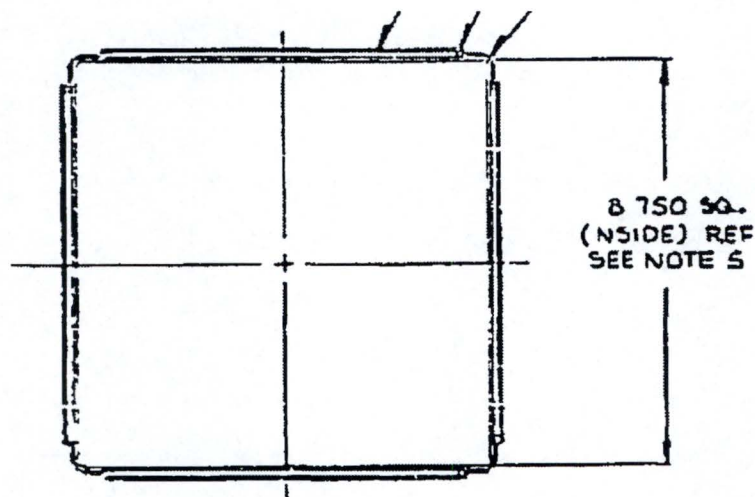


Figure 4.10: X-Y Drawing of a Region 3 Rack Wrapper

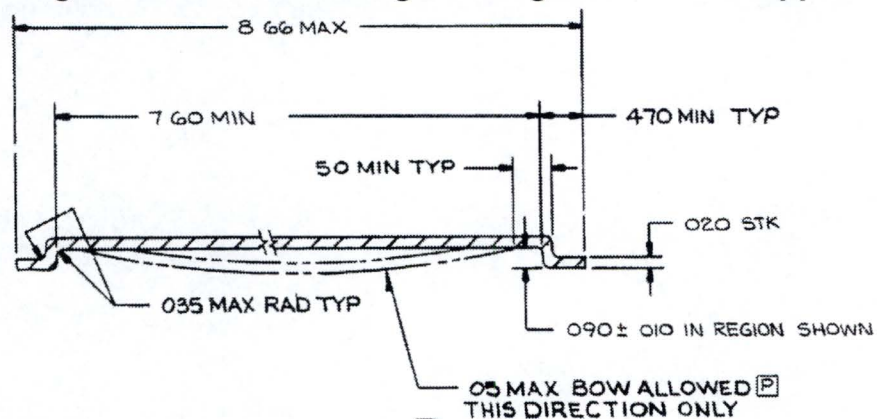


Table 4.8: Region 3 Rack Model X-Y Dimensions (cm)

Parameter	Dimension	Tolerance
Cell ID	22.2250	[] ^{a,c}
Cell Wall Thickness	0.1524	[] ^{a,c}
Wrapper Thickness	0.0508	[] ^{a,c}
Raised Wrapper Width	19.4056	N/A ¹
Flush Wrapper Width	2 x 1.1430	N/A ¹
Wrapper Channel Thickness	0.2286	[] ^{a,c 5}
Wrapper Channel Width	19.3040	N/A ¹
Cell Pitch	26.2890	[] ^{a,c}
Rack Wall (aka Shear Panel) Thickness	0.635	[] ^{a,c 2}
Cell Wall to Rack Wall Separation	0.7747 ³	N/A ³
Rack to Rack Separation	2.54	[] ^{a,c 4}

¹The tail portion of the wrapper is not modeled in KENO. Wrapper thickness cases show less wrapper material is bounding.

²No tolerance available. Conservative upper bound value for quarter inch thick, hot rolled steel plate is used.

³ Separation between cells at the rack module edge is important not the precise placement of the rack wall in the area between cells.

⁴Minimum installed rack separation modeled. Uncertainty evaluated to address seismic event.

⁵The wrapper channel thickness tolerance slightly affects the amount of water in the flux trap region of the rack, but that is unimportant to this analysis since no Boraflex is credited. Therefore, no tolerance case will be run for the wrapper channel thickness.

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 48 of 300

Figure 4.11: Axial Drawing of a Region 3 Rack Cell

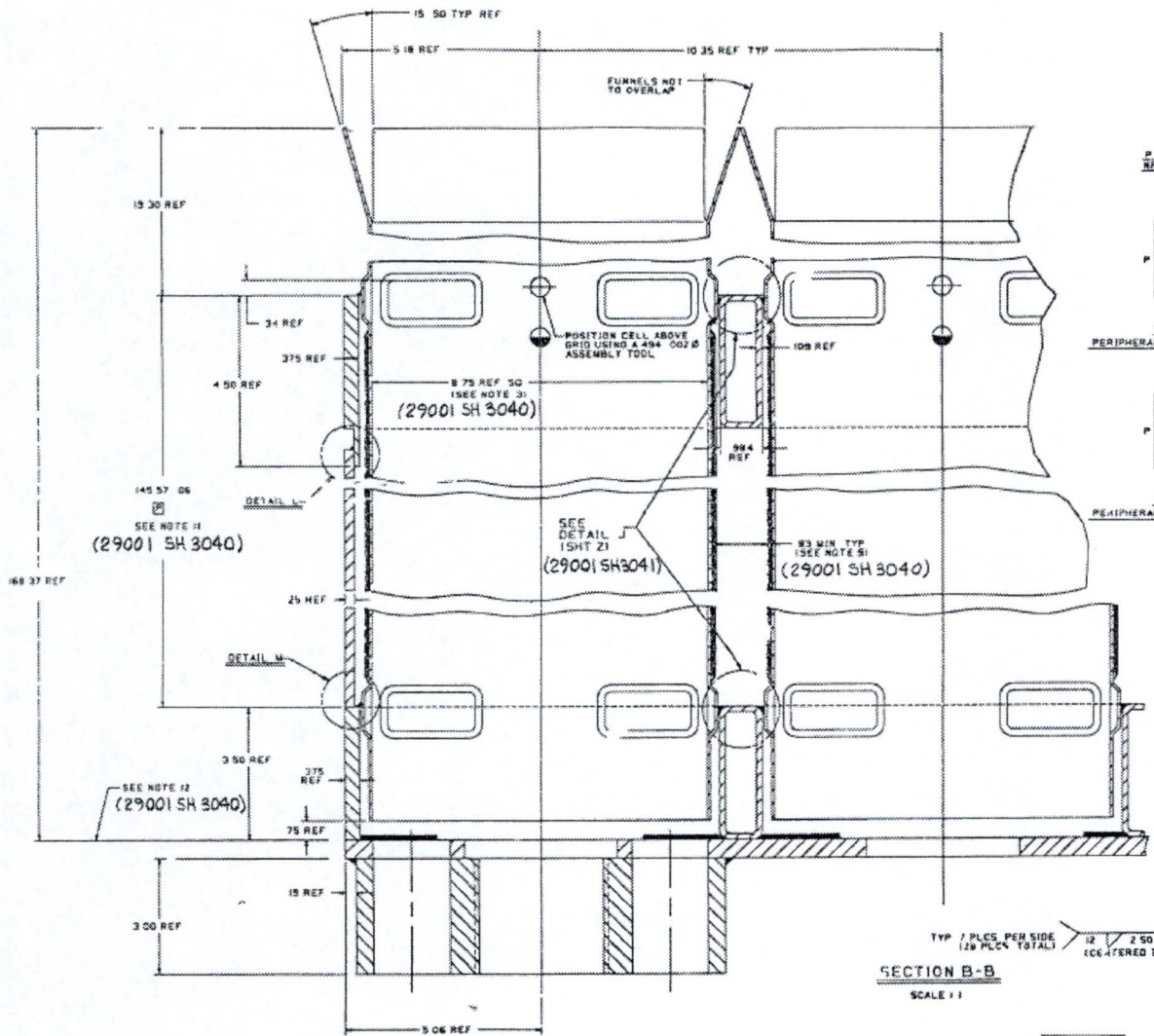


Table 4.9: Region 3 Rack Model Axial Dimensions

Parameter	Dimension ² (inches)
Bottom of Fuel Assembly and Rack Shear Wall	0
Bottom of Cell Walls	0.75
Bottom of Active Fuel	3.18 ¹
Bottom of Wrapper	5.25
Top of Active Fuel	147.18 ³
Top of Wrapper	147.37
Top of Rack Shear Wall	149.07

¹There were several different fuel assembly designs used at MPS3. Each fuel assembly design has the fuel starting at slightly different elevations. The base case uses an unrealistically short elevation and the "fuel position" cases tested the highest elevation. The delta is incorporated as the "Fuel Axial Position" bias.

²There are no uncertainties for the axial model except for the stack length. This is because "Fuel Axial Position" bias makes the model bounding.

³All fuel assembly designs have the same nominal stack length. See Note 1.

5 Overview of the Method of Analysis

5.1 *New Fuel Storage Area*

The criticality analysis of the New Fuel Storage Area (NFSA) is performed assuming two accident conditions, fully flooded with water and fully flooded with low density water (optimum moderation). Each storage cell of the NFSA is modeled with the most reactive fuel design at the highest enrichment (5 wt% U-235). No credit is taken for burnable absorbers. The analysis considers both centered and asymmetric placement of the assemblies in the storage cells. The analysis models the full NFSA with boundaries extended sufficiently to justify use of zero flux boundary conditions. A range of moderator temperatures from 32°F to 150°F is covered. The concrete composition, which is important to the optimum moderation cases, is conservatively determined.

A single composite fuel design is created from the range of historical fuel design features to bound new fuel storage reactivity for current and anticipated future fuel designs.

Reduced enrichment and annular pellet axial blankets are not modeled. Using higher than actual enrichment in the blanket region is conservative. The stack fuel density (combination of density, dish, and chamfer) in the model conservatively bounds the highest expected stack density for MPS3. New fuel is conservatively assumed to have no U-234 or U-236.

The analysis uses SCALE 6.0 and the ENDF/B-VII cross section library. [7] Criticality calculations are done with the CSAS5 module (KENO-V.a). [8]

Determination of the 95/95 k-eff includes uncertainties for manufacturing tolerances of both the fuel and rack calculated at both the full and optimum moderation. In order to cover the variation in zirconium alloys, pure zirconium is used in the analysis because the alloying elements reduce reactivity.

5.2 Spent Fuel Storage Racks

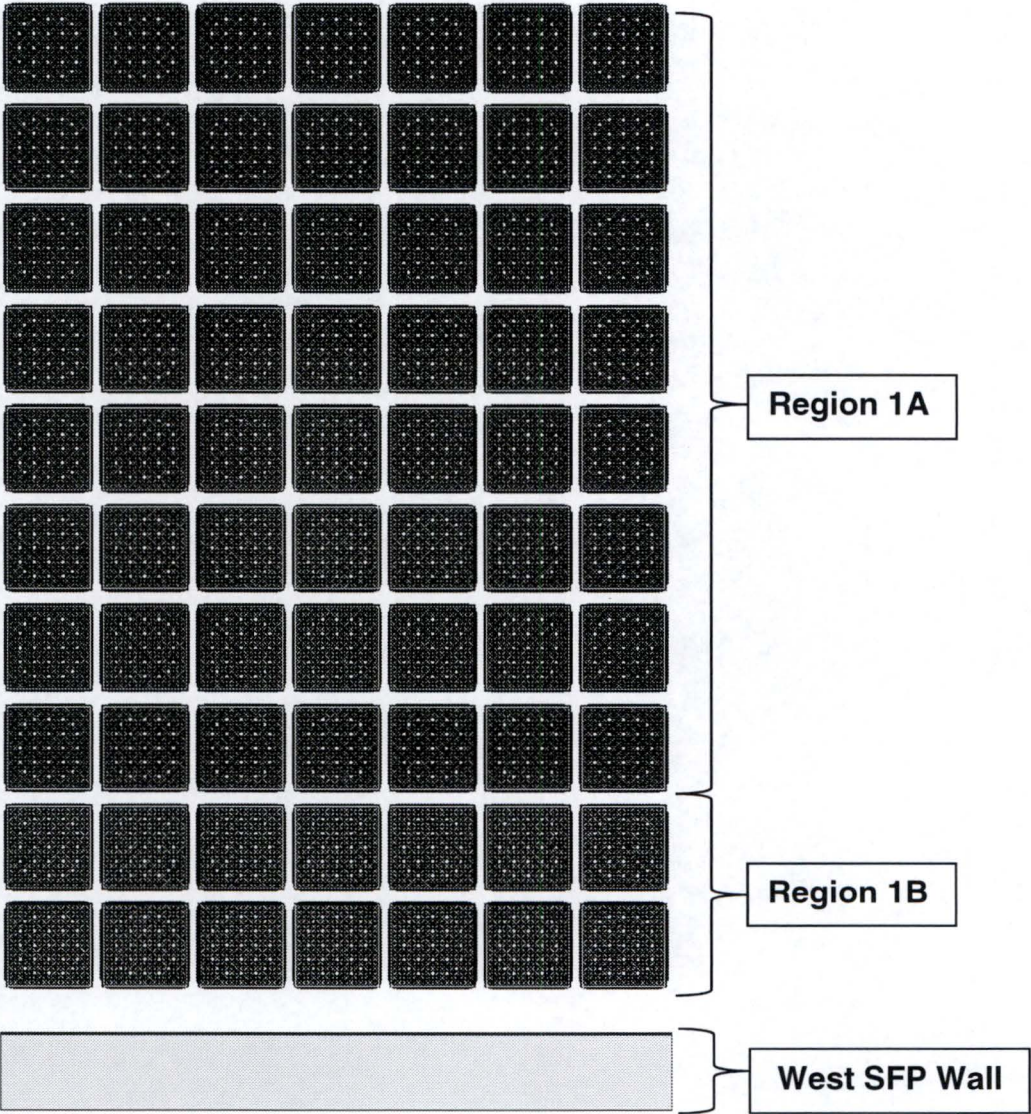
5.2.1 Storage Geometry

Storage geometries for each Region of the SFP are listed in Table 5.1. Region 1 is composed of subregions 1A and 1B. Region 1B occupies the 2 rows of storage cells in each Region 1 rack module closest to the West SFP wall. Figure 5.1 shows the Region 1A/1B configuration which credits neutron leakage at the SFP wall. Asymmetric positioning of fuel in the storage cells is calculated as a bias.

Table 5.1 – Storage Geometries

Region	Geometry	Credits	Description
1A	4 out of 4	None	≤ 4.75 wt% U-235
1A	4 out of 4	≥ 12 IFBA	≤ 5.00 wt% U-235
1A	4 out of 4	Burnup	≥ 2 GWD/MTU, ≤ 5.00 wt% U-235
1B	4 out of 4	SFP wall	≤ 5.00 wt% U-235, Region limit is \leq two rows of rack cells nearest the West SFP wall.
2	4 out of 4	Burnup	Burnup curve, Table 13.5
2	4 out of 4	Control rod	No burnup credit required for fuel enrichment ≤ 5.0 wt% U-235
3	4 out of 4	Burnup, decay time	One burnup curve for each minimum decay time (0, 3, 9, 18, and 25 years), Table 13.5

Figure 5.1: KENO Region 1A/1B Wall Credit Model



5.2.2 Bounding Fuel Design

A single composite fuel design is created from the range of historical fuel design features to bound SFP reactivity for historical and anticipated future fuel designs. Selection of a bounding design feature such as grid volume is different for depletion analysis and SFP k-eff calculations. Maximum grid volume is used in the depletion analysis. SFP rack criticality calculations use minimum grid volume and a bias is added if sensitivity cases indicate maximum grid volume produced higher k-eff.

Four fuel designs have been used at MPS3. Other than small changes in axial fuel stack position, these are the features that vary among these designs:

- Guide tube / instrument tube diameter (all similar)
- Clad alloy (all are Zr based)
- Grid material (3 designs are Zr based, one has Inconel in the fuel region)
- Grid volume (variation in grid size and number of grids)
- Fuel loading (combination of density, dish, and chamfer)
- Reduced enrichment axial blankets (some with annular pellets)
- Burnable Poison Types (Pyrex or IFBA)

Inconel grids are significant neutron absorbers and do not need to be considered for the SFP rack model. Clad alloys and grid materials are represented in the KENO rack model as Zr, since the alloying materials are small quantities and are stronger neutron absorbers than Zr.

Grid volume is maximized in TRITON depletions (a harder neutron spectrum produces higher reactivity depleted fuel isotopic content) and minimized in the SFP rack models.

Higher fuel density increases k-eff (verified in rack model tolerance cases), so a bounding high value is used. The larger of two guide tube options (greater water displacement) is used in the TRITON model and the smaller is used in the KENO rack model. Reduced enrichment and annular pellet axial blankets are not modeled except as part of a justification argument. Using higher than actual enrichment in the blanket region is conservative.

TRITON depletions are performed using the maximum WABA loading (24 fingers), bounding fuel density, bounding high power and temperatures, and instrument thimble water displacement by an incore detector thimble. Depletion with 24 WABA bounds depletion with maximum IFBA. WABA has not been used in MPS3, but performing depletions with WABA inserted for the entire burnup range will allow for potential future WABA use. MPS3 burnable poison configurations not bounded by 24 WABA are identified and justified.

Nominal fuel assembly dimensions are used in the TRITON depletions. Clad creep and grid expansion with depletion are neglected in the depletion because both increase the amount of water in the fuel lattice, soften the neutron spectrum during depletion, reduce Pu production, and reduce fuel reactivity in the SFP. Clad creep and grid expansion effects are included as a bias in the SFP k-eff calculations. Grid expansion used in the analysis is based on modern clad material (ZIRLO). Older fuel with Zircaloy-4 grids that potentially have more grid expansion are identified and justified.

5.2.3 Soluble Boron Credit

The spent fuel storage rack analysis for Regions 1, 2, and 3 credits soluble boron. To support use of boron credit, a boron dilution analysis is provided in Attachment 7 that initiates below the Technical Specifications minimum allowable normal soluble boron concentration (2600 ppm in proposed TS). The boron dilution analysis confirms that the time needed for a dilution event to reduce the soluble boron concentration to the minimum acceptable level (600 ppm, a level at which the criticality analysis normal operation k-eff remains less than 0.95) is greater than the time needed for actions to be taken to prevent further dilution. The boron dilution accident analysis confirms that operators have sufficient time to identify, diagnose and correct the cause of the inadvertent dilution, thereby preventing SFP k-eff from exceeding the regulatory limit.

5.2.4 Burnup Credit

Fuel burnup is credited in Regions 1, 2, and 3. To conservatively calculate the isotopic content of depleted fuel, these depletion variables are considered:

- Fuel and moderator temperature
- Power history
- Axial burnup profile
- Soluble boron history
- Burnable absorber
- Control rod insertion
- Fuel inserts

Bounding axial node depletion temperatures are determined in part by selecting a bounding high fuel assembly average power. The assembly average power selected bounds the power history (average lifetime depletion power) experienced by MPS3 fuel at each analyzed assembly average burnup. High assembly power results in high fuel and moderator

temperatures, which hardens the neutron spectrum, increases Pu production, and increases fuel reactivity in the SFP.

Axial burnup profiles from DOE database reviewed in NUREG/CR-6801 [9] are used within their respective burnup ranges. With fuel assembly power and the axial burnup profile known, the nodal moderator temperature is determined for each of the 18 axial nodes starting from the core inlet and integrating the enthalpy added in each successive node. Data used for the axial heat balance calculation includes bounding high core power, minimum historical measured RCS coolant flow, conservatively high core bypass flow, and conservatively high HFP RCS inlet temperature. These heat balance input values result in core average RCS outlet temperature 2.5 °F higher than any MPS3 historical cycle.

The bounding high assembly power produces a maximum assembly RCS exit temperature only 6.1 °F below saturation temperature. Due to the margin to prior cycle RCS exit temperatures and the proximity to saturation temperature, it is unlikely that a future cycle would operate with moderator temperatures in an actual fuel assembly higher than the analyzed bounding assembly.

Fuel temperatures are determined for each node by adding to the nodal moderator temperature a value for fuel temperature increase above local moderator temperature. The fuel temperature increase value is a function of fuel burnup and nodal power. Fuel temperature data is from the SIMULATE model used by Dominion for its licensed fuel management analysis [10]. Fuel temperatures are pellet average, which bound the resonance effective fuel temperatures.

Bounding high power history is selected to maximize fuel and moderator temperature during depletion. However, depleting at low power produces less Pm-149, which has the effect of reducing Sm-149 peaking after fuel discharge, increasing fuel reactivity in the SFP. High power history combined with reduced power near end of life (EOL) is used to capture both effects. Depletion is performed at constant high power conditions for all but the last 20 days of depletion. Depletion for 40 days at half the bounding power is performed at EOL instead of 20 days of depletion at the bounding power. Bounding high power level fuel and moderator temperatures are retained throughout the depletion for simplicity and conservatism.

High depletion soluble boron hardens the neutron spectrum, increases Pu production, and increases fuel reactivity in the SFP. A constant soluble boron that bounds the cycle average boron for all historical cycles (one exception) and expected future cycle average boron is used

in the TRITON depletions. All fuel depleted in the unbounded cycle were depleted in another cycle. The average boron concentration of any two cycles is lower than the boron concentration used in the TRITON depletions.

Depleted fuel content is determined for each of the 18 axial fuel nodes of a fuel assembly with independent TRITON depletions. All isotopes produced during depletion are used in the SFP rack k-eff calculations except for those of very low concentration (less than $1\text{E-}10$ at/b-cm). With no credit for fuel decay time, isotopic content decayed for 5 days after the at-power fuel depletion maximizes fuel reactivity. Isotopic changes due to decay time are calculated within the TRITON depletions. There is a post-TRITON reduction applied to some fission gases and volatile elements to account for the possibility of migration of these isotopes away from where they were produced.

Determination of bounding assembly depletion conditions includes consideration of neutron absorbers and fuel assembly inserts that displace water. Neutron spectrum hardening by absorbers and inserts increases Pu production during depletion and fuel reactivity in the SFP. Some inserts (WABA, control rods, source rods, etc) are mutually exclusive because they occupy guide tubes. The following absorbers and water displacers are evaluated:

- Pyrex BP (used in early cycles)
- IFBA (currently used)
- WABA (potential future use)
- Primary and secondary sources
- In-core detector thimble
- Control rod insertion (≤ 10 steps cycle average)

Depletion analysis assumes all fuel assemblies are depleted with 24 WABA rods and an in-core thimble inserted. WABA is depleted with the fuel and is retained throughout the entire depletion. With the exception of some early cycle assemblies in which Pyrex BP was used, this approach conservatively bounds the depletion history of all MPS3 fuel assemblies to date. In particular, BP are removed after one cycle, only about a third of the core has in-core thimbles, and only five out of 193 fuel assemblies reside in lead control bank locations.

The analysis also evaluates some IFBA / WABA combinations and confirms which combinations are bounded by depletion with 24 WABA. Depletion with maximum IFBA / maximum source rods is demonstrated to be bounded by depletion with 24 WABA. Fuel depleted with a potentially more limiting than assumed history (two cycles had >10 steps average lead bank control rod insertion and some Cycle 1 fuel had 24 Pyrex BP) are identified and justified.

Maximum reactivity occurs after approximately 5 days decay time. The burnup credit analysis with "no" decay time are performed with a TRITON decay time of 5 days. The analyses which credit the reactivity reduction resulting from additional decay time are performed by increasing the decay time used at the end of the TRITON fuel depletions.

A bias is calculated to account for maximum horizontal burnup tilts for Regions 2 and 3.

5.2.5 Other Credit

A portion of the Region 1 analysis takes credit for neutron leakage at the boundary of Region 1 and the West SFP wall. A portion of the Region 2 analysis includes control rod credit (fresh 5.0 wt% U-235 fuel with a new or discharged control rod assembly inserted).

Credit for a minimal number of IFBA rods (12) is used in the Region 1 analysis for fresh 5.0 wt% fuel. A small number of IFBA rods (32) are also credited in the multiple misload analysis to represent the lower reactivity fuel assemblies in a maximum reactivity fuel batch. IFBA is not credited in the determination of normal storage minimum burnup requirements (loading curves).

5.2.6 Neutron Absorbers

Regions 1 and 2 take credit for BORAL neutron absorbers. A BORAL coupon monitoring program is used to confirm that the material condition of the BORAL in service remains bounded by the criticality analysis. The monitoring program description has been provided to the NRC [5,6] Because Region 1 and 2 racks are of the same age, manufacturer, and BORAL specification, the MPS3 coupon program covers both Regions. Although the MPS3 BORAL coupons have not experienced blisters, a blister bias is calculated for Region 1 and 2 racks. No B-10 areal density uncertainty is calculated because a bounding low value is used in the SFP k-eff calculations.

5.2.7 Accident Analysis

The accident analysis is dominated by the multiple misload. In this analysis it is assumed that 24 co-located fresh 5.0 wt% fuel assemblies are simultaneously misloaded in the most reactive Region. These assemblies are surrounded by a large number of minimally poisoned fresh 5.0 wt% fuel assemblies. With credit for the TS minimum soluble boron and including biases and uncertainties, SFP 95/95 k-eff is less than 0.95.

An assembly misplacement and drop are analyzed and are considerably less limiting than the multiple misload. The assembly drop is modeled to occur between racks and into a rack cell. The drop is assumed to damage the grids resulting in a fuel pin pitch increase that maximizes k-eff.

For a postulated boron dilution event, it is shown that the 95/95 k-eff is much less than 0.95. Normal operation SFP temperatures (including the associated water density change) are included as part of the normal operation rack k-eff calculations. Analysis of loss of cooling moderator temperatures (up to and including boiling) shows that in an over-temperature accident the 95/95 k-eff is much less than 0.95.

5.2.8 Normal Operations

Analysis is performed for all normal operating conditions including movement, inspection, and reconstitution. All fuel bearing containers in the SFP are identified and analyzed to determine the limitations on location in the pool.

6 Cross Sections, Computer Codes, and Validation

6.1 *Cross Sections and Computer Codes*

This analysis uses the CSAS5 [8] module of SCALE 6.0 [7] to calculate SFP k-eff. Depleted fuel isotopic content is calculated using the SCALE 6.0 t5-depl TRITON [11] module which uses KENO-V.a for the flux calculations needed to collapse neutron cross sections spatially and in energy for the depletion. All analyses are performed using the 238 group ENDF/B-VII cross section library (v7-238).

Code validation is consistent with the DSS-ISG-2010 [4] and NUREG/CRs 6698 [12] and 7109 [13].

6.1.1 CSAS5

The CSAS5 module uses BONAMI to provide resonance corrected cross sections in the unresolved resonance range and WORKER, CENTRM, and PMC to provide resonance-corrected cross sections in the resolved resonance range. This is followed by KENO V.a which uses the processed cross sections to calculate the k-eff of three dimensional system models. Most of the SFP CSAS5 computer runs use a Monte Carlo sampling of 8000 generations, 16000 neutrons per generation, and 1000 generations skipped to achieve a converged statistical uncertainty in k-eff of less than 0.0001.

Unless otherwise specified, all of the k-eff values reported in this document are raw calculated k-eff values with no adjustment for bias and uncertainty. The final values to be compared to the criticality criteria are the calculated values plus the total bias and uncertainty (notated as " $k_{95/95}$ " or 95/95 k-eff).

A uniform initial source distribution in fissile material was used for the k-eff calculations (the default option in CSAS5).

K-eff convergence of KENO cases is verified by comparing the k-eff at the midpoint of retained generations (3500 retained generations) to the final k-eff (7000 retained generations) with the difference in k-eff expressed as the number of final sigma. Nearly all cases had a final k-eff within 2 sigma of the midpoint value (approximately ± 15 pcm). Cases which failed the screening were either rerun with additional generations and generations skipped, or were shown to be non-limiting or otherwise non applicable cases.

The isotopic atom densities used come directly from the initial fuel content or the depletion analysis, except for an adjustment for gaseous or volatile fission products (hereafter called fission gases). Table 6.2 lists the fission gases used in the criticality analysis. Two of the isotopes, Cs-133 and Xe-131, are about 80% of the reactivity worth of the fission gases. The treatment of the fission gases for this criticality analysis is the same as was done for a recently approved Millstone Unit 2 criticality analysis [14] and the key elements of the position are repeated here.

Table 6.2: Fission Product Gases and Volatiles

Noble Gases	^{80}Kr , ^{82}Kr , ^{83}Kr , ^{84}Kr , ^{85}Kr , ^{86}Kr ^{126}Xe , ^{128}Xe , ^{129}Xe , ^{130}Xe , ^{131}Xe , ^{132}Xe , ^{133}Xe , ^{134}Xe , ^{135}Xe , ^{136}Xe
Alkali Metals	^{85}Rb , ^{86}Rb , ^{87}Rb ^{133}Cs , ^{134}Cs , ^{135}Cs , ^{136}Cs , ^{137}Cs
Halogens	^{79}Br , ^{81}Br ^{127}I , ^{129}I , ^{130}I , ^{131}I , ^{135}I

Most of the fission gases remain in the active fuel near where they were created. The mobility of fission gases is important in assessing the consequences of reactor accidents. Regulatory Guide RG 1.183 provides conservative release fractions for fission gases. [15]

It is expected that RG 1.183 will be updated to reflect higher linear powers than were used in developing the current limits. PNNL-18212 Rev. 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," which was completed in June 2011, provides a new analysis. [16] Table 2.9 of PNNL-18212 provides new limiting release rates. PNNL-18212 Appendix C also provides an example of calculated release rates when the linear power is known ("PWR Gap Inventories Based on Realistic Power Histories"). The limiting stable release linear heat rate in that example is ~9 kw/ft from BOC to 32 GWd/MTU and declines thereafter. The product of the MPS3 assembly average linear heat rate and the bounding assembly relative power (Section 8.2) is 8.8 kw/ft through 30 GWd/MTU and declines thereafter. Because the peak linear heat rate magnitude and function of burnup is similar to the bounding MPS3 power history, the Appendix C example release rates may be considered representative for MPS3. Table 6.3 shows the fission gas release fractions from the current RG

1.183, the PNNL-18212 recommended change to the RG, the PNNL-18212 plant dependent release rate from Appendix C, and the release rate assumed for the MPS3 criticality analysis. As shown on Table 6.3, the fission gas release fraction selected is generous compared to Appendix C of PNNL-18212 and is more bounding than the current RG (note that I-131 is low worth).

Table 6.3: Fission Gas Release Fractions

Isotopes	Current RG 1.183	Bounding Proposed in PNNL-18212	Appendix C of PNNL-18212	Implemented in MPS3 Criticality Analysis
K-85	0.10	0.38	0.13	0.20
I-131	0.08	0.08	0.02	0.05
Other Nobles Gases	0.05	0.08	0.02	0.05
Other Halogens	0.05	0.05	0.01	0.05
Alkali Metals	0.12	0.50	0.16	0.20

The release rates for PNNL-18212 assume cladding breach. For the criticality analysis the concern is for the fission products moving into the plenum and therefore no longer acting as effective absorbers. The boiling point of Cesium (951 K) is much higher than the clad temperature during normal fuel use (boiling point of water at 2250 psia is 618 K). It is unlikely that a significant amount of Cesium can migrate away from the fuel due to the clad acting as a condensing surface. This contention is supported by two observations. It is common to use Cs-137 as a measure of burnup. The agreement between Cs-137 and Nd-148 as a burnup measurement for chemical assays has been generally good [17-21]. Also, the BNFL burnup measurement device which is based on Cs-137 has agreed well with the reactor record burnup. [22] For this analysis it is assumed that 20% of the Cs-133 is lost, and Cs-133 is about a third of the worth of the fission gases. Therefore, the fission gas loss assumptions are considered conservative.

6.1.3 TRITON

The t5-depl sequence of SCALE's TRITON [11] enables depletion calculations to be performed by coordinating iterative calls between cross-section processing codes (CENTRM and BONAMI), KENO-V.a, and the ORIGEN-S point-depletion code. A 2D KENO model of the fuel assembly in the core provides the flux distribution needed to collapse the cross sections spatially and in energy for the ORIGEN-S depletion calculations. All the fuel pins are treated as a single depletion material. TRITON parameter addnux=3 is selected to obtain the maximum number of depletion isotopes (Table 6.1). Other than removing light elements (less than oxygen), removing isotopes with concentration too low to affect k-eff, and reducing the concentration of volatile nuclides, the depleted fuel content is taken directly from the TRITON output.

The flux solution for each TRITON depletion step uses a KENO calculation with input requesting 3000 generations and 3000 neutrons per generation (NPG). By default, SCALE increases the number of generations to 3599 and skips 602 generations for the final tally. Past analyses have shown that this many neutron histories or fewer provide adequate convergence [14,23]. The convergence verification in Reference 23 used essentially the same fuel design used in this analysis.

It is important that the depletion time steps are short enough to assure convergence. For this analysis time steps of 10, 40, 50, 50, 50 days are used for the low burnups. The time steps for the rest of the calculations are uniform to meet the desired burnup. The "nlib" is set so that all the remaining time steps are approximately 50 days.

To further confirm adequate convergence, a high burnup credit depletion (54 GWd/MTU) was run four ways:

- Standard convergence (3599 generations, 3000 NPG, 602 skipped)
- More histories (7199 generations, 6000 NPG, 1202 skipped)
- More skipped (3480 generations, 3000 NPG, 1082 skipped)
- Twice as many depletion steps (3599 generations, 3000 NPG, 602 skipped)

A Region 3 rack k-eff KENO case was run for each of the four sets of TRITON depleted fuel content. The largest k-eff difference between the standard convergence case and the other three cases was -24 pcm (slightly lower rack k-eff).

Fuel reactivity after discharge changes fairly rapidly in the first few days due to short half life isotopes such as Xe-135 decaying. Previous analyses performed for Millstone Unit 2 [14] and North Anna [23] confirmed that peak reactivity occurs at 5 days (k-eff is essentially plateaued from 4-7 days). MPS3 and North Anna have the same fuel design. For MPS3, 5 day cooling is selected for peak reactivity with no decay time credit.

6.2 *Uncertainty in Depleted Fuel Isotopic Content*

The MPS3 burnup credit analysis includes an uncertainty term equal to 5% of the depletion reactivity to account for uncertainty in fuel isotopic content given a known fuel assembly burnup. An additional uncertainty term accounts for the uncertainty in the measured fuel burnup (Section 8.15). At the time of this submittal the NRC's interim staff guidance, DSS-ISG-2010-01 [4], permits use of a historical estimate of 5% of the depletion reactivity for the uncertainty and a zero bias. The zero bias is supported by ORNL chemical assay work [19].

Depletion reactivity uncertainty allowance of 5% has been supported based on a conservative estimate of the uncertainty of state of the art fuel management analysis computer codes. Since SCALE has not been used for fuel management, a comparison of in-rack depleted fuel reactivity was performed using fuel isotopic content from the TRITON t5-depl sequence of SCALE, CASMO-4, and CASMO-5. CASMO-4 and CASMO-5 are widely used NRC licensed state of the art fuel management depletion codes. CASMO-4 and CASMO-5 are licensed for MPS3 fuel management use [10, 24].

Results of the TRITON / CASMO-4 / CASMO-5 comparison study were previously reported for Millstone Unit 2 [14] and North Anna. [23] Both studies show that using TRITON atom densities results in somewhat higher SFP rack k-eff than the same calculation using CASMO atom densities. The analysis performed for North Anna is particularly applicable to MPS3 because both plants use essentially the same fuel design (17x17 lattice, discrete and integral boron-based burnable absorbers).

6.3 *Validation of Criticality Analysis*

Criticality computer codes and cross sections are validated for their ability to predict k-eff. The MPS3 criticality validation attempts to reasonably match the MPS3 racks for isotopic content, neutron energy spectrum and geometry.

Due to isotopic limitations in the critical experiments the validation is done in two steps. The first step is to use laboratory critical experiments to validate the structural materials and major actinides in a variety of geometries which produce a range of neutron spectra. The second step is to validate the minor actinides and fission products. Since there is little to no use of these latter isotopes in critical experiments, this validation is based on the uncertainty in the cross section measurement.

6.3.1 Major Actinides and Structural Materials

The validation for the major actinides and structural materials follows NUREG/CR-6698 [12]. Three hundred sixty eight (368) critical experiments were selected from the OECD/NEA handbook [26] and the HTC critical experiments [27] that match the conditions of the MPS3 new fuel storage area and spent fuel pool. These experiments were analyzed with SCALE 6.0 using the 238-group ENDF/B-VII cross-section library. The resulting predicted k's were then fit for trends on the key parameters influencing k. Using these trends, the most limiting bias and uncertainty in the area of applicability was determined. Although some of the trends may not be statistically significant, it is conservative to use all of the trends in determining the limiting bias and uncertainty. Table A.6.1 is the area of applicability for the validation. The MPS3 spent fuel pool is covered by the area of applicability of the validation. Specifically,

1. Enrichment: The benchmarks selected range from 2.35 to 4.74 wt% U-235. The fuel in the spent fuel pool ranges from 2.4 to 5 wt% U-235. The bias decreases with enrichment and the slope is small allowing for a small extrapolation for higher enrichments. Although no assemblies in the pool are below 2.35 wt% U-235, analyses will be performed for enrichments as low as 1.7 wt% U-235. For an enrichment of 1.7 wt% U-235 the extrapolated bias is 0.0028 and the limiting bias is at least 0.0033. This margin makes the extrapolation acceptable.
2. Spectrum: The benchmarks cover a wide range of spectrum by varying the pin pitch. The Energy of the Average Lethargy causing Fission (EALF) of the benchmarks ranges from 0.0605 to 0.8432 eV. This covers the range of spectrum in borated and

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 66 of 300

non-borated conditions in the spent fuel pool and the full moderated condition in the new fuel storage area. Some extrapolation is required for the optimum moderator condition in the new fuel storage area.

3. Fuel Pin Pitch: The fuel pin pitch of the benchmarks ranges from 1.075 to 2.54 cm. The MPS3 fuel pin pitch is 1.26 cm.
4. Assembly Spacing: The benchmarks include spacing between assemblies of 0 to 15.4 cm of water. The maximum separation between assemblies in the flux trap racks is less than []^H. The NFSA has a separation of about 40 cm. The critical experiments cover the entire relevant range since neutron transport through >15.4 cm of full density water has small effect on k. If the water has decreased density, then the separation effectively decreases. The optimum density of water in the NFSA is less than 20%.
5. Boron Areal Density: The critical benchmarks selected cover a range of areal density up to 0.067 gm B-10/cm². The maximum areal density credited for the MPS3 analysis is []^H. Cd containing experiments were included to cover credited control rods.
6. Soluble Boron: The benchmarks have soluble boron concentrations up to 5030 ppm. The soluble boron credited to meet the k less than 0.95 and credited for the accident analysis is well within the range of experiments.

Details on the area of applicability can be found in Appendix A.

For the UO₂ and MOX critical experiments, the change in the k predicted for the critical experiments as a function of a parameter is performed for five different parameters; enrichment, pin diameter, pin pitch, soluble boron content, and EALF. For the MOX set a trend on the Pu enrichment was also performed. Rather than eliminate any trend, the largest bias from any of the trends is selected. The largest bias was generally from the trend as a function of EALF. Table 6.6 provides the biases and uncertainties as a function of EALF. The EALF tabulated is the maximum allowed EALF for a given bias and uncertainty.

For the spent fuel pool, the bias and uncertainty depends on the burnup since at low burnup the dominant fissile material is U-235 and at high burnup the dominant fissile material is Pu-239. In order to avoid trying to properly weight the critical experiments for the amount of U-235 and Pu-

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 67 of 300

239, two sets of bias and uncertainty are employed; one from the fresh UO_2 critical experiments and one from the MOX critical experiments. The final bias and uncertainty employed is that which produces the highest 95/95 k. The UO_2 critical experiments have a higher bias but lower uncertainty than the MOX experiments. Since the uncertainty is statistically combined with other uncertainties it is not possible to determine which set is more limiting until the other uncertainties due to factors such as manufacturing tolerances are determined. For the new fuel storage area or unburned fuel in the spent fuel pool only the UO_2 set from Table 6.6 is used.

Table 6.6: Summary of Validation Bias and Uncertainty From Major Actinides and Structural Materials(Calculate with both UO₂ and MOX bias and uncertainty and use the set that provides the highest 95/95 k)

UO ₂		
Maximum EALF (ev)	Bias	Uncertainty
0.35	0.0034	0.0048
0.40	0.0034	0.0048
0.45	0.0037	0.0048
0.50	0.0039	0.0048
0.55	0.0042	0.0049
0.60	0.0044	0.0050
0.65	0.0047	0.0051
0.70	0.0049	0.0052
0.75	0.0052	0.0053
0.80	0.0054	0.0054
0.85	0.0056	0.0055
0.90	0.0059	0.0057
0.95	0.0061	0.0058
1.00	0.0064	0.0059
1.05	0.0066	0.0060
1.10	0.0069	0.0061
1.15	0.0071	0.0062
1.20	0.0074	0.0063

MOX		
Maximum EALF (ev)	Bias	Uncertainty
0.35	0.0017	0.0083
0.40	0.0018	0.0088
0.45	0.0019	0.0094
0.50	0.0021	0.0099
0.55	0.0022	0.0104
0.60	0.0023	0.0110
0.65	0.0025	0.0115
0.70	0.0026	0.0121
0.75	0.0027	0.0126
0.80	0.0029	0.0132
0.85	0.0030	0.0137
0.90	0.0031	0.0143
0.95	0.0033	0.0148
1.00	0.0034	0.0154
1.05	0.0035	0.0159
1.10	0.0037	0.0165
1.15	0.0038	0.0170
1.20	0.0039	0.0176

6.3.2 Minor Actinides and Fission Products

Since there are few to no critical experiments that contain some of the isotopes used in this criticality evaluation, validation is done by estimating the maximum error in k-eff due to cross section measurement uncertainty. NUREG-7109 has shown that applying a bias of 1.5% of the worth of the minor actinides and fission products conservatively accounts for both the bias and uncertainty due to the minor actinides and fission products. [13] NUREG-7109 mainly addresses the 28 highest worth isotopes but on the last sentence of page 106 it states, "An upper value of 1.5% of the worth is also applicable for SNF isotopic compositions consisting of all nuclides in the SFP configuration." NUREG-7109 limits the applicability to certain cross section sets but ENDF/B-VII used here is one of those sets. The use of the 1.5% bias is part of the NRC's transport division in ISG-8 Rev.3. [28]

The minor actinides are defined as actinides not contained in the criticality validation benchmarks. Table 6.1 lists all the isotopes used in the analysis. The major actinides are U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241. U-236 is not a major actinide although it has a significant worth in spent fuel. Am-241 is treated as a major actinide because it decays from Pu-241 and is included in the MOX critical experiments.

The fission products used are listed on Table 6.1. Pb-208 is neither a fission product nor an actinide but is included in the analysis of burned fuel. Its atom density is extremely small with no real impact on the criticality analysis. It is treated as a fission product.

6.3.3 Temperature Dependence

All of the critical experiments utilized in Section 6.3.1 were done at room temperature. There is one set of critical experiments which were run as a function of temperature in the range of interest for spent fuel pools. There are a couple of sets of experiments with temperatures greater than 200 C [29, 30] but LEU-COMP-THERM-046 [30] is ideal for determining a bias as a function of temperature in the range of interest. LEU-COMP-THERM-046 is not used in the set of experiments from Section 6.3.1 since in general they are at elevated temperatures and as such represent a unique set. The analysis of this temperature dependent set is detailed in Appendix A Section 8.

The analysis of the only set of thermal critical experiments in the International Handbook that uses elevated temperatures has shown a small increase in the bias with temperature. This increase can be conservatively handled by a bias from room temperature (293K) of $1.7\text{E-}05 \Delta k/\Delta^\circ\text{C}$.

6.3.4 Absorbers Credited

BORAL is directly credited in the NFSA and SFP analyses. Control rod credit is also used in portions of the Region 2 analyses. Two different control rod absorber materials are evaluated (Ag-In-Cd and Hf/Zr). To support credit for BORAL in the validation area of applicability, 28 International handbook experiments and 19 HTC experiments with borated plates were included in the fresh fuel and burned fuel sets, respectively. To support control rod credit, 17 International Handbook experiments and 8 HTC experiments utilizing cadmium absorber plates were included in the fresh fuel and burned fuel sets, respectively. Although not all control rod nuclides are explicitly represented in the validation, the control rod credit cases include a large amount of margin to the limit.

7 Criticality Safety Analysis of the New Fuel Storage Area

7.1 *New Fuel Storage Area KENO Model*

The description of the New Fuel Storage Area (NFSA) is given in Section 4.1. The SCALE CSAS5 (KENO) model for the NFSA is a three dimensional model of the entire rack including the concrete walls and floor. Table 4.1 and 4.2 provides the dimensions and tolerances.

Figure 7.1 shows a cutaway view of the NFSA KENO model. This may be compared to the actual rack shown on Figures 4.1 and 4.2. Structural components are omitted, and the area above the active fuel is simplified. Cell walls are assumed to be straight rather than flared out at the top. Except for the cell wall, the area above the active fuel is assumed to be water at the same density as throughout the rack. The top and bottom nozzles are also modeled as water. Figure 7.2 shows a portion of the NFSA KENO model from the top with water removed. Figure 7.3 is a less detailed view of the whole NFSA model.

The New Fuel Storage Vault has concrete walls and floor that is lined with a ¼ inch stainless steel liner. Behind that liner is a concrete wall. It is unclear how thick the walls and floor is in some places, so the KENO model conservatively extended the wall and floor thickness to 100 cm (>3 ft). Sensitivity cases confirm that thicker walls are conservative for the k-eff calculations.

Unless otherwise noted, each of the KENO cases run use 8000 generations, 16000 neutrons per generation, and skips 1000 generations. The initial source distribution of neutrons is uniform in the fuel. Void boundary conditions are used on the six sides of the model.

Figure 7.1: Cutaway View of the New Fuel Storage Area Model

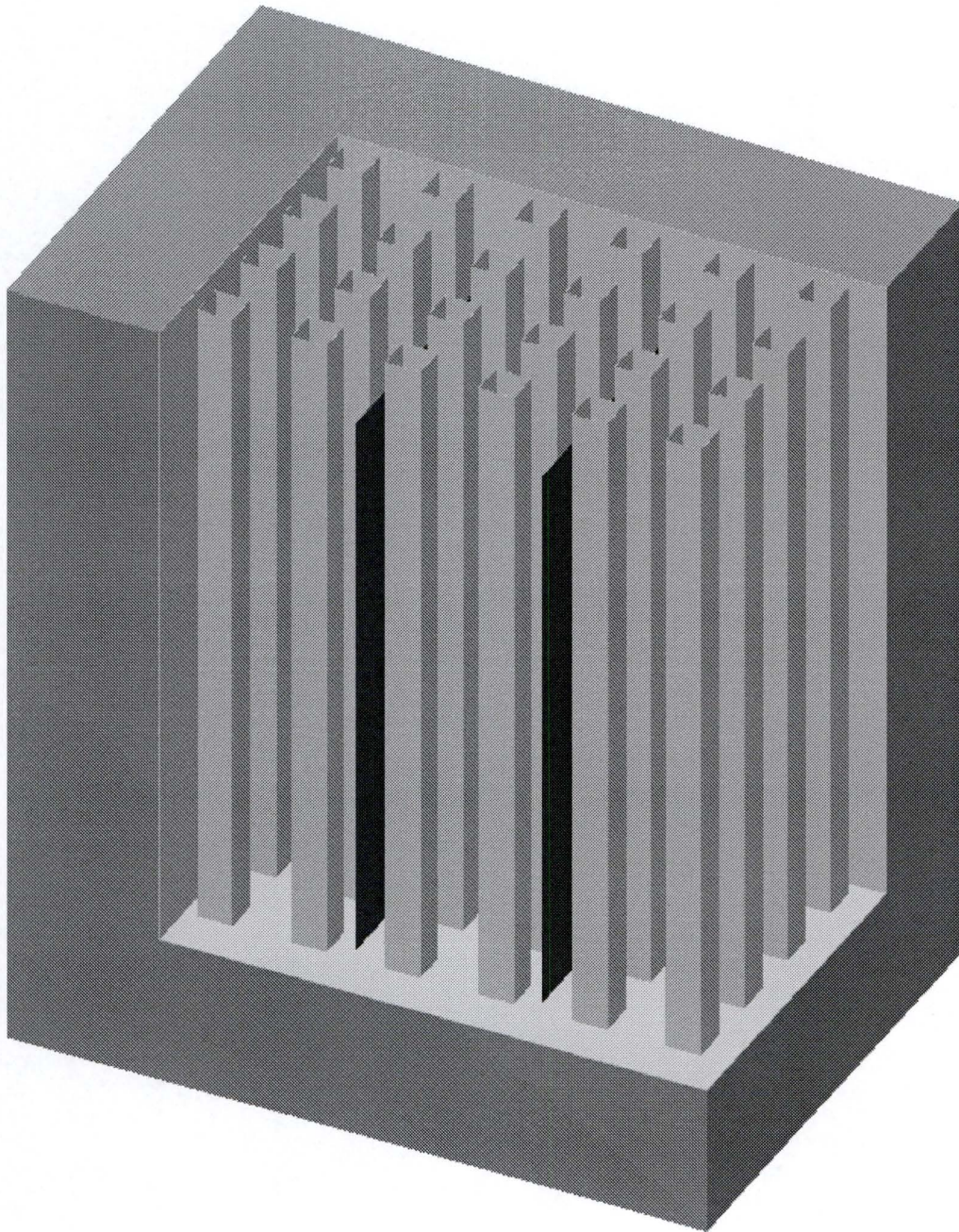


Figure 7.2: Top View of Part of the New Fuel Storage Area Model

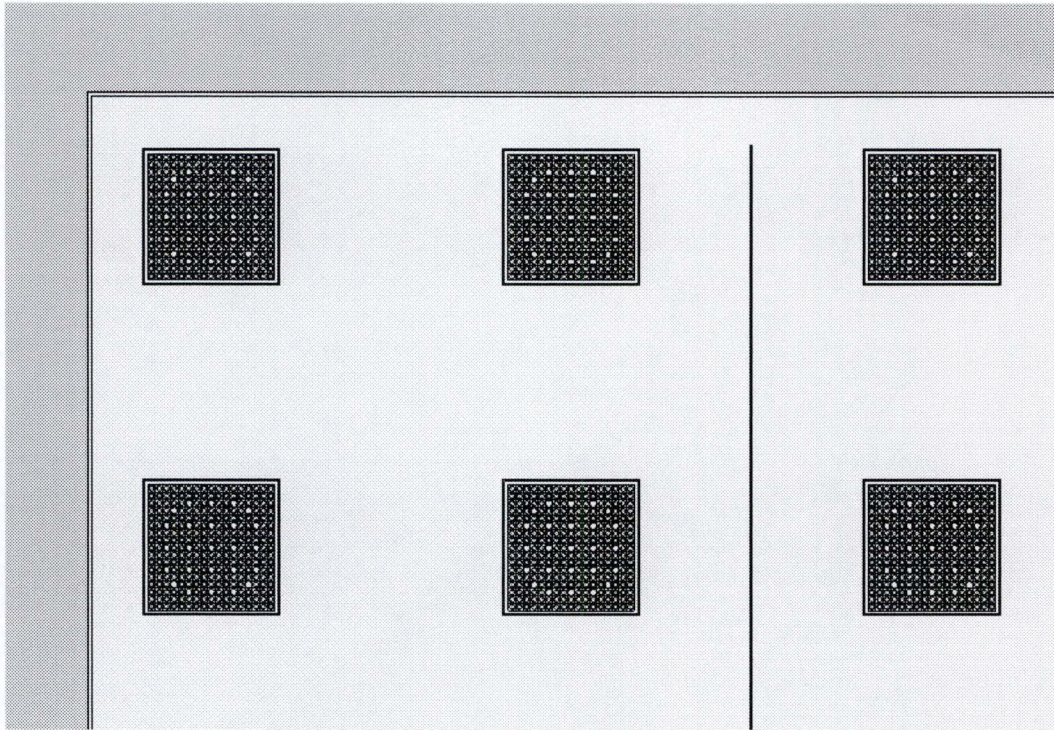
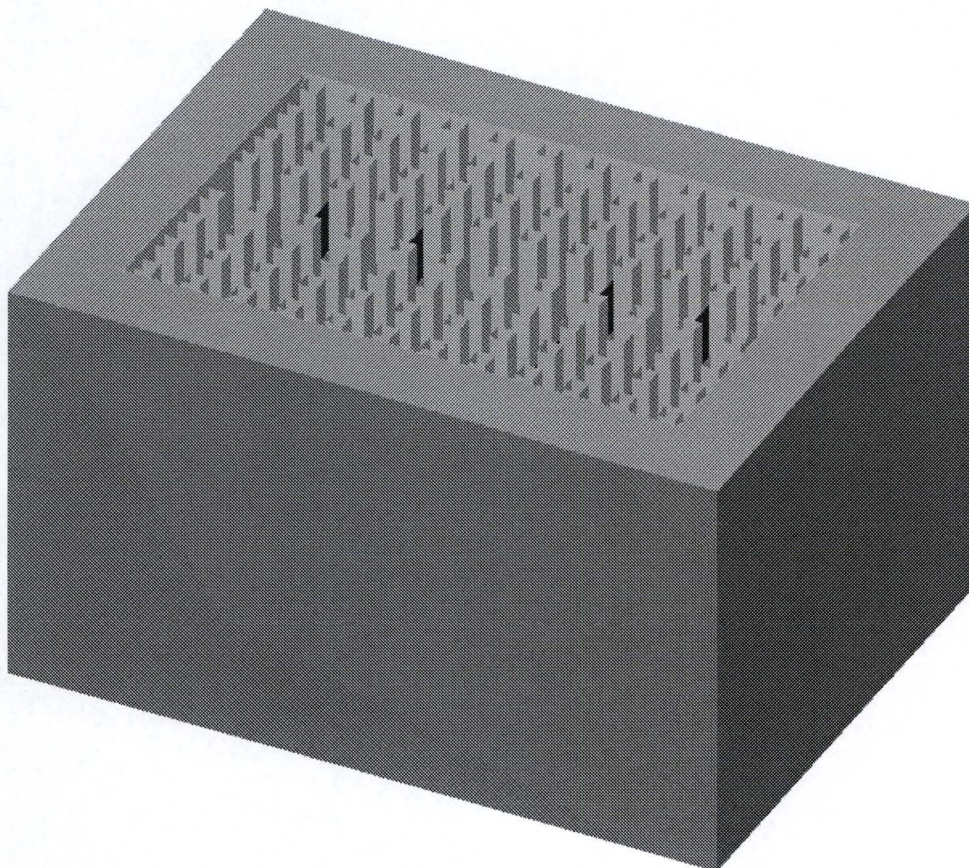


Figure 7.3: Full New Fuel Storage Area Model



7.2 Limiting Fuel Design

The NFSR analysis employs a composite fuel design that bounds the range of variation in MPS3 fuel.

7.2.1 Fuel Dimensions and Materials

Variation in fuel dimensions and clad materials over the history of MPS3 is summarized in Table 3.1. To bound the neutronic effect of various clad materials, clad is modeled as zirconium. Omitting alloying elements conservatively increases k-eff.

The difference in guide tube designs in Table 3.1 is small. Tolerance cases (Section 7.4) indicate that less water displacement in the lattice increases k-eff. Therefore, the guide tube design with the least water displacement (V5H) is used in the NFSR model. For the same reason, the minimum zirconium-based grid volume design (V5H) is used.

Tolerance cases (Section 7.4) confirm that adding more stainless steel rack structure (thicker storage cell walls) to the model reduces k-eff. Therefore, NFSR cell support structure is omitted and fuel assembly hardware above and below the active fuel region is modeled as moderator.

There are normally small amounts of U-234 and U-236 in the fuel that decrease k-eff. U in UO_2 is conservatively assumed to be composed of only U-235 and U-238.

7.2.2 Fuel Density, Burnable Poisons, and Axial Blankets

Fuel density is calculated as a fraction of theoretical UO₂ density (10.96 g/cm³ maximum). The calculation credits the nominal dish and chamfer volume as a reduction in net stack density in order to preserve the correct amount of fuel in the central (non-annular pellet) zone. For example, if a pellet is produced that has a material density 97 percent of theoretical density (PTD) but has 2% volume reduction due to geometric dish and chamfer, the density used in the KENO model is 97% x 98% or 95.06 PTD. Note that the missing fuel in annular blanket pellets is not credited when calculating the stack density.

Figure 7.4 shows the PTD net stack density for each fuel assembly used at MPS3 through Cycle 17. Based on the as-built fuel data, a bounding net stack density of 95.5 PTD is used in the NFSR KENO models. A fuel density uncertainty tolerance of []^{a,c} will be included to provide an estimate of the reactivity variation about the average within a fuel batch due to fuel loading variations. By inspection of Figure 7.4, the historical variation in stack density of fuel in most batches is less than 1% (maximum – minimum).

Figure 7.4: MPS3 Fuel Assembly Stack Density

a,c

Burnable poisons are not credited in the NFSR analysis.

All MPS3 fuel pins have dished and chamfered pellets, some may have reduced enrichment axial blankets, and some may use some annular pellets. In the NFSR KENO models, fuel is modeled as a right circular cylinder. Ignoring annular pellets overstates the amount of fuel in the assembly. Tolerance cases in Section 7.4 (increased fuel density and increased pellet OD) confirm that this modeling simplification is conservative. Fuel in the axial blanket region is modeled as 5.0 wt% fuel. This modeling simplification is conservative because modeling higher than actual fuel enrichment increases k-eff.

7.3 Limiting Rack Model

7.3.1 NFSR Materials and Dimensions

The new fuel storage area has concrete walls and a concrete floor and is lined with a ¼ inch stainless steel liner. The KENO model encompasses the entire area and uses nominal dimensions described in Tables 4.1 and 4.2. Storage cells are stainless steel. BORAL is modeled with the minimum certified B-10 areal density (0.005 g/cm^2). A fresh 5.0 wt% U-235 fuel assembly is modeled in each storage cell.

Due to some uncertainty about the thickness of the concrete, the KENO model conservatively assumes a thickness of 100 cm (>3 ft) thick. Thickness sensitivity cases verified that k-eff is insensitive to concrete thicknesses greater than about 70 cm. The storage cell and concrete wall height result in over 50 cm of moderator above the top of the active fuel. Moderator displacement above the fuel by the assembly top nozzle and rack structural materials is not modeled.

Previous new fuel storage criticality calculations [23] performed to determine the effect of different concrete compositions showed that “EPRI Dry Concrete” composition [31] results in the highest NFSR k-eff of the options considered. The composition of EPRI Dry Concrete is shown in Table 7.1.

Table 7.1: EPRI Dry Concrete Composition

Element	Concentration (wt%)
Fe	0.45
H	0.0
C	14.00
O	42.52
Na	2.32
Mg	7.53
Al	2.72
Si	26.94
Ca	3.52
Density (g/cc)	2.90

To verify that EPRI Dry Concrete is a bounding composition for the MPS3 NFSR, KENO k-eff calculations were run for five different concrete compositions. The four additional concrete compositions come from the SCALE manual [7]. Table 7.2 shows the results. EPRI Dry Concrete will be used in the KENO NFSR models because it produces the highest k-eff.

Table 7.2: k-eff of NFSR vs. Concrete Composition

Case Description	k-eff	σ	Δk
EPRI Dry Concrete	0.90678	0.00008	N/A
SCALE Magnuson	0.90643	0.00008	-0.00035
SCALE Oak Ridge	0.90630	0.00008	-0.00049
SCALE Regulatory	0.90611	0.00008	-0.00067
SCALE Rocky Flats	0.90629	0.00008	-0.00050

MPS3 fuel assembly designs have an active fuel length of 144 inches. However, different fuel assembly designs have had different bottom nozzle heights, end plug heights and gap size between the bottom nozzle and end plug. The axial location of the active fuel relative to the New Fuel Storage Rack structure will change depending on the future fuel design type. Table 7.3 shows the distance from the bottom of the fuel assembly to the bottom of the active fuel for current and past MPS3 fuel designs.

Table 7.4 shows that the reactivity of the NFSR is insensitive to small changes to the axial position of the active fuel. The base model uses the RFA-2 fuel axial position.

Table 7.3: Axial Location of Active Fuel for Various Fuel Design Types

Component	Design Characteristics (cm)				
	Standard	V5H Batch 6-7	V5H Batch 8	RFA-2	NGF
End Plug Height	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}
Gap (rods to bottom nozzle)	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}
Bottom Nozzle Height	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}
Bottom of Bottom Nozzle to Bottom of Active Fuel	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}

Table 7.4: k-eff of NFSR vs. Active Fuel Axial Location

Case Description	k-eff	σ	Δk
Bottom of Active Fuel at [] ^{a,c}	0.90678	0.00008	N/A
Bottom of Active Fuel at [] ^{a,c}	0.90671	0.00008	-0.00008

7.3.2 Temperature and Flooding

Under normal storage conditions, fuel in the NFSR is stored dry and at room temperature. The maximum normal temperature in the fuel building is less than 100°F. The maximum normal operating temperature of the adjacent SFP is 150°F. Analysis of the NFSR includes flooding with un-borated water (the SFP contains borated water) and with optimum density hydrogenous moderator.

Flooding of the NFSR is not part of normal fuel storage, therefore the temperature of the moderator is not known. The freezing point of water (32°F) is a practical lower bound for NFSR temperature. Because the primary effect of water temperature change on NFSR k-eff is related to the associated water density change, water density at 32°F is set to 1.0 g/cm³ to cover the maximum density of water at atmospheric pressure. The upper NFSR temperature is assumed to be 150°F (0.98 g/cm³).

For temperatures above 68°F a k-eff bias (1.7E-05 $\Delta k/^{\circ}\text{C}$) is added based on KENO code validation results (Section A.8). At 150°F the bias is 0.00078. Base case NFSR KENO results in Table 7.5 show that higher temperatures are limiting for low density moderator, but low temperatures are limiting for NFSR flooding with water. The validation temperature bias has not been added to the Table 7.5 k-eff values.

Table 7.5: NFSR Reactivity Sensitivity to Water Density

Case Description	k-eff	σ	Δk
32°F, 1.00 g/cc	0.90509	0.00009	N/A
150°F, 0.98 g/cc	0.90136	0.00008	-0.00373
32°F, 0.065 g/cc	0.87071	0.00007	-0.01043
150°F, 0.065 g/cc	0.88114	0.00007	N/A

7.3.3 Low Density Moderator

Table 7.6 and Figure 7.5 show the k-eff of the NFSA over a range of moderator densities.

These cases were run at 150°F or 32°F, with the assemblies centered in their cells, and with EPRI dry concrete. NFSR k-eff is maximized when flooded with full density water.

The full density water case bounds low density moderator cases for both temperature ranges.

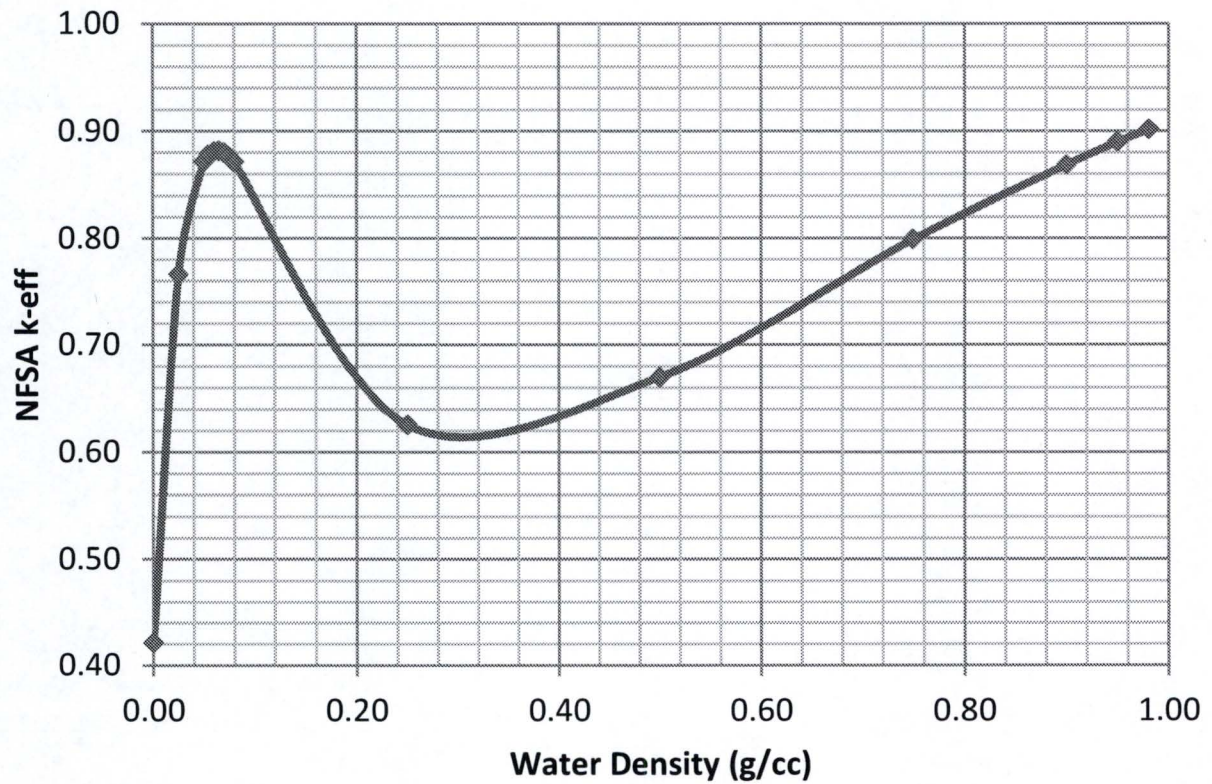
The calculated k-eff with full flooding is at least 0.02 dk higher than the highest k-eff of the low density moderator cases. In addition, the regulatory limit of the full density case is 0.03 dk lower than the low density moderator limit. Therefore, the limiting condition is the fully flooded case.

Key bias and uncertainty cases in Section 7.4 confirm that when bias and uncertainty are included, fully flooded is the limiting condition.

Table 7.6: NFSR Reactivity Sensitivity to Water Density

Water Density	k-eff	σ	Δk
0.000 g/cc, 150°F	0.42128	0.00004	-0.48008
0.025 g/cc, 150°F	0.76546	0.00007	-0.13590
0.045 g/cc, 150°F	0.87124	0.00007	-0.03012
0.050 g/cc, 150°F	0.87764	0.00007	-0.02372
0.055 g/cc, 150°F	0.88061	0.00007	-0.02075
0.060 g/cc, 150°F	0.88114	0.00007	-0.02022
0.065 g/cc, 150°F	0.87949	0.00007	-0.02188
0.070 g/cc, 150°F	0.87608	0.00008	-0.02528
0.075 g/cc, 150°F	0.87119	0.00007	-0.03017
0.250 g/cc, 150°F	0.62515	0.00006	-0.27622
0.500 g/cc, 150°F	0.66980	0.00008	-0.23157
0.750 g/cc, 150°F	0.79893	0.00008	-0.10243
0.900 g/cc, 150°F	0.86851	0.00008	-0.03285
0.950 g/cc, 150°F	0.88978	0.00008	-0.01158
0.980 g/cc, 150°F	0.90136	0.00008	N/A
0.055 g/cc, 32°F	0.86951	0.00007	-0.03558
0.060 g/cc, 32°F	0.87127	0.00007	-0.03383
0.065 g/cc, 32°F	0.87071	0.00007	-0.03438
1.000 g/cc, 32°F	0.90509	0.00009	N/A

Figure 7.5: k-eff of NFSR vs. Water Density



7.3.4 Asymmetric Fuel Placement

Asymmetric placement of the fuel within the storage cells is evaluated using four different eccentric configurations. Figure 7.6 shows how the assemblies were oriented for each case.

Table 7.7 shows that the highest k-eff is achieved when the assemblies are shifted towards the B or D locations (both cases have essentially the same k-eff). The NFSR KENO model has assemblies asymmetrically skewed towards the B locations shown in Figure 7.6.

Figure 7.6: Asymmetric Skewing of Assemblies

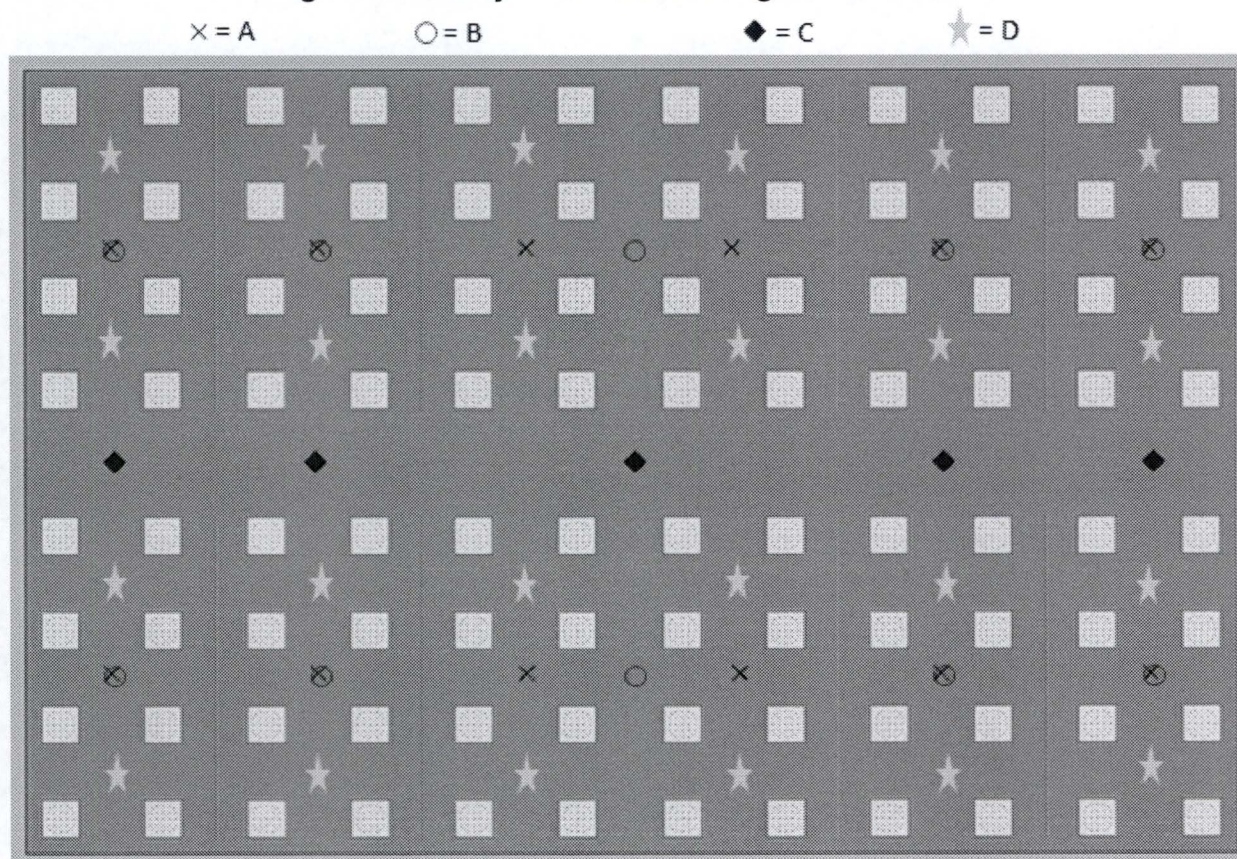


Table 7.7: k-eff of NFSR vs. Assembly Asymmetric Position

Case Description	k-eff	σ	Δk
Fuel Assembly Centered in Cell	0.90509	0.00009	-0.00169
Fuel Assemblies Shifted Towards A	0.90661	0.00008	-0.00017
Fuel Assemblies Shifted Towards B	0.90678	0.00008	N/A
Fuel Assemblies Shifted Towards C	0.90671	0.00008	-0.00007
Fuel Assemblies Shifted Towards D	0.90680	0.00008	0.00001

7.3.5 Summary of the Base Case for the NFSA Analysis

Sections 7.1-7.3 describe the development of the limiting MPS3 NFSR model. The base case model includes these features:

1. All storage cells contain a 5 wt% U-235 fresh fuel assembly with no burnable absorbers.
2. The fuel contains no U-234 or U-236.
3. The fuel stack density is 95.5% of the UO₂ theoretical density.
4. The fuel model includes minimum water displacement guide tubes and grids
5. The fuel clad material is modeled as zirconium.
6. Rack structure except the storage cells, liner, and concrete is ignored.
7. EPRI dry concrete composition is used to represent the concrete.
8. Concrete thickness is conservatively assumed to be 100 cm.
9. The minimum certified BORAL B-10 areal density is used.
10. Fuel is asymmetrically positioned in each rack cell to maximize k-eff.
11. The fuel and moderator temperature for the fully flooded condition is 273 K with water density set to 1.0 gm/cm³.
12. KENO cases are run with 8000 generations, 16,000 neutrons per generation, and 1000 generations skipped.

7.4 Biases and Uncertainties for the New Fuel Storage Area Analysis

Table 7.8 shows the results of manufacturing uncertainty and bias cases. A full set of cases were run at the limiting water temperature and density (32°F and 1.0 g/cm³). Cases that were found to be most significant at low density moderation in a previous analysis [23] are also included in the Table for comparison (150°F and 0.06 g/cm³) to support the conclusion that flooding with water is the most limiting condition when biases and uncertainty are included.

Table 7.8: MPS3 NFSR Bias and Uncertainty Cases

Case Description	k-eff	σ	Δk
Nominal, 32 F (Base case)*	0.90678	0.00008	N/A
Fuel Stack PTD, Increase	0.90726	0.00008	0.00047
Pellet OD, Increase	0.90700	0.00008	0.00022
Active Fuel Length, Increase	0.90654	0.00008	-0.00024
Active Fuel Length, Decrease	0.90638	0.00008	-0.00040
Clad ID, Increase	0.90625	0.00009	-0.00053
Clad ID, Decrease	0.90697	0.00008	0.00019
Clad OD, Decrease	0.90838	0.00008	0.00160
Guide Tube ID, Increase	0.90691	0.00008	0.00013
Guide Tube ID, Decrease	0.90646	0.00008	-0.00032
Guide Tube OD, Increase	0.90624	0.00008	-0.00054
Guide Tube OD, Decrease	0.90691	0.00008	0.00013
Pin Pitch, Increase	0.90727	0.00008	0.00049
Cell Wall Thickness, Increase	0.90556	0.00008	-0.00122
Cell Wall Thickness, Decrease	0.90793	0.00008	0.00115
Cell Pitch, Increase	0.90665	0.00008	-0.00013
Cell Pitch, Decrease	0.90637	0.00008	-0.00041
Rack Pitch, Decrease	0.90687	0.00008	0.00009
BORAL Radial Position	0.90655	0.00008	-0.00023
Stainless Steel Liner, Decrease	0.90668	0.00009	-0.00010
Concrete Thickness, Increase (+100 cm)	0.90684	0.00009	0.00006
BORAL Cutouts**	0.90673	0.00008	-0.00005
Nominal, 150 F, 0.060 g/cc (Base case)*	0.88152	0.00007	N/A
Cell Pitch, Decrease, 0.060 g/cc	0.88285	0.00007	0.00132
Cell Wall Thickness, Decrease, 0.060 g/cc	0.88700	0.00008	0.00547
Concrete Thickness, Increase, 0.060 g/cc	0.88259	0.00008	0.00107
Pin Pitch, Increase, 0.060 g/cc	0.88186	0.00007	0.00033
Foam Flooded Fuel Building 0.060 g/cc	0.88481	0.00007	0.00329

*Base cases differ from Table 7.6 due to asymmetric fuel placement.

** The "BORAL cutouts" case models holes in the BORAL through which rack structure passes.

The uncertainty case results for the fully flooded condition are combined in Table 7.9 using Equation 7.1:

$$\text{Total Unc.} = \left[\sum \left(k_{\pm \text{Unc.}} - k_{\text{Nominal}} + 2 * \sqrt{\sigma_{\pm \text{Unc.}}^2 + \sigma_{\text{Nominal}}^2} \right)^2 \right]^{1/2} \quad (7.1)$$

The “Code Benchmarking Unc.” and “Code Benchmarking Bias” are described in Section A.7. The base case has an EALF of 0.24 eV. There is no “Code Temperature Bias” for the flooded cases because the cases are run at below room temperature. Table 7.9 shows that the MPS3 NFSR has 0.0242 Δk Dominion margin.

Table 7.9: MPS3 NFSR Maximum k-eff, Full Density Water

Case Description	k-eff	σ	Δk^*	Max Δk^*
Base Case				
Nominal, 32F (Base case)	0.9068	0.0001	N/A	N/A
Uncertainties				
Fuel Stack PTD, +	0.9073	0.0001	0.0005	0.0007
Pellet OD, +	0.9070	0.0001	0.0002	0.0004
Active Fuel Length, +	0.9065	0.0001	-0.0002	0.0000
Clad ID, -	0.9070	0.0001	0.0002	0.0004
Clad OD, -	0.9084	0.0001	0.0016	0.0018
GT ID, +	0.9069	0.0001	0.0001	0.0003
GT OD, -	0.9069	0.0001	0.0001	0.0003
Pin Pitch, +	0.9073	0.0001	0.0005	0.0007
Cell Wall Thickness, -	0.9079	0.0001	0.0011	0.0014
Cell Pitch, +	0.9066	0.0001	-0.0001	0.0001
Rack Pitch, -	0.9069	0.0001	0.0001	0.0003
Wall Thickness, +	0.9068	0.0001	0.0001	0.0003
Stainless Steel Liner Thickness, -	0.9067	0.0001	-0.0001	0.0001
KENO Case Uncertainty	N/A	N/A	N/A	0.0002
Code Benchmarking Unc.	N/A	N/A	N/A	0.0048
RSS of Uncertainties	0.0055			
Biases				
BORAL Cutout Bias	0.9067	0.0001	0.0000	0.0002
Code Temperature Bias	N/A	N/A	N/A	0.0000
Code Benchmarking Bias	N/A	N/A	N/A	0.0034
Sum of Biases	0.0036			
Summary				
Base Case k-eff	0.9068			
Total Bias and Uncertainty	0.0091			
NRC Administrative Margin	0.0100			
Maximum k-eff	0.9258			
10CFR50.68 Limit	0.9500			
DOMINION Margin	0.0242			

* Δk is k₂-k₁. Max Δk is Δk + 2*RSS(σ₁, σ₂)

To support the conclusion that the flooded case is more limiting than the low density moderator case, an estimate of the low density moderator total bias and uncertainty is calculated using six uncertainty and three biases values. Four uncertainty cases and one bias case are shown in Table 7.8. The two remaining uncertainties are the Monte Carlo uncertainty and code benchmarking uncertainty, and the two remaining biases are both code benchmarking biases. The six uncertainty values accounted for most of the total uncertainty of the North Anna New Fuel Storage Area [23].

Computer code validation bias and uncertainty requires extrapolation due to the high EALF values of the low density moderator cases. The water density 0.06 g/cm³ water density case has an EALF of 1.34 eV. The upper end of the EALF range in the benchmark cases is 0.85 eV (Appendix A). The required extrapolation is 0.5 eV. Based on a linear trend, the code validation bias increases 0.0024 dk (to 0.0081) and the uncertainty increases 0.0012 dk (to 0.0067). Due to the large extrapolation, the code validation bias and uncertainty will be determined by doubling the amount of extrapolation. Table 7.10 shows the margin calculation for the low density moderator condition. There is over 0.06 dk margin to the limit based on this estimate. The low density moderator condition is not limiting.

Table 7.10: Maximum k-eff for MPS3 NFSR, Optimum Moderation

Case Description	k-eff	σ	Δk	Max Δk
Base Case				
Nominal, 32F (0.065 g.cc)	0.8824	0.0001	N/A	N/A
Base Case for Biases & Uncertainties (0.060 g/cc)	0.8815	0.0001	N/A	N/A
Uncertainties				
Pin Pitch, +	0.8819	0.0001	0.0003	0.0005
Cell Wall Thickness, -	0.8870	0.0001	0.0055	0.0057
Cell Pitch, -	0.8828	0.0001	0.0013	0.0015
Concrete Thickness, +	0.8826	0.0001	0.0011	0.0013
KENO Case Unc	N/A	N/A	N/A	0.0001
Code Benchmarking Unc.	N/A	N/A	N/A	0.0078
RSS of Uncertainties	0.0099			
Biases				
Foam Flooded Building Bias	0.8848	0.0001	0.0033	0.0035
Code Temperature Bias	N/A	N/A	N/A	0.0008
Code Benchmarking Bias	N/A	N/A	N/A	0.0105
Sum of Biases	0.0148			
Summary				
Base Case k-eff	0.8824			
Total Bias and Uncertainty	0.0246			
NRC Administrative Margin	0.0100			
Maximum k-eff	0.9170			
10CFR50.68 Limit	0.9800			
DOMINION Margin	0.0630			

7.5 Accident Conditions

7.5.1 Optimum Moderation

It should be noted that it would be difficult for the Millstone NFSR to be flooded with low density moderator. This is because most of the NFSR is usually covered. Regardless, the accident scenarios of flooding the NFSR with water and optimum moderator were analyzed in Section 7.4. Based on calculated margin to the k-eff limits, flooding the NFSR with water bounds flooding with foam by about $3\%\Delta k$.

7.5.2 Dropped/Misplaced Assembly

A dropped/misplaced assembly would be analyzed in an air moderated NFSR. This is because of the double contingency principle which specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions. In addition, the Millstone NFSR has stainless steel tube covers over the whole vault which prohibits misplacing an assembly or crushing another assembly. The rack structure does not provide enough open space between storage cells for a fuel assembly to be mis-placed. The maximum opening (assuming all the NFSR covers were off) is $8 \frac{1}{16}$ inches wide. A fuel assembly is about 8.4 inches wide. Therefore, neither the misplaced assembly nor crushed assembly accidents need to be analyzed.

7.5.3 Seismic Event

During a seismic event, free standing equipment shifts around which can move fuel closer or farther apart. However, the MPS3 NFSA has no free standing equipment except for the assemblies inside the stainless steel tubes. Fuel assemblies can shift inside the stainless steel tubes, but that scenario is covered by the base case model asymmetric fuel positioning.

8 Depletion Modeling and Burnup Effects

This section describes the MPS3 SCALE 6.0 (TRITON) depletion models and conservative depletion conditions suitable for use in the MPS3 SFP criticality calculations.

8.1 *Depletion Method Overview*

TRITON depletions are used to determine the isotopic content of depleted fuel for Spent Fuel Pool criticality analysis, specifically to develop burnup curves (required minimum fuel burnup as a function of initial enrichment). Performing TRITON depletions requires a SCALE model of an axial segment of a fuel assembly that includes geometry, material content, and depletion conditions (fuel temperature, moderator temperature and density, soluble boron, presence of burnable absorbers and or control rods, and depletion power).

Conservatism (maximizing SFP fuel reactivity) is incorporated via use of a bounding fuel assembly design and by choice of input depletion conditions that bound anticipated actual fuel depletion conditions. The methodology associated with determination of conservative models and conditions is consistent with that used for the recently accepted License Amendment Requests. [14, 23]

Section 6 provides information on the computer code used (t5-depl sequence of SCALE 6.0 TRITON), the cross section library (238 group ENDF/B-VII), how it was run (number of neutrons followed, time step size, isotopes followed, and cooling time), and the validation.

Each fuel assembly is modeled using 18 equal size axial nodes. For each node the burnup is the product of the assembly average burnup and the relative axial burnup distribution (Section 8.6). The axial burnup distribution also provides the relative burnup averaged axial power distribution which is used to determine depletion conditions for each node. The following parameters are included in the considerations for each depletion node:

- Assembly power (Section 8.2)
- Soluble boron (Section 8.3)
- Moderator temperature and density (Section 8.4)
- Fuel temperature (Section 8.5)
- Axial burnup shape (Section 8.6)
- Control rod insertion history (Section 8.8)
- Horizontal burnup shape (Sections 9.6.6 and 9.9.5)
- Spacer grids (Section 8.12)
- Fuel density (Section 7.2.2)

- Variation in fuel dimensions (Section 8.11)
- In-core thimble for flux map detectors (Section 8.9)
- Burnable absorber and history (Section 8.7)
- Decay time (Section 5.2.4)
- Reduced Power Prior to Discharge (Section 8.10)

TRITON models are conservative in the sense that fuel features and depletion conditions will be selected to accommodate past, present, and expected future fuel designs and depletion history in a way that maximizes Spent Fuel Pool (SFP) k. TRITON models are best estimate in the sense that uncertainties in fuel features (such as clad OD design tolerance) are not considered. The methodology associated with determination of conservative models and conditions is consistent with previous analyses. [14, 23]

Each of these features or conditions will be evaluated using fuel design information, core design history, and operating history. Justification for the conservatism of each feature or condition will be provided using first principles, prior evaluations, or TRITON depletion sensitivity cases.

8.2 Bounding Fuel Assembly Depletion Power

Fuel and moderator temperatures depend in part on the fuel assembly power. Depletion with higher fuel and moderator temperature increases fuel reactivity in the SFP via increased Pu production resulting from a harder depletion neutron spectrum. Therefore, a bounding high power is desired for depletion.

The assembly average power for depletion to a particular fuel burnup is chosen to bound the average power an actual assembly could sustain from initial use through the fuel burnup being analyzed (the burnup averaged assembly power). Burnup averaged nodal fuel and moderator temperatures are calculated for the depletion analysis using the highest burnup averaged relative assembly power (BARAP).

The BARAP at the end of each cycle is the accumulated assembly burnup divided by the sum of the cycle burnups for all cycles the assembly has resided in the core. For each assembly in MPS3 Cycles 1-17, the burnup at the end of each cycle divided by the accumulated cycle burnup was calculated and plotted in Figure 8.1 versus the assembly burnup. Figure 8.1 also shows a bounding a line for the BARAP as a function of burnup. Table 8.1 shows the

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 91 of 300

breakpoints and values for the bounding (high) BARAP function. This BARAP function will be used as input to calculate depletion power, fuel temperature, and moderator temperature.

Figure 8.1: Bounding Burnup Averaged Relative Assembly Power

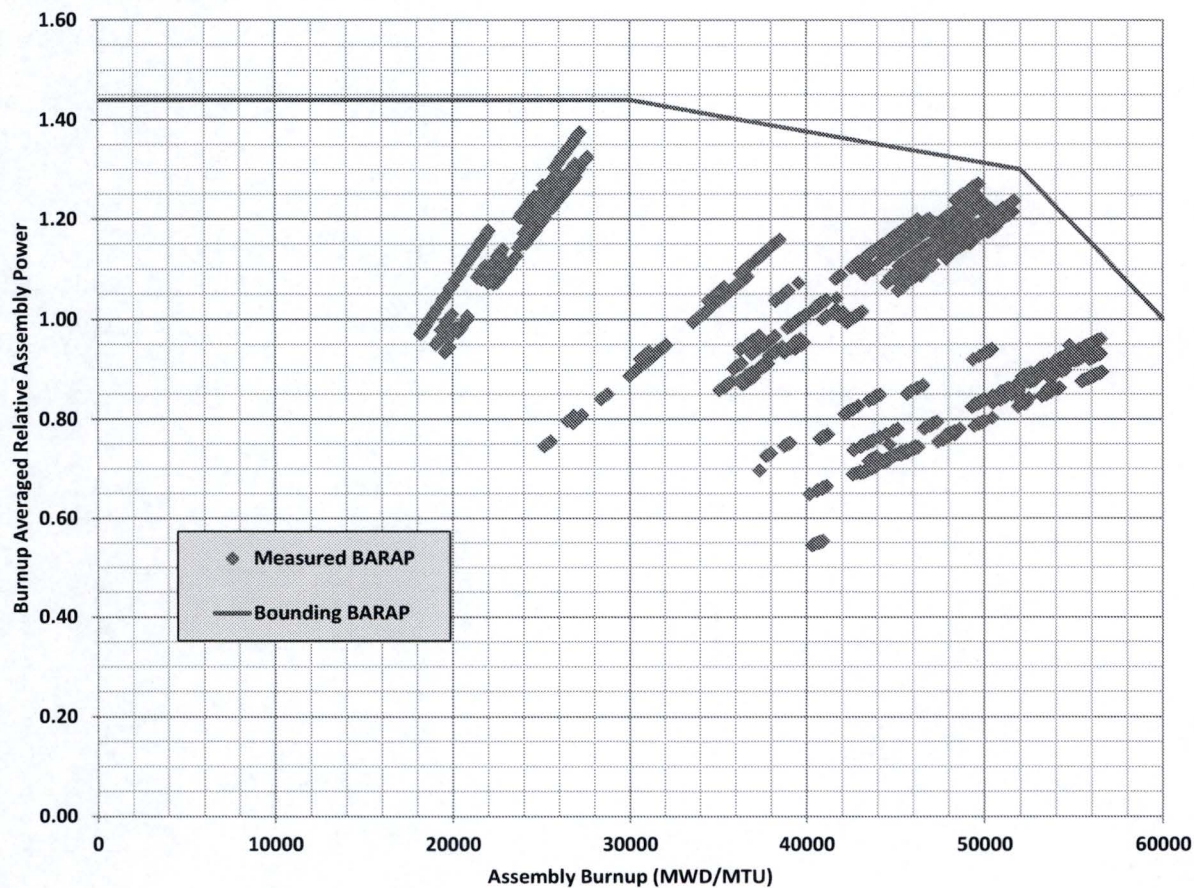


Table 8.1: Bounding Burnup Averages Relative Assembly Power versus Burnup
(Linearly interpolate for intermediate points)

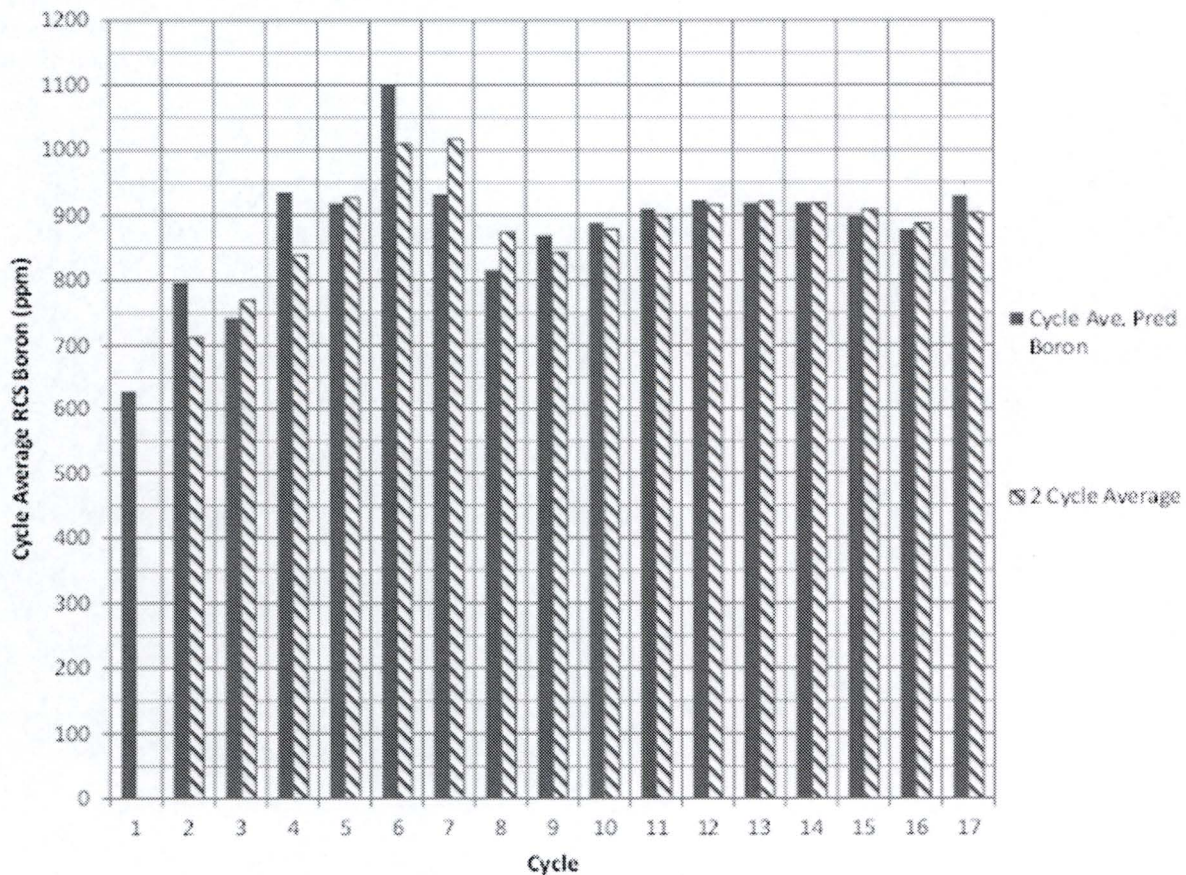
Burnup (MWD/MTU)	RPD
0	1.44
30000	1.44
52000	1.3
60000	1

8.3 Bounding Depletion Boron

Spent fuel reactivity is increased by depletion at higher soluble boron. Cycle average soluble boron concentration is calculated by trapezoidal integration of boron versus cycle burnup for cycles 1 through 17.

Figure 8.2 shows that a boron concentration of 1050 ppm bounds all two cycle averages and all but one single cycle average. Cycle 6 was a 21 month cycle intended to be a transition to 24 month cycle and is not expected to be repeated. Fuel depleted in Cycle 6 was also depleted in at least one other cycle, so depletion with 1050 ppm boron bounds all assemblies depleted at MPS3 as well as expected future cycle average boron.

Figure 8.2: MPS3 Bounding Cycle Average Soluble Boron



8.4 ***Bounding RCS Temperature***

Higher depletion moderator temperature produces more plutonium and increases burned fuel k-eff in the SFP. Calculation of the bounding high Reactor Coolant System (RCS) temperature is performed using a simple heat balance with the input variables: 1) minimum RCS flow (bounds all historical cycles), 2) burnup averaged relative assembly power (maximum from Figure 8.1), 3) core power (maximum of all historical cycles increased by 2% to cover potential future power uprates), 4) a high inlet temperature, and 5) nominal RCS pressure. The RCS flow is further reduced by a high bypass flow.

Rated core power, inlet temperature, RCS flow, and bypass flow data is given on Table 8.2. Average fuel assembly moderator exit temperature is calculated via simple heat balance to determine most limiting historical T/H conditions. Table 8.2 shows the results of the exit temperature calculation. The most limiting cycle is Cycle 16. Exit temperature is taken as the key metric for bounding RCS temperature because the reactivity of depleted PWR fuel is dominated by the top region of the fuel. The following input is used for the bounding depletion moderator temperature and density calculation:

- Core power (3725 MWth; 2% higher than current license)
- Assembly power (Section 8.2)
- RCS pressure (2250 psia)
- Minimum vessel flow of historical cycles (391358 gpm)
- Maximum core bypass flow (5.60% with Cycle 4 as the exception)
- HFP core inlet temperature (556.6 °F)

Table 8.2 shows that the RCS temperature calculation inputs described above produce a bounding core average moderator outlet temperature (2.4 °F higher than any historical cycle). These inputs are expected to remain bounding for future cycles.

The depletion moderator temperature of each of the 18 nodes of a fuel assembly is calculated as follows. The fuel assembly average power for this calculation is BARAP times the core average assembly power. The axial power profile is the appropriate NUREG normalized burnup profile or uniform profile (Section 8.6). With the fuel assembly power and the axial power profile known, the nodal average moderator enthalpy (average of the enthalpy at the upper and lower boundaries of the node) is determined for each of the 18 axial nodes starting from the core inlet and integrating the enthalpy added in each successive node. Moderator temperature is determined from moderator enthalpy using a pressure of 2250 psia.

Table 8.2: RCS Thermal Hydraulic History for MPS3

Cycle	Power (MW)	Pressure (psia)	HFP Inlet Temp (F)	Inlet Density (lbm/ft ³)	Vessel Flow (gpm)	Bypass Flow Fraction	Core Flow (gpm)	RCS Mass Flow (lbm/s)	Inlet Enthalpy (Btu/lbm)	Outlet Enthalpy (Btu/lbm)	Outlet Temp. (F)
1	3411	2250	558.4	46.295	424937	0.056	401141	41376	557.62	635.75	615.46
2	3411	2250	558.4	46.295	413333	0.056	390186	40246	557.62	637.94	616.89
3	3411	2250	558.4	46.295	404620	0.056	381961	39398	557.62	639.67	618.02
4	3411	2250	558.4	46.295	403168	0.076	372527	38425	557.62	641.75	619.37
5	3411	2250	558.4	46.295	391522	0.056	369597	38122	557.62	642.41	619.80
6	3411	2250	558.4	46.295	391358	0.056	369442	38106	557.62	642.45	619.82
7	3411	2250	558.4	46.295	403018	0.056	380449	39242	557.62	639.99	618.23
8	3411	2250	558.4	46.295	400841	0.056	378394	39030	557.62	640.44	618.52
9	3411	2250	558.4	46.295	403263	0.056	380680	39266	557.62	639.94	618.20
10	3411	2250	558.7	46.275	399290	0.056	376930	38862	558.00	641.18	619.00
11	3411	2250	558.7	46.275	398401	0.056	376091	38775	558.00	641.36	619.12
12	3411	2250	558.7	46.275	400661	0.056	378224	38995	558.00	640.89	618.82
13	3650	2250	556.6	46.414	402113	0.056	379595	39255	555.35	643.47	620.47
14	3650	2250	556.6	46.414	400202	0.056	377791	39068	555.35	643.89	620.74
15	3650	2250	556.6	46.414	400167	0.056	377758	39065	555.35	643.90	620.75
16	3650	2250	556.6	46.414	399976	0.056	377577	39046	555.35	643.94	620.77
17	3650	2250	556.6	46.414	400553	0.052	379724	39268	555.35	643.44	620.45
18	3650	2250	556.6	46.414	402787	0.052	381842	39487	555.35	642.95	620.14
Bounding	3725	2250	556.6	46.414	391358	0.056	369442	38205	555.35	647.75	623.19

8.5 Bounding Fuel Temperature

Higher depletion fuel temperature increases depleted fuel reactivity in the SFP by increasing U-238 neutron capture and production of Pu. Calculation of fuel temperature is performed on a nodal basis and is dependent on local specific power, local moderator temperature, and fuel burnup.

Depletion fuel temperatures are determined for each fuel node using the nodal moderator temperature and fuel temperature difference from the moderator temperature, which is dependent on the nodal fuel power and burnup. The fuel temperature data is taken from the SIMULATE-3 model used by Dominion for its licensed fuel management analysis. [10] Fuel temperatures are pellet average, which bound the resonance effective fuel temperatures. Fuel temperature data (a function of power and burnup) used for this calculation is also integrated to the burnup of interest to obtain an appropriate depletion average temperature rather than a point value at a particular burnup.

Table 8.3 shows the results of the nodal depletion condition calculation for a 20 GWd/MTU fuel assembly modeled using NUREG/CR-6801 [9] axial burnup shape 8 (Section 8.6).

Table 8.3: Example of Nodal Depletion Conditions (20 GWd/MTU)

Node	Burnup (GWd/MTU)	Specific Power (MW/MTU)	Fuel Temp. (K)	Moderator Temp. (K)	Moderator Density (g/cc)
1 (Bottom)	13.4	40.7	905	566	0.741
2	20.7	63.0	1031	568	0.736
3	23.0	70.0	1082	572	0.729
4	21.9	66.6	1062	575	0.722
5	21.1	64.1	1048	579	0.715
6	21.0	63.8	1050	582	0.708
7	21.3	64.8	1059	585	0.700
8	21.9	66.7	1075	588	0.693
9	22.4	68.3	1089	591	0.685
10	22.7	69.1	1098	595	0.676
11	22.8	69.4	1104	598	0.668
12	22.8	69.3	1106	601	0.659
13	22.6	68.8	1105	604	0.650
14	22.1	67.3	1098	606	0.641
15	21.0	63.9	1077	609	0.632
16	18.7	56.8	1037	611	0.623
17	13.4	40.7	952	613	0.616
18 (Top)	7.5	22.7	817	614	0.612

8.6 ***Bounding Axial Burnup Profiles***

DSS-ISG-01-2010 [4] provides guidance on use of axial burnup profiles. Consistent with this guidance, MPS3 SFP burnup credit analysis uses NUREG/CR-6801 [9] profiles within their respective burnup ranges as well as uniform profiles at low burnups.

Reduced enrichment axial blankets were used in new fuel batches after Cycle 2. These axial blankets are conservatively ignored. NUREG/CR-6801 concludes, "*Because the end effect for assemblies with low enrichment axial blankets is typically very small or negative, this approach [using the non-blanketed shapes for axial blanketed fuel] will bound those assemblies.*"

DSS-ISG-01-2010 also directs that applications for plant designs that set the limiting profiles in NUREG/CR-6801 should provide a site specific justification for the axial burnup distributions. The NUREG/CR-6801 bounding profile for burnup group 1 (> 46 GWd/MTU) was set by MPS3 profiles. The authors of NUREG/CR-6801 note that the MPS3 profiles were from un-blanketed fuel assemblies in cycles that underwent a transition to fuel with axial blankets. The un-blanketed fuel burnup profiles were influenced by neighboring blanketed fuel. Use of the group 1 shape for MPS3 burnup credit analysis is justified for these reasons:

- 1) The transition shapes are represented in the NUREG/CR-6801 database.
- 2) The transition shapes are from cycles with natural enrichment axial blankets. Natural enrichment blankets affect the burnup shape more than low enriched blankets, so the transition effects represented in the data can be considered bounding.
- 3) There has only been one transition to axial blankets at MPS3, which is represented in the data.

Specific NUREG/CR-6801 burnup profiles used for the MPS3 analysis are listed in Table 9.17 (Region 2) and Table 9.27 (Region 3).

8.7 Burnable Absorbers

Two types of burnable absorber have been used at MPS3. Pyrex BP was only used for Cycles 1 (24 fingers) and 2 (8 fingers) while IFBA has been used for the remaining cycles. WABA is considered in this analysis to allow for potential future use. Although MPS3 Pyrex BP had no absorber cutback (a region near the top and bottom of the fuel assembly with no burnable absorber), cutback regions are typical for modern core designs. MPS3 IFBA has had at least 11 inches of cutback at each end.

Pyrex BP, WABA, and IFBA absorb low energy neutrons and harden the depletion neutron spectrum, increasing spent fuel reactivity. Pyrex BP and WABA also displace moderator in the guide tubes, which further hardens the depletion neutron spectrum. Region 3 KENO rack model single fuel node (node 16) test cases (Table 8.6) confirm that depletion with maximum WABA (24 rods) bounds depletion with maximum IFBA (200 rods at []^{a,c} loading), even if the IFBA assembly is assumed to contain a six finger secondary source assembly for the entire depletion. MPS3 has used 4 and 6 fingered source assemblies. Source assemblies cannot be used in combination with WABA or Pyrex BP.

Table 8.6 KENO rack model cases take no credit for residual poisons or water displacement of the WABA in the SFP. Table 8.6 cases are run with 1050 ppm soluble boron during depletion and with no soluble boron in the SFP.

Table 8.6: Depletion with Different Burnable Absorbers (5.0 wt%, 54 GWD/T)

Case Description	SFP k-eff	σ	Δk
24 WABA w/Zr grids	0.93663	0.00006	N/A
8 Pyrex w/Zr grids	0.92927	0.00006	-0.00736
24 Pyrex w/Inconel grids	0.93807	0.00006	0.00144
IFBA w/secondary sources and Zr grids	0.93342	0.00006	-0.00322

Depletion with 24 Pyrex BP results in somewhat higher SFP reactivity than depletion with 24 WABA. However, only Cycle 1 fuel (Batches 2 and 3) are affected by this issue. Batch 2 and 3 fuel is dispositioned in the Region 2 and Region 3 SFP analyses. Depletion analysis for MPS3 is performed using 24 WABA which are left in the fuel assembly for all burnups. The WABA is conservatively assumed to have no absorber cutback.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 99 of 300

MPS3 has not had an assembly with IFBA and another burnable absorber at the same time. However, other plants have used IFBA and WABA simultaneously. Depletion with maximum WABA, which has been shown to bound depletion with maximum IFBA, is the standard configuration for MPS3 burnup credit analysis. To assess the effect of IFBA and WABA in combination, test cases are provided comparing the depletion effect of maximum WABA with that of maximum IFBA with varying numbers of WABA rods.

The burnups chosen for the test cases are 20 and 54 GWD/T which represent an assembly after one cycle of operation and after three cycles of operation. The fuel enrichments (2.6 and 5.0 wt% respectively) are consistent with the Region 3 burnup credit curve (Section 9.9). A Region 3 KENO rack model is used for the comparison. For these cases, fuel is represented by Node 16 isotopic content. Node 16 is the third node from the top of the fuel and is in a neutronically important axial location for depleted fuel.

Table 8.7 shows the results of the KENO comparison cases. For both burnups evaluated, the depletion with 24 WABA results in higher SFP k-eff than depletion with maximum IFBA and 4, 8, or 12 WABA. The 20 GWD/MTU cases credit 50% of the TRITON calculated residual IFBA content (less than 1% of the initial B-10 loading). The Node 16 results are not only representative of the whole fuel assembly, but are also conservative in that the Node 16 model does not credit the effect of the IFBA or WABA cutback.

Table 8.7: Depletion with IFBA and WABA Simultaneously (Node 16)

Case Description	SFP k-eff	σ	Δk
24 WABA Rods, 0 IFBA Pins 2.60 wt%, 20 GWD/T	0.96715	0.00006	N/A
4 WABA Rods, 200 IFBA Pins 2.60 wt%, 20 GWD	0.96204	0.00006	-0.00511
8 WABA Rods, 200 IFBA Pins 2.60 wt%, 20 GWD	0.96437	0.00006	-0.00278
12 WABA Rods, 200 IFBA Pins 2.60 wt%, 20 GWD	0.96686	0.00006	-0.00028
24 WABA Rods, 0 IFBA Pins 5.00 wt%, 54 GWD/T	0.93663	0.00006	N/A
0 WABA Rods, 200 IFBA Pins 5.00 wt%, 54 GWD/T	0.92780	0.00006	-0.00883
4 WABA Rods, 200 IFBA Pins 5.00 wt%, 54 GWD/T	0.93027	0.00006	-0.00636
8 WABA Rods, 200 IFBA Pins 5.00 wt%, 54 GWD/T	0.93258	0.00006	-0.00405
12 WABA Rods, 200 IFBA Pins 5.00 wt%, 54 GWD/T	0.93484	0.00005	-0.00179

8.8 Control Rod History

Depleting fuel assemblies with control rods inserted hardens the neutron spectrum and increases depleted fuel reactivity. Only fuel assemblies in lead control bank locations (5 of 193 core locations) accumulate significant rodded exposure. Control rod insertion is mutually exclusive with any other insert component (Pyrex BP, sources, WABA). Control rod dimensions are listed in Table 8.8.

Table 8.8: Control Rod Model

Dimension	Value
Absorber Pellet O.D. (cm)	0.8661
Clad I.D. (cm)	0.874
Clad O.D. (cm)	0.9677
End Plug Length (cm)	2.8575
Absorber Pellet Material 1 (Cycles 1 – 4)	95.5 w/o Hf, 4.5 w/o Zr*
Absorber Pellet Material 2 (Cycles 4 – Present)	80 w/o Ag, 15 w/o In, 5 w/o Cd
Clad Material	SS-304
Step Size (cm/step)	1.5875

Assessment of the effect of rodded depletion history is performed to determine whether the standard maximum WABA depletion bounds depletion scenarios involving control rod insertion. In particular, fuel assemblies can have IFBA and control rod insertion.

Figure 8.5 shows the cycle average control rod insertion of lead bank (D bank) fuel assemblies and the average insertion of the remaining control rods. Except for five fuel assemblies in Cycle 1 and five fuel assemblies in Cycle 4, 10 steps average insertion (6.25 inches) bounds the most rodded fuel assemblies in each cycle. Cycle 1 and 4 lead bank assemblies are justified in the Region 3 analysis (Section 9.9.8). Average lead bank insertion has been well below 10 steps for 10 cycles and is expected to remain so in future cycles.

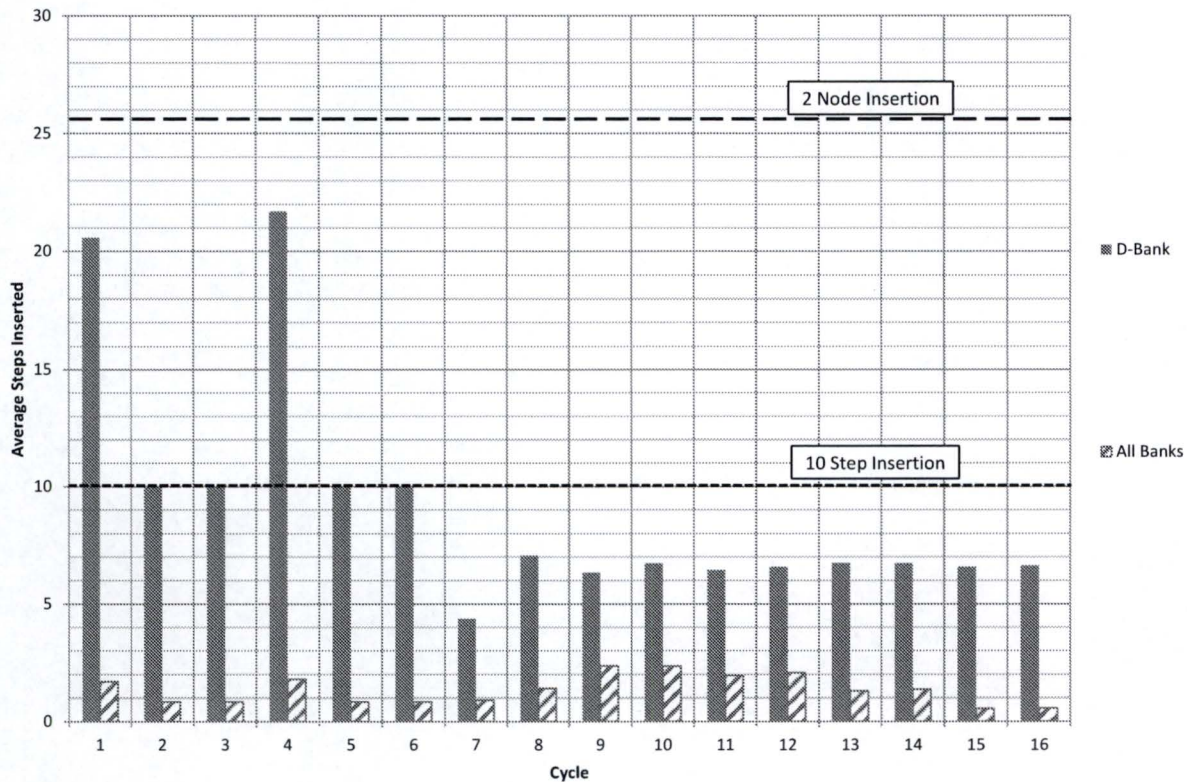
Figure 8.5: Cycle Average Control Rod Insertion

Table 8.9 contains results of Region 3 KENO rack model cases combining control rod and IFBA (200 rods) depletion history. MPS3 IFBA has had an un-poisoned cutback region of at least 11 inches on the top and bottom of the active fuel. For Table 8.9 cases, the depletion is performed with IFBA in nodes 2-17 but not in Nodes 1 and 18 (1 node = 8 inches). Sensitivity cases are provided at 20 and 54 GWd/MTU. WABA and control rods are assumed to remain inserted throughout the depletion.

Table 8.9 results illustrate several points:

- Depleting with a Ag-In-Cd control rod is slightly more limiting than a Hf control rod.
- Depleting with 2 nodes of control rod insertion is more limiting than depleting with full length WABA.
- Depleting with full length WABA bounds depleting with 200 IFBA pins plus a control rod insertion of 10 steps at both 20 and 54 GWD/MTU.

Table 8.9: Control Rod Use Sensitivity Cases

Case Description	Enrich / BU	Reg 3 k-eff	σ	Δk
Maximum WABA	2.60 wt% 20 GWD/T	0.96300	0.00006	N/A
Ag-In-Cd Control Rod, 10 Step* Inserted & 200 IFBA	2.60 wt% 20 GWD/T	0.96150	0.00006	-0.00150
Maximum WABA	5.00 wt% 54 GWD/T	0.95081	0.00006	N/A
Ag-In-Cd Control Rod, 1 Node Inserted & 0 IFBA	5.00 wt% 54 GWD/T	0.95090	0.00007	0.00009
Hafnium Control Rod, 1 Node Insertion & 0 IFBA	5.00 wt% 54 GWD/T	0.95011	0.00007	-0.00070
Hafnium Control Rod, 2 Node Inserted & 0 IFBA	5.00 wt% 54 GWD/T	0.97087	0.00007	0.02006
Ag-In-Cd Control Rod, 10 Step* Inserted & 200 IFBA	5.00 wt% 54 GWD/T	0.95070	0.00006	-0.00011

*10 step cases have 19 axial nodes. The bottom 17 nodes are modeled with the standard methodology. The top node is split up into a rodged section (6.25 inches) and an un-rodged section (1.75 inches).

These cases confirm that the standard full length maximum WABA depletion is acceptable for modeling fuel assemblies, with the exception of lead bank assemblies in Cycles 1 and 4.

8.9 In-core Thimble

MPS3 has in-core instrument thimbles inserted in the fuel assembly center instrument tube in 58 of 193 assembly locations. The in-core thimble is a hollow, thin wall tube that displaces water and hardens the neutron spectrum during depletion and increase k-eff in the SFP. Table 8.10 shows the relevant dimensions of the in-core thimble and instrument tube. The in-core thimble is included in the TRITON models as part of the standard depletion. This is conservative because the instrument thimble is present in only 58 of 193 fuel assemblies in each core.

Table 8.10: In-Core Thimble and Instrument Tube Dimensions

Dimension	Value
I.T. Inner Diameter (cm)	1.123
In-Core Thimble Housing I.D. (cm)	0.5105
In-Core Thimble Housing O.D. (cm)	0.7595
In-Core Thimble Housing Material	ASME Type 316L SS

8.10 *Reduced Power before Storage*

Bounding high power history is used to maximize spent fuel reactivity. Higher power increases fuel and moderator temperature during depletion. However, depleting at low power produces less Pm-149, which has the effect of reducing Sm-149 peaking after fuel discharge. Reducing Sm-149 in SFP fuel increases k-eff. Therefore, high power history combined with reduced power near end of life (EOL) can result in higher k-eff in the SFP. Pm-149 has a 2.2 day half life, so reduced power for at least several days at EOL is required for substantial reduction in SM-149 peaking.

There are two plausible ways to achieve low power near EOL for sufficient duration: (1) Fuel placement in a low power location during the last cycle, and (2) power coastdown. A final cycle at low power would reduce fuel assembly burnup average moderator and fuel temperature substantially, which would tend to reduce fuel reactivity. Power coastdown would have far less impact on burnup average moderator and fuel temperature because of the relatively short duration. Power coastdown below 50% power is not likely due to economic considerations.

Simulation of power coastdown is performed by depleting at high power conditions for all but the last 20 days of depletion. Twenty days of operation is more than sufficient for Pm-149 to reach a reduced equilibrium level. For the last 20 days, depletion proceeds at 50% of the prior power (40 days at reduced power) retaining high power fuel and moderator temperatures for simplicity and conservatism. Table 8.11 shows the typical effect that reduced power near EOC has on depleted fuel reactivity. The cases shown use Node 16 isotopic content to represent the fuel in a Region 3 KENO rack model.

Table 8.11: Low Power EOC Effect on SFP Reactivity (5.0 w/o, 54 GWD/T)

Case Description	SFP k-eff	σ	Δk
Low Power EOL	0.93663	0.00006	N/A
Nominal Power EOL	0.93360	0.00006	-0.00303

8.11 *Grid Growth and Clad Creep Depletion Effects*

Zircaloy based grids can experience irradiation growth with increasing fuel burnup. Grid growth can increase fuel pin pitch. Because fuel pins are constrained within the grid lattice, fuel pin pitch growth is assumed to be the same as grid growth. Fuel clad diameter tends to decline due to creep during fuel depletion prior to increasing again at high fuel burnup.

Expanding the pin pitch and reducing clad diameter during fuel depletion would soften the depletion neutron spectrum. To simplify the depletion model, grid expansion and clad creep down are conservatively ignored in the TRITON depletions but are analyzed in the spent fuel pool KENO models.

8.12 *Grids*

Grids displace water and harden the neutron spectrum during depletion, increasing k-eff in the SFP. In the SFP, grids displace water and may decrease k-eff, depending on the boron concentration. These effects are confirmed using a Region 3 SFP single fuel node KENO rack model. In the TRITON depletion model and in the KENO rack model, Zr based grids are represented by homogenizing the grid and the water in the fuel lattice over the length of the fuel. Water inside the guide tubes are modeled as pure water. The total grid volume and water displacement is conserved.

Table 8.12 shows that with no soluble boron it is conservative to use the minimum Zr grid volume in the spent fuel pool KENO model. Table 8.13 shows that it is conservative to use maximum Zr grid volume in the TRITON depletion model.

Table 8.12: SFP Grid Modeling Effect (Node 16, Fresh Fuel, 0 ppm soluble boron)

Case Description	SFP k-eff	σ	Δk
5.00 w/o, 0 GWD/T, Minimum Zr Grid	1.25414	0.00007	N/A
5.00 w/o, 0 GWD/T, Maximum Zr Grid	1.25328	0.00007	-0.00085

Table 8.13: Effect of Depleting with Grids (Node 16, 0 ppm soluble boron)

Case Description*	SFP k-eff	σ	Δk
5.00 w/o, 54 GWD/T, Min. Zr Grid Depletion	0.93561	0.00006	-0.00102
5.00 w/o, 54 GWD/T, Max. Zr Grid Depletion	0.93663	0.00006	N/A

*Both cases have minimum Zr grids in the KENO rack model.

8.13 Instrument and Guide Tube Design

The difference in guide tube and instrument tube dimensions between MPS3 fuel designs is small. Table 8.14 shows the instrument and guide tube dimensions for each fuel type. The RFA-2 guide thimble displaces the most water and will tend to increase spent fuel reactivity during depletion due to spectrum hardening. Therefore, the RFA-2 instrument and guide tube dimensions are used in the TRITON depletion model.

The V5H guide thimble displaces the least amount of water. For the pool model, guide tubes that displace the least amount of water will increase the reactivity because the fuel lattice is under moderated. Therefore, the Region 1, 2, and 3 pool models use the V5H guide tube dimensions.

Table 8.14: Instrument and Guide Tube Dimensions

Fuel Type	Inner Diameter (cm)	Outer Diameter (cm)	Cross Section Area (cm ²)
Standard	1.143	1.224	0.1506
V5H	1.123	1.204	0.1480
RFA-2	1.123	1.224	0.1862

8.14 Comparison of CASMO and TRITON Depletion Reactivity

The information in this section was previously submitted for NRC review in a North Anna License Amendment Request [23] and is copied below. The analysis was not rerun for MPS3 because the North Anna and MPS3 analyses use the same fuel design and computer codes.

A depletion reactivity uncertainty of 5% of depletion worth (k of fresh fuel – k of depleted fuel; no credit for burnable poison worth) has been accepted based on a conservative estimate of the state of the art of fuel management analysis computer codes. Since SCALE has not been used for fuel management, a study has been performed to compare the delta k depletion predicted using the TRITON t5-depl sequence of SCALE to CASMO-4 [10], as well as CASMO-5. [24] The study shows that using TRITON atom densities is more conservative than using CASMO atom densities. Similar results were found when CASMO was compared to TRITON for Millstone 2. [14] Section 6.2 describes the analysis.

To confirm that TRITON is acceptable for depletion analysis for North Anna, representative comparisons are provided for North Anna (17x17 fuel) Region 2 (all cells loaded) fuel rack k using TRITON, CASMO-4 and CASMO-5 to generate depleted fuel isotopic content. The same conservative depletion conditions were used in all 3 codes, including 24 BPRA, 1100 ppm soluble boron, and high moderator and fuel temperature. A single node (Node 15; nodes are numbered from 1 at the bottom of fuel to 18 at the top) was used as a reasonable representation of the fuel for this comparison. No grids were used in the depletion models for simplicity. Two enrichment and burnup combinations representing a low and high burnup credit requirement for Region 2 are modeled.

For CASMO-4, isotopes common to both CASMO-4 and TRITON (49 nuclides) were used in the KENO rack models. CASMO-5 has no lumped fission products, so all nuclides available in the SCALE standard composition library were retained. All depletions end with 5 days decay after shutdown.

Results of the KENO Region 2 rack k cases are provided in Table 8.15. For 2.45 w/o fuel depleted to 10 GWd/MTU and 49 nuclides, the TRITON depletion produces a SFP rack k approximately 0.008 Δk higher than CASMO-4, and 0.0018 Δk higher than the CASMO-5 depletion. With all available nuclides included, the TRITON case rack k is higher than the CASMO-5 case by 0.00035 Δk .

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 107 of 300

For 5.0 w/o fuel depleted to 44 GWd/MTU and 49 nuclides, the TRITON depletion produces rack k approximately 0.015 Δk higher than the equivalent CASMO-4 depletion. With all available nuclides included, the TRITON case rack k is higher than the CASMO-5 case by 0.0014 Δk .

Table 8.15 results show that depletion with TRITON produces the highest rack k as compared to depletion with CASMO-4 and CASMO-5. TRITON results are much closer to CASMO-5 results, probably because CASMO-5 and TRITON use ENDF/B-VII cross sections and CASMO-4 uses an earlier cross section set. Use of the older CASMO-4 cross section data to produce isotopic content that is then used in a KENO rack model with newer (ENDF/B-VII) cross section data creates a potential mismatch that may explain some of the large difference in rack k results between TRITON and CASMO-4. There is no cross section difference present in the TRITON and KENO models used for the MPS3 analysis.

Table 8.15: Comparison of CASMO and TRITON Depletion Worth

Enrich. (U235 w/o)	Number of Nuclides	Burnup (GWd/MTU)	Depletion Code	Calculated Rack k	Monte Carlo Sigma	Burnup Worth (Δk)
2.45	N/A	0	N/A	1.025768	0.000059	N/A
2.45	49	10	TRITON	0.957684	0.000056	0.0681
2.45	49	10	CASMO4	0.949563	0.000053	0.0762
2.45	49	10	CASMO5	0.955930	0.000054	0.0698
2.45	ALL	10	TRITON	0.951871	0.000053	0.0739
2.45	ALL*	10	CASMO5	0.951523	0.000056	0.0742
5.0	N/A	0	N/A	1.190491	0.000064	N/A
5.0	49	44	TRITON	0.930972	0.000055	0.2595
5.0	49	44	CASMO4	0.916284	0.000053	0.2742
5.0	ALL	44	TRITON	0.914146	0.000052	0.2763
5.0	ALL*	44	CASMO5	0.912796	0.000054	0.2777

*Some minor nuclides not in SCALE 6.0 library

8.15 TRITON Depletion Model Summary

Table 8.16 summarizes the key elements of the MPS3 TRITON depletion model.

Table 8.16: Depletion Parameters for TRITON Depletion Model

Parameter	Value	Reference/Basis										
Assembly pitch	21.50 cm	Section 3.1										
Pellet diameter	0.8192 cm	Section 3.1										
Clad inner diameter	0.8356 cm	Section 3.1										
Clad outer diameter	0.95 cm	Section 3.1										
Clad material	Zirconium	Section 3.1										
Fuel rod pitch	1.2598 cm	Section 3.1										
Guide tube inner diameter	1.123 cm	Section 3.1, 8.13										
Guide tube outer diameter	1.224 cm	Section 3.1, 8.13										
Grid volume fraction (lattice water displacement outside fuel rods and guide tubes)	1.7%	Section 3.1, 8.12										
Fuel pellet net density	95.5%	Section 3.1										
Burnable absorber	24 WABA	Section 8.7										
Inner clad I.D.	0.572 cm	Section 3.2.1										
Inner clad O.D.	0.6782 cm	Section 3.2.1										
BP I.D.	0.706 cm	Section 3.2.1										
BP O.D.	0.808 cm	Section 3.2.1										
Outer clad I.D.	0.8357 cm	Section 3.2.1										
Outer clad O.D.	0.968 cm	Section 3.2.1										
Clad material	Zirconium	Section 3.2.1										
Absorber loading	[] ^{a,c}	Section 3.2.1										
Absorber material	B ₄ C	Section 3.2.1										
Absorber material density	[] ^{a,c}	Section 3.2.1										
Bounding assembly average depletion power	<table><tr><th>Burnup (MWD/MTU)</th><th>BARAP</th></tr><tr><td>0</td><td>1.44</td></tr><tr><td>30000</td><td>1.44</td></tr><tr><td>52000</td><td>1.3</td></tr><tr><td>60000</td><td>1</td></tr></table>	Burnup (MWD/MTU)	BARAP	0	1.44	30000	1.44	52000	1.3	60000	1	Section 8.2
Burnup (MWD/MTU)	BARAP											
0	1.44											
30000	1.44											
52000	1.3											
60000	1											
RCS boron concentration	1050 ppm	Section 8.3										
Core average thermal power	3725 MW	Section 8.4										
Core inlet temperature	556.6°F	Section 8.4										
RCS pressure	2250 psia	Section 8.4										
Vessel flow	391358 gpm	Section 8.4										
Bypass flow fraction	0.056	Section 8.4										
Bounding moderator temperature	Varies	Section 8.4										
Bounding fuel temperature	Varies	Section 8.5										

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 109 of 300

Axial burnup shapes	Uniform and NUREG/CR-6801 shapes	Section 8.6
In-core thimble material	316L Stainless Steel	Section 8.9
In-core thimble I.D.	0.5105 cm	Section 8.9
In-core thimble O.D.	0.7595 cm	Section 8.9
Decay time	5 days or decay time credit	Section 5.2.4,5.2.5,6.1
Reduced power at EOL	50% reduction for last 40 days	Section 8.10
Volatile fission products	Select nuclides reduced	Section 6.1.2

9 Spent Fuel Rack Analysis

9.1 Region 1 Analysis

9.1.1 Region 1 Fuel Storage

A fuel assembly may be placed in Region 1 if it meets one of the following criteria:

- 1) The enrichment is ≤ 4.75 wt% U-235.
- 2) The enrichment is ≤ 5.0 wt% U-235 and the fuel assembly contains 12 or more IFBA rods.
- 3) The enrichment is ≤ 5.0 wt% U-235 and the assembly average burnup is ≥ 2 GWd/MTU.
- 4) The enrichment is ≤ 5.0 wt% U-235 and the assembly is stored in one of the two rows of rack cells adjacent to the West SFP wall.

Criteria 1-3 are confirmed by analysis using an infinite lattice 6x6 rack cell model. Criterion 4 analysis uses a finite model described in Section 9.3.

9.1.2 Region 1 Modeling Assumptions

These modeling simplifications and assumptions are used in the Region 1 analysis:

- a) The plenum, guide tubes above and below the active fuel, and top and bottom nozzles are modeled as water. This practice was investigated as part of the Millstone Unit 2 criticality analysis in Ref. 14 Section 2.6 and found to be acceptable.
- b) The storage cell is modeled at its nominal length. The rack structure below the bottom of the storage cell tube is modeled as water. A water reflector extends above the storage tube at the top of the model.
- c) Neither reduction in fuel enrichment nor annular pellets are modeled for axial blankets. This is conservative because it increases SFP k-eff.
- d) BORAL is conservatively modeled as the same length as fuel. Table 4.4 shows that using nominal dimensions, the BORAL (using minimum BORAL length) extends well below the fuel in current and prior fuel designs and extends slightly above the fuel in current and prior fuel designs. This obviates the need for fuel axial position tolerance and BORAL length tolerance.
- e) The BORAL wrapper length is modeled equal to the active fuel stack length. BORAL wrappers actually extend beyond the upper and lower boundaries of the fuel stack. Since the region above and below the fuel is insensitive (item b above) and the un-modeled BORAL is a strong neutron absorber, there is no need for a wrapper length tolerance.

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 111 of 300

- f) Zr-based clad materials are modeled as Zr.
- g) BORAL clad thickness is approximated. BORAL clad thickness for thickness tolerance cases is conservatively assumed to be 0 (no clad) and thick enough to fill the design wrapper space (maximum).
- h) Water density is determined assuming the pressure is 15 psia.
- i) Grids are represented by homogenizing the grid and the water in the fuel lattice over the length of the fuel. Water inside the guide tubes are modeled as pure water.

9.2 *Region 1 Infinite Lattice KENO Model*

Drawings and dimensions of Region 1 fuel racks are provided in Section 4.2.1. The primary KENO Region 1 model is a 6x6 model with reflective axial and periodic X-Y boundary conditions. The actual rack baseplate protrudes past the outer cell walls such that storage cell pitch across the rack to rack interface is larger than the normal rack cell pitch. Figure 9.1 shows a 2-D representation of the Region 1 model. Figure 9.2 shows a 3-D cutaway representation of the Region 1 model. Fuel is stored in all cells of the infinite lattice model.

A second Region 1 model used to confirm that neutron leakage at the SFP wall is sufficient to support storage of un-poisoned 5.0 wt% fuel is detailed in Section 9.3.

Figure 9.1: X-Y Representation of Region 1 KENO Model

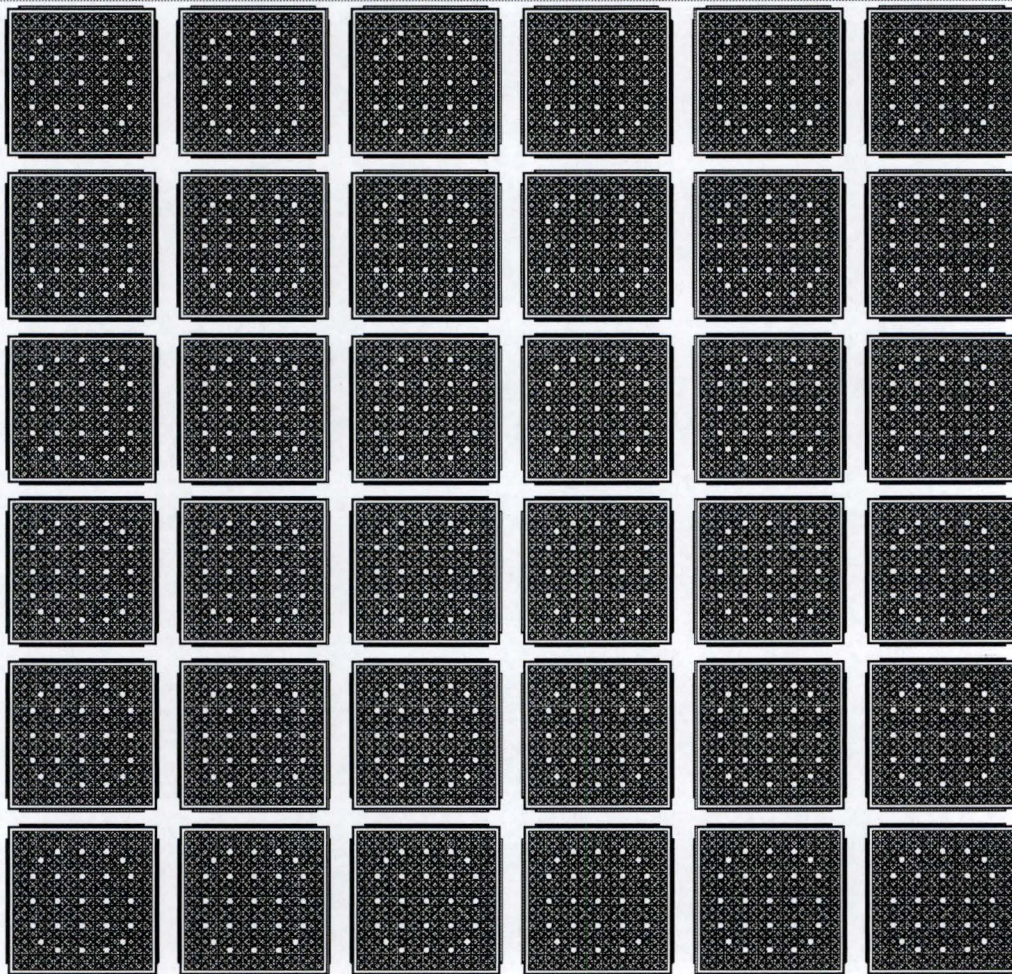
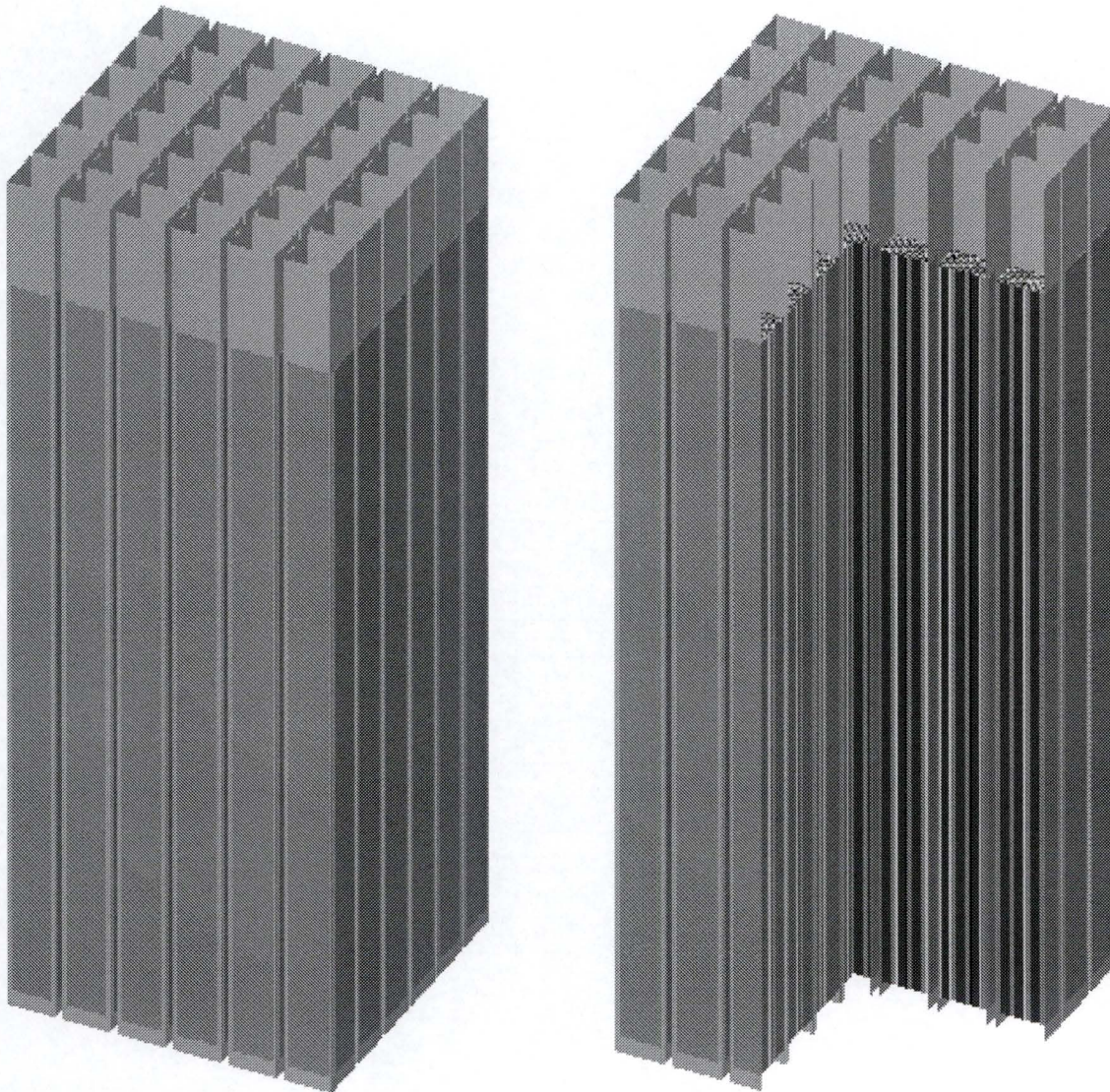


Figure 9.2: 3D Representation of Region 1 KENO Model (full model and cutaway)



9.2.1 BORAL Blisters

BORAL blisters have been observed in surveillance coupons and in rack cells [34, 35]. The size of blisters (depth and diameter) appears to be a partial function of manufacturer (particularly manufacture date). No blisters have been observed in the MPS3 coupons.

To approximate the effect on rack k-eff of possible future blistering of BORAL in the Region 1 racks, full panel blisters are modeled in the KENO rack model by replacing the water surrounding the BORAL in the wrapper space (nominally []^H, approximately the thickness of BORAL clad) with void. This approach is conservative because full panel blisters have not been observed and because no blisters have been observed on the MP3 coupons. The modeled blister thickness is only about 20% of that found in some coupons [36], however, the blister model covers the entire surface of all BORAL panels and the BORAL sheathing tends to limit blister thickness.

9.2.2 SFP Normal Operation Water Temperature and Density

Rack k-eff calculations are performed at four water temperatures (Table 9.1). Water density is calculated assuming atmospheric pressure.

Table 9.1: SFP Normal Operation Water Temperature and Density

Temp. K	Temp. F	Density
273.15	32	1.000
293.15	68	0.998
316.48	110	0.991
338.71	150	0.980

9.2.3 Simplified TRITON Depletion Input for Burnup Credit

Because the maximum Region 1 burnup credit is very small (<2 GWd/MTU), a uniform axial burnup shape is appropriate. The small burnup requirement also makes nodal variations in depleted fuel isotopic content unimportant. To simplify the Region 1 model, a single fuel composition for the entire fuel stack is used. To ensure conservatism, the TRITON depletion uses the highest nodal moderator temperature (lowest moderator density) and highest nodal fuel temperature of the 18 nodes. Depletion soluble boron is arbitrarily increased to 2000 ppm to bound the at-power boron concentration near beginning of cycle.

The following non-standard TRITON input is used:

Moderator temp.:	613.5 K
Fuel temperature:	1104 K
Moderator density:	0.6145 g/cc
Specific power:	60.9 MW/MTU
Depletion days:	32.8
Soluble Boron:	2000 ppm

Depletion steps for this very short depletion are reduced to a maximum of 9.4 days to more accurately capture relatively rapid isotopic changes in fuel content. A 5 day decay step is included.

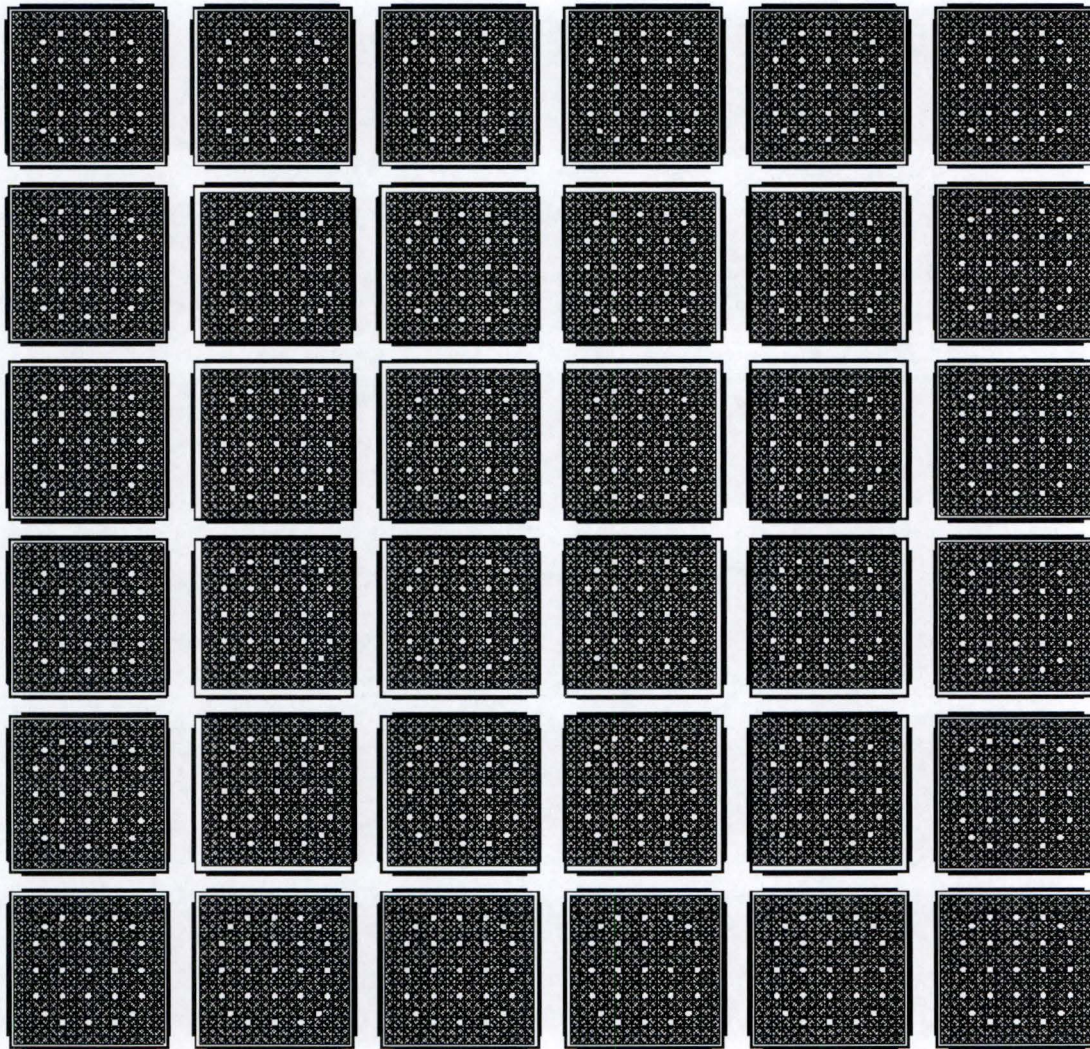
9.2.4 IFBA Credit

IFBA credit cases include 12 fuel rods with IFBA coating. A 12 inch IFBA cutback region is modeled at the top and bottom of the fuel. IFBA B-10 content is reduced 10% below the lower of the two standard IFBA product loadings (Table 3.4) for conservatism. The minimum number of IFBA per assembly for fuel containing any IFBA at MPS3 is 32.. Although no standard lattice pattern exists for 12 IFBA, IFBA pins were modeled in a symmetric pattern adjacent to guide tubes as is typical of other IFBA patterns.

9.2.5 Asymmetric Fuel Placement

Asymmetric placement of fuel in the storage cell can increase k-eff. The Region 1 analysis determines the effect of asymmetric placement by co-locating 16 fuel assemblies toward the center of the model into the corner of each storage cell in the 4x4 central region of the 6x6 Region 1 model. Figure 9.3 shows the orientation. Test cases showed that fuel assemblies centered in their storage cell maximizes k-eff.

Figure 9.3: Asymmetric Fuel Placement (4x4 in a 6x6 model)



9.2.6 Code Validation Bias and Uncertainty

Validation of SCALE/KENO using ENDF/B-VII 238 group cross sections identified EALF as the primary bias and uncertainty trend. Region 1 KENO cases have $EALF \leq 0.35$ eV. The applicable bias and uncertainty for fresh fuel and depleted fuel (MOX validation cases) are shown in Table 9.2.

Region 1 rack k-eff cases at normal operating temperatures show that temperatures below room temperature are clearly limiting, therefore no code temperature bias is necessary.

To cover gaps in the validation for minor actinides and fission products, a bias of 1.5% of the minor actinide and fission product worth is added. [13]

Table 9.2: Validation Bias and Uncertainty ($EALF \leq 0.35$ eV, Appendix A)

Validation cases	Bias (dk)	Uncertainty (dk)
UO2	0.0034	0.0048
MOX*	0.0018	0.0088

*The MOX values are for $EALF \leq 0.4$ eV and are conservative for Region 1 analysis.

9.2.7 Measured Burnup Uncertainty

Measured fuel burnup and as-built fuel enrichment are used to qualify fuel for storage when burnup credit is required. Measured burnup uncertainty is included in the development of the burnup curves. Fuel burnup is the time integral of specific fuel power. Measured burnup uncertainty includes these effects:

- 1) Assembly relative power measurement uncertainty (flux maps)
- 2) Uncertainty for unmonitored assemblies (flux maps)
- 3) Calorimetric power uncertainty (core power)
- 4) Fuel assembly loading uncertainty (MTU per assembly)
- 5) Uncertainty from time integration of measured power (adequacy of the number of power vs time points).

A conservative estimate of assembly relative power measurement uncertainty is the peak pin power nuclear uncertainty factor. For MPS3 a value of 2.8% was determined using the CASMO-4 and SIMULATE-3 core design models. This value is conservative as an estimate for measured fuel assembly power uncertainty because it includes the effect of assembly power prediction uncertainty, which is not a component of measured burnup uncertainty. It is also conservative in that it includes peak pin predictive uncertainty, which is not a component of

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 118 of 300

measured burnup uncertainty. However, this value is only applicable to monitored fuel assemblies, roughly 30% of fuel assemblies in any given cycle.

The additional uncertainty for unmonitored assemblies is calculated using MPS3 flux maps to determine the change in measured assembly relative power caused by changing each unique monitored location from monitored to non-monitored one assembly at a time. Based on 681 measured test cases, the difference between non-monitored and monitored relative power is $0.15\% \pm 1.22\%$ with a -1.99% lower 95/95 tolerance limit. Therefore, the uncertainty of the flux map's unmonitored locations will be 2.0%.

MPS3 calorimetric uncertainty is $\pm 1.7\%$ RTP at present. In the event of a measurement uncertainty recapture update, it would be substantially lower. Fuel assembly loading variation within each batch (MTU per assembly) is too small to affect the total uncertainty if included by RSS. For example, for MP3 cycle 18 the maximum UO₂ weight for any fuel assembly is 1069 lbs and the minimum is 1063 lbs. This represents total variation of $\pm 0.3\%$ about the average.

Regarding time integration uncertainty, power distribution measurement is performed at least monthly (nominally 18 times per cycle). Fuel assembly relative power changes gradually through the cycle. Integration of the assembly power over burnup using an 18 point approximation assuming constant power between points introduces relatively little uncertainty. To evaluate the integration uncertainty, 18 point integrated burnups were calculated for 193 fuel assemblies in the Cycle 15 core using SIMULATE-3 predicted assembly power. Integrated burnups were then compared to actual SIMULATE-3 burnups. The average burnup increase error over a cycle is 0.03%. The standard deviation of burnup increase differences is 0.24%. Based on this calculation, a burnup integration uncertainty allowance of 0.5% is used for integration uncertainty.

These uncertainty contributors are assumed to be independent and are combined by root sum square to obtain a burnup measurement uncertainty estimate. The RSS of 2.8%, 2.0%, 1.7%, 0.3%, and 0.5% is 3.9%. A measured burnup uncertainty of 4% is used based on this assessment of uncertainty contributors.

9.2.8 Summary of Bias and Uncertainty

Table 9.3 summarizes the biases and uncertainties incorporated in the Region 1 analysis.

Consideration of early cycle Pyrex BP is not needed because all early cycle fuel has much lower enrichment and much higher burnup than the small amount of burnup credit required for Region 1. Consideration of early cycle depletion with more than 10 steps of D-bank insertion is not needed because all early cycle fuel has lower enrichment and much higher burnup than the small amount of burnup credit required for Region 1. A flux trap tolerance is not directly modeled but is a result of the combined tolerances for cell ID and cell pitch, which are modeled individually. No burnup tilt, clad creep or grid growth cases are run due to the low burnup credit burnup.

Table 9.3: Summary of Region 1 Biases, Uncertainties, and Conservatism

Analysis	Item / Type (B=bias, C=conservatism, U=uncertainty)	Explanation
Depletion	Burnable absorbers	C Bounding BPRA (24 WABA bounds IFBA)
Depletion	Soluble boron	C 2000 ppm is arbitrarily high
Depletion	Fuel and Water Temperature	C Single node with bounding high power, fuel temp., and moderator temperature
Depletion	Specific power	C Includes all core power uprates, bounding high power assembly history
Depletion	Axial burnup shapes	C Uniform (single node due to low burnup)
Depletion	Grids	C Maximum Zr grids
Depletion	Volatile fission products	C Reduced based on gap release fractions method
Depletion	Fuel density	C 95.5% net bounds all fuel batches
Depletion	Power history	C Bounding high constant specific power. No coastdown due to very low burnup credit
Depletion	Incore thimble	C Included in base depletion
Rack k-eff	Fuel density	U 95.5% [] ^{a,c} net bounds all fuel batches
Rack k-eff	Grids (base case)	C Minimum volume Zr grids
Rack k-eff	Boral B-10	C Minimum [] ^H
Rack k-eff	Boral width	U [] ^H
Rack k-eff	Boral thickness	U [] ^H
Rack k-eff	Boral wrapper thickness	U [] ^H
Rack k-eff	Boral wrapper width	U [] ^H (welded end tab removed)
Rack k-eff	Fuel pin pitch	U 0.496 [] ^{a,c}
Rack k-eff	Fuel pellet OD	U 0.3225 [] ^{a,c}
Rack k-eff	Fuel clad ID	U 0.329 [] ^{a,c}
Rack k-eff	Fuel clad OD	U 0.374 [] ^{a,c}
Rack k-eff	Guide tube ID	U 0.442 [] ^{a,c}
Rack k-eff	Guide tube OD	U 0.474 [] ^{a,c}
Rack k-eff	Fuel stack height	U [] ^{a,c}
Rack k-eff	Burnup worth	U 5% of burnup worth
Rack k-eff	Measured burnup	U 4% of burnup worth (Section 9.2.7)
Rack k-eff	Enrichment	U [] ^{a,c}
Rack k-eff	Cell wall thickness	U [] ^H
Rack k-eff	Rack cell pitch	U [] ^H
Rack k-eff	Rack cell ID	U [] ^H
Rack k-eff	Code uncertainty	U Section 9.2.6
Rack k-eff	KENO case uncertainty	U 2 standard deviations
Rack k-eff	Minor actinides + FP	B 1.5% of worth to cover validation gaps (9.2.6)
Rack k-eff	Code bias	B Section 9.2.6
Rack k-eff	Temperature	C Most reactive temp. (32 F) used
Rack k-eff	Eccentric fuel placement	C Most reactive position (centered) used
Rack k-eff	Boral blisters	B Full panel blister 0.009 in. thick
Rack k-eff	NRC admin. margin	B 1% Δk

9.2.9 Region 1 Infinite Lattice k-eff and Margin Calculation

Region 1 KENO infinite lattice model k-eff calculations are performed for fresh 4.75 wt% fuel, 5.0 wt% fuel with 12 IFBA, and 5.0 wt% fuel depleted to 2 GWd/MTU. Uncertainty and bias calculations are only performed for fresh fuel because of the small amount of burnup credit (only a small change in fuel characteristics over 2 GWd/MTU burnup) and the large amount of margin (IFBA credit and burnup credit).

All tolerance cases except the temperature change are performed at 32 °F (1.00 g/cc) with no soluble boron. The 32 °F base case has the highest k-eff of the four temperature cases (0.0085 dk higher than the 150 °F case). Most tolerance cases (cell wall thickness, cell ID, cell pitch, fuel clad OD and ID, pellet OD, BORAL wrapper thickness, guide tube OD and ID, fuel pin pitch, and BORAL thickness) were run bidirectional rather than assume monotonic behavior.

Eccentric positioning cases including fuel skewed toward the model center (Figure 9.3) and skewed into the same cell corner confirmed that the highest k-eff results from fuel centered in each rack cell. As examples, a few of the key KENO results are shown in Table 9.4.

Table 9.4: Selected Tolerance Results for Region 1 (Fresh 4.75 wt% fuel)

Enrichment (w/o U235)	Tolerance	K-eff	Uncert.	Sensitivity (dK)	EALF	Note
4.75	Base case	0.96745	0.00008	N/A	0.32	Base case full model 32F
4.75	Wrapper width	0.96766	0.00008	0.0004	0.32	No wrapper tabs. Used as base case for other tolerances.
4.75	Cell ID+	0.97353	0.00008	0.0061	0.31	Increase cell ID
4.75	Cell Pitch-	0.97287	0.00008	0.0054	0.32	Decrease cell pitch
4.75	150F	0.95896	0.00008	-0.0085	0.35	Increase temperature
4.75	Asymmetry	0.96721	0.00008	-0.0002	0.35	Eccentric position

Calculation of the burnup worth for the calculation of depletion uncertainty (5% of burnup worth) and measured burnup uncertainty (4% of burnup worth) is demonstrated in Table 9.5. An additional case crediting only the major actinides is also shown to demonstrate the calculation of minor actinide and fission product worth needed for the validation bias (1.5% of worth) calculation. Nuclides included in the major actinides case depleted fuel composition are U-234, U-235, U-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, and O-16.

Table 9.5: Worth of Burnup and Minor Actinides and Fission Products

Enrichment (w/o U235)	Tolerance	K-eff	Uncert.	Sensitivity (dK)	EALF	Note
5.0	N/A	0.97646	0.00008	N/A	0.33	Fresh fuel base case for burnup worth
5.0	2 GWd/MTU	0.95476	0.00008	0.0219	0.35	Burnup worth 2 GWd/T
5.0	2 GWd/MTU	0.97292	0.00008	0.0184	0.34	Minor actinide+FP worth

Bias and uncertainty values resulting from KENO cases include the difference in k-eff between the worth case and the base case plus two times the root sum square of the KENO case uncertainties.

Table 9.6 shows the calculation of 95/95 k-eff margin to the limit. Uncertainty values are combined by root sum square. Bias values are added. NRC administrative margin of 0.01 dk is added to the total bias and uncertainty for the Dominion margin calculation. Dominion margin is 1.0 minus base case k-eff minus total bias and uncertainty. The IFBA and burnup credit cases used fresh fuel uncertainty and bias values because the scenarios are similar and because of the excess amount of margin to the limit for these cases.

For depleted fuel, margin is calculated using fresh fuel and depleted fuel code validation bias and uncertainty. The option with the least margin is shown.

The storage configuration with fresh 4.75 wt% fuel in each storage cell results in the least Dominion margin (0.0079 dk). The IFBA credit and burnup credit cases are non-limiting with approximately 0.02 dk margin. Satisfaction of the storage requirements shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 123 of 300

Table 9.6: Region 1 Total Bias, Uncertainty, and Margin (0 ppm soluble boron)

Burnup (GWd/MTU)	0	0	2
Enrichment (wt%)	4.75	5.00	5.00
IFBA Rods	0	12	0
Worths (dk)			
Minor Actinides and FP	0.0000	0.0000	0.0184
Burnup	0.0000	0.0000	0.0219
Uncertainties (dk)			
Fuel Stack PTD	0.0007	0.0007	0.0007
Enrichment, +0.05	0.0022	0.0000	0.0000
Pellet OD	0.0004	0.0004	0.0004
Active Fuel Length	0.0002	0.0002	0.0002
Clad ID	0.0002	0.0002	0.0002
Clad OD	0.0021	0.0021	0.0021
GT ID	0.0006	0.0006	0.0006
GT OD	0.0005	0.0005	0.0005
Pin Pitch	0.0009	0.0009	0.0009
Cell Wall Thickness	0.0023	0.0023	0.0023
Cell I.D.	0.0061	0.0061	0.0061
Cell Pitch	0.0054	0.0054	0.0054
Wrapper Thickness	0.0005	0.0005	0.0005
Wrapper Width	0.0004	0.0004	0.0004
BORAL width	0.0013	0.0013	0.0013
BORAL thickness	0.0010	0.0010	0.0010
Depletion Worth Unc.	0.0000	0.0000	0.0011
Burnup Measurement Unc.	0.0000	0.0000	0.0009
Code Benchmarking Unc.	0.0048	0.0048	0.0088
KENO Case Uncertainty	0.0002	0.0002	0.0002
RSS OF UNCERTAINTIES	0.0104	0.0102	0.0127
Biases (dk)			
Minor Actinides and Fission Products	N/A	N/A	0.0003
Clad Creep and Grid Growth**	N/A	N/A	0.0000
Radial Burnup Tilt**	N/A	N/A	0.0000
BORAL Blisters*	0.0009	0.0009	0.0009
Code Benchmarking Bias	0.0034	0.0034	0.0018
SUM OF BIASES	0.0043	0.0043	0.0029
Summary			
Base Case k-eff	0.9675	0.9534	0.9548
Total Bias and Uncertainty	0.0147	0.0145	0.0156
NRC Administrative Margin	0.0100	0.0100	0.0100
Maximum k-eff	0.9921	0.9779	0.9804
10CFR50.68 Limit	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0079	0.0221	0.0196

*Due to small amount of depletion, fresh fuel values used at 2 GWd/MTU

** Due to small amount of depletion, these values are negligible

9.3 Region 1 KENO SFP Wall Credit Model

The Region 1 KENO wall model is derived from the Region 1 infinite lattice model. The number of storage cells is increased to match a Region 1 rack (7 x 10) and one side of the model includes the rack-wall gap, the SFP liner (0.25 inch stainless steel), and the SFP wall. Reflective boundary conditions are used on all sides of the model.

Figure 9.4 shows a 2-D representation of the Region 1 wall model with 10 cm of water between the racks and the SFP liner. The 8x7 group of cells at the top of Figure 9.4 (Region 1A) have un-poisoned fresh 4.75 wt% fuel. The remaining two rows of cells adjacent to the SFP wall (Region 1B) have un-poisoned fresh 5.0 wt% fuel. This storage configuration allows storage of 5.0 wt% fuel with no credit for burnup or integral absorber.

Tolerance cases evaluate the significance of concrete content, concrete thickness, and rack-wall gap size. The tolerance cases demonstrate that the size of the gap between the rack and the SFP is unimportant and the concrete composition and thickness is unimportant.

KENO cases in Table 9.7 establish the sensitivity to rack-wall water gap size, concrete thickness, and concrete content (0.5, 1.0, and 1.5 volume fraction SCALE regulatory concrete, concrete replaced by water, and EPRI "dry" concrete). The results are insensitive to any of these perturbations. The Region 1 wall model highest k-eff (0.96707) is obtained with 10 cm rack-wall gap and 30 cm of concrete. All of the wall model k-eff values are lower than the Region 1 infinite lattice model. The infinite lattice analysis bounds the Region 1A/1B configuration.

Figure 9.4: KENO Region 1 Wall Credit Model

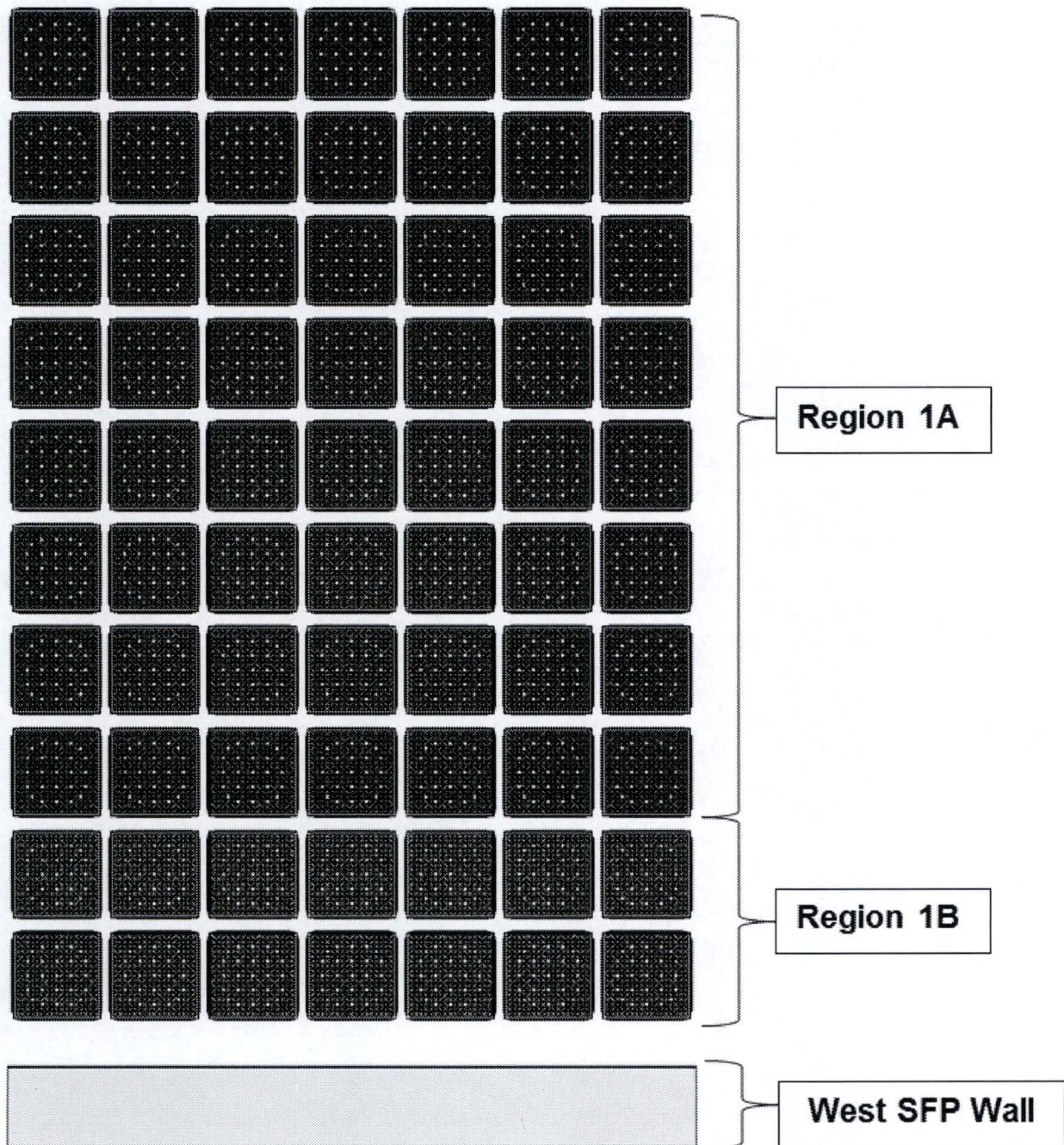


Table 9.7: Region 1 Wall Model Base and Sensitivity Cases

Enrichment (wt% U235)*	Tolerance	K-eff	Uncert.	EALF	Note
4.75 /4.75	32F Base	0.96748	0.00008	0.316	No SFP wall (infinite lattice)
4.75 /5.0	Wall gap	0.96689	0.00008	0.317	20 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Wall gap	0.96682	0.00008	0.317	16 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Wall gap	0.96701	0.00008	0.317	10 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Wall gap	0.96698	0.00008	0.317	5 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Wall gap	0.96696	0.00008	0.317	2 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Wall gap	0.96694	0.00009	0.318	0 cm gap, SS liner, 20 cm concrete
4.75 /5.0	Concrete	0.96707	0.00008	0.317	10 cm gap, SS liner, 30 cm concrete
4.75 /5.0	Concrete	0.96686	0.00008	0.317	10 cm gap, SS liner, 40 cm concrete
4.75 /5.0	Concrete	0.96670	0.00008	0.317	10 cm gap, SS liner, 20 cm 0.5 VF conc
4.75 /5.0	Concrete	0.96700	0.00008	0.317	10 cm gap, SS liner, 20 cm 1.5 VF conc
4.75 /5.0	Concrete	0.96687	0.00008	0.317	10 cm gap, SS liner, 20 cm dry conc
4.75 /5.0	Concrete	0.96703	0.00008	0.317	10 cm gap, SS liner, 20 cm water

*Region 1A / Region 1B

9.4 *Region 1 Analysis Summary*

Region 1 has been demonstrated to satisfy the SFP k-eff requirement for spent fuel pool analyses as follows:

- k-eff < 1.0 if flooded with unborated water (95% probability, 95% confidence) if credit for soluble boron is taken

A fuel assembly may be placed in Region 1 if it meets one of the following criteria:

- 1) The enrichment is ≤ 4.75 wt% U-235.
- 2) The enrichment is ≤ 5.0 wt% U-235 and the fuel assembly contains 12 or more IFBA rods.
- 3) The enrichment is ≤ 5.0 wt% U-235 and the assembly average burnup is ≥ 2 GWd/MTU.
- 4) The enrichment is ≤ 5.0 wt% U-235 and the assembly is stored in one of the two rows of rack cells adjacent to the West SFP wall.

Region 1 analysis includes calculation and application of bias and uncertainty as well as identification of NRC administrative margin and Dominion margin to the k-eff limit. Satisfaction of the storage requirements shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

9.5 *Region 2 Analysis*

9.5.1 Region 2 Fuel Storage

A fuel assembly may be placed in Region 2 if it meets one of the following criteria:

- 1) The fuel assembly burnup and enrichment satisfy the burnup credit curve (Figure 9.14)
- 2) The fuel assembly enrichment is ≤ 5.0 wt% and the fuel assembly contains a 24 finger control rod assembly.

Options 1 and 2 are confirmed by analysis using an infinite lattice rack cell model.

9.5.2 Region 2 Modeling Assumptions

These modeling simplifications and assumptions are used in the Region 2 analysis:

- a) The plenum, guide tubes above and below the active fuel, and top and bottom nozzles are modeled as water.
- b) The storage cell is modeled at its nominal length. The rack structure below the bottom of the storage cell tube is modeled as water. A water reflector extends above the storage tube at the top of the model.
- c) Neither reduction in fuel enrichment nor annular pellets are modeled for axial blankets. This is conservative because it increases SFP k-eff.
- d) BORAL is conservatively modeled as the same length as fuel. Table 4.7 shows that using nominal dimensions, the BORAL (using minimum BORAL length) extends well below the fuel in current and prior fuel designs and extends slightly above the fuel in current and prior fuel designs. This obviates the need for fuel axial position tolerance and BORAL length tolerance.
- e) The BORAL wrapper length is modeled equal to the active fuel stack length. BORAL wrappers actually extend beyond the upper and lower boundaries of the fuel stack. Since the region above and below the fuel is insensitive (item b above) and the un-modeled BORAL is a strong neutron absorber, there is no need for a wrapper length tolerance.
- f) Fuel pin pitch growth is assumed to be the same relative magnitude as grid growth. This is reasonable because fuel pins are constrained within the grid lattice.
- g) Clad OD is conservatively assumed to decrease linearly from 0 to 20 GWd/MTU and remain at the minimum value at higher burnup. This assumption is conservative based on the data in Section 9.6.5.
- h) Zr-based clad materials are modeled as Zr.

- i) Certain tolerances are only calculated for fresh fuel and not for depleted fuel for efficiency. This practice is justified because results for other tolerances calculated for fresh and depleted fuel that change only modestly and because the tolerances assumed to be constant with burnup are small compared to the total uncertainty and any changes would not be seen in the RSS total.
- j) Boral clad thickness is approximated. Boral clad thickness for thickness tolerance cases is conservatively assumed to be 0 (no clad) and thick enough to fill the design wrapper space (maximum).
- k) Water density is determined assuming the pressure is 15 psia.
- l) Grids are represented by homogenizing the grid and the water in the fuel lattice over the length of the fuel. Water inside the guide tubes are modeled as pure water.

9.6 *Region 2 Infinite Lattice Model*

Drawings and dimensions of Region 2 fuel racks are provided in Section 4.2.2. The KENO Region 2 model is a 2x2 model with reflective axial and periodic X-Y boundary conditions. The rack module to module spacing is ignored since the rack module baseplate protrudes past the outer cell walls such that storage cell pitch across the rack to rack interface is larger than the normal rack storage cell pitch. Figure 9.7 shows a 2-D representation of the Region 2 model. Figure 9.8 shows a 3-D cutaway representation of the Region 2 model. Fuel is stored in all cells of the infinite lattice model. A 6x6 expanded version of the Region 2 model is used to evaluate the horizontal burnup tilt bias.

Figure 9.7: X-Y Representation of Region 2 2x2 KENO Model

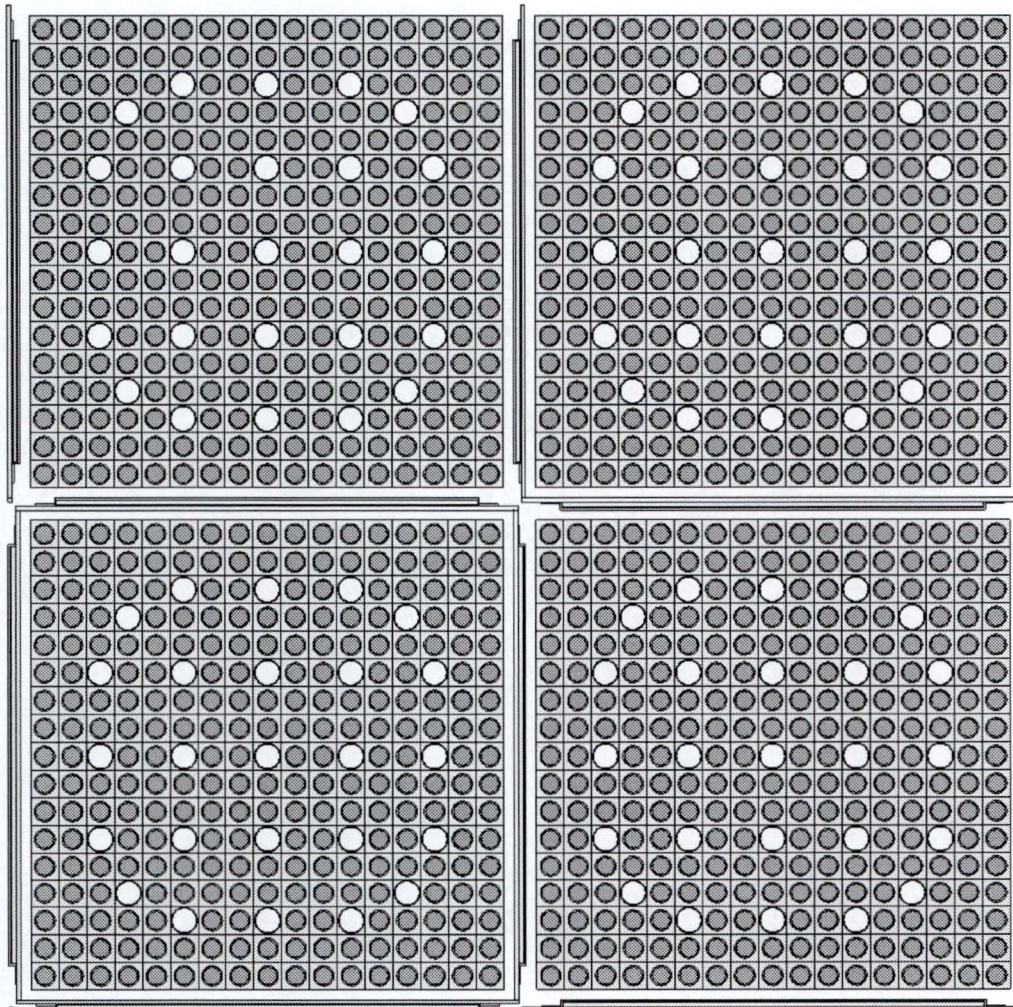
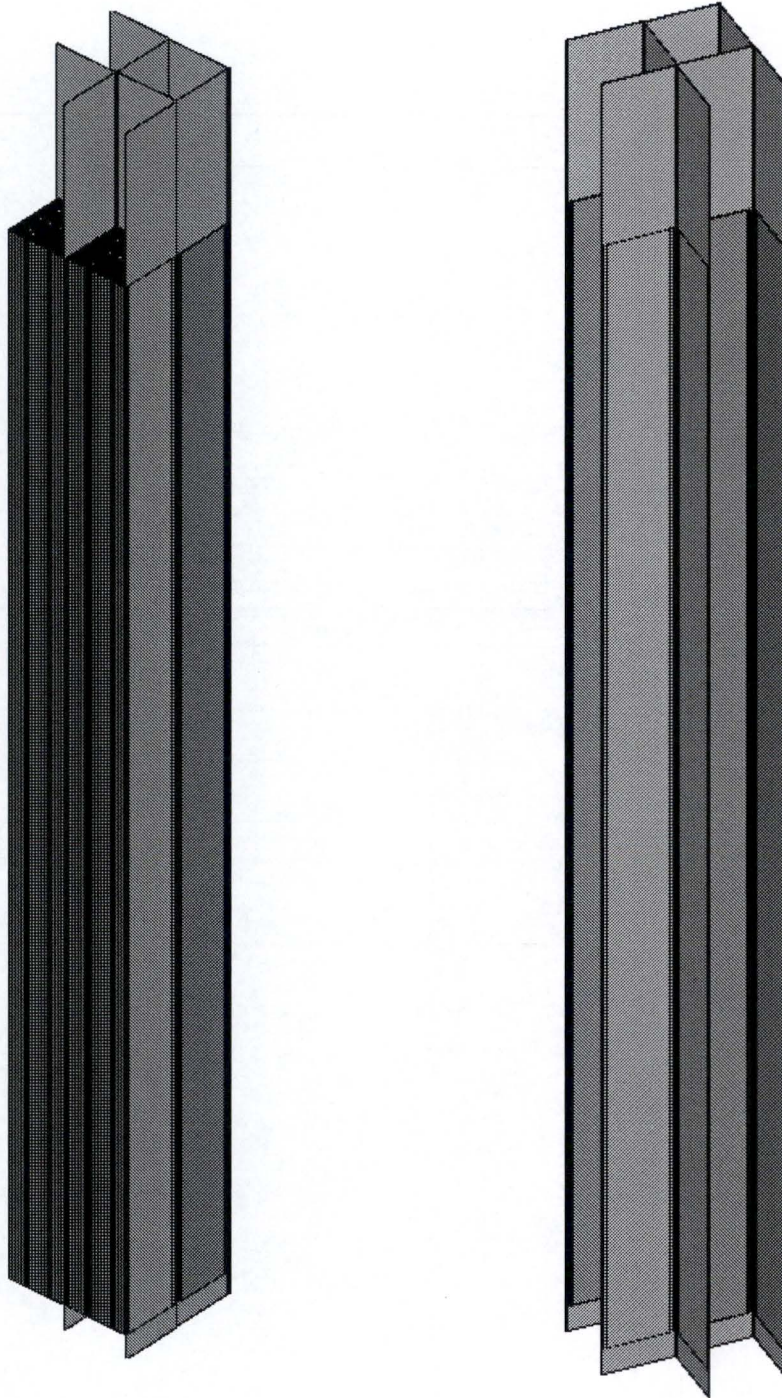


Figure 9.8: 3D Representation of Region 2 2x2 KENO Model



9.6.1 BORAL Blisters

BORAL blisters are modeled in the same way as the Region 1 analysis described in Section 9.2.1.

9.6.2 SFP Normal Operation Water Temperature and Density

Rack k-eff calculations are performed at the same four water temperatures as Region 1 (Table 9.1).

9.6.3 TRITON Depletion Input for Burnup Credit

Depleted fuel content is calculated as described in Section 8. Uniform and NUREG/CR-6801 burnup profiles are used to determine the most reactive profile.

9.6.4 Asymmetric Fuel Placement

Asymmetric placement of fuel in the storage cell is evaluated by co-locating fuel in the 2x2 model toward the center of the model and by co-locating fuel in the same direction into the corner of each storage cell. As in the Region 1 calculation, centered placement in each cell maximized k-eff.

9.6.5 Fuel Geometry Changes with Burnup

Zircaloy based grids can experience expansion during fuel assembly depletion, which can increase fuel pin pitch. Pin pitch tolerance cases confirm that with no soluble boron, increased pin pitch increases fuel rack k-eff. MPS3 fuel has used Zircaloy-4 and ZIRLO grids. For Region 2 burnup credit analysis, grid growth is included in a bias term along with clad creep.

Figure 9.9 shows grid growth data for Zircaloy-4 grids (red data points). The Zirc-4 grid growth model used in this analysis was added to the figure and its equation is displayed below. This model was developed using non-public data but fits reasonably well with the public data in Reference 38.

Similarly, Figure 9.10 shows the degree of expansion for ZIRLO grids. A grid growth model fit line is added over the burnup range of interest (maximum burnup credit for Region 2 is ~40 GWd/MTU). The ZIRLO grid growth model used in this analysis was developed using non-public data. The growth model bounds or matches most of the public data in Reference 42.

Grid expansion used in the KENO Region 2 model to determine fuel pin pitch expansion at all axial elevations (the fit line on Figures 9.9 and 9.10) is described by the following equations (X = burnup in GWd/MTU, burnup > 20 GWd/MTU):

$$\text{Growth (Zircaloy-4, \%)} = 0.0186X - 0.3726$$

$$\text{Growth (ZIRLO, \%)} = 0.0147X - 0.2941$$

The standard Region 2 fuel model is based on ZIRLO grids, which exhibit less growth than Zircaloy-4. Fuel batches 1 through 5 used Inconel grids. One batch of fuel (Batch 6, introduced in Cycle 4 and discharged after Cycle 5) was built with Zircaloy-4 grids. Batch 7 (introduced in Cycle 5 and discharged after Cycle 9) was re-caged into assembly skeletons with Zircaloy-4 grids. For these 4.2 – 4.5 w/o fuel assemblies, the Region 2 burnup requirement is ~36 GWd/MTU or less. At 36 GWd/MTU, the additional grid growth of Zircaloy-4 vs. ZIRLO is []^{a,c}. Table 9.12 shows that the pin pitch tolerance is []^{a,c} of the nominal dimension, and Table 9.20 shows that tolerance is conservatively worth 0.0012 dk at 35.65 GWD/MTU. Therefore, based on fuel pitch tolerance cases, the effect of additional Zircaloy-4 grid growth at 36 GWd/MTU is about 0.0007 dk. Burnup in excess of the Region 2 requirement (~0.0035 dk) and decay time of at least 12 years (~0.02 dk) is available to more than offset the grid effect for these Zircaloy-4 grid assemblies. Fuel batches 1 through 5 used Inconel grids.

SFP k increases with reduced clad outer diameter (OD) with no soluble boron. Clad behavior with burnup is a complex function of many variables. A simple and conservative approach is used for MPS3 SFP criticality calculations.

The zirconium alloys Zircaloy-4 and ZIRLO have been used for cladding at MPS3. Reference 25 shows maximum fuel clad diameter reduction for Zircaloy-4 of about 70 microns (approximately 0.7% of the MPS3 fuel clad OD).

The NRC approved Westinghouse PAD 4.0 model [37] predicts ZIRLO clad OD reduction of []^{a,c} occurring from 0 to about 20 GWd/MTU, followed by clad OD increase to []^{a,c} greater than the initial value. Growth of the outer oxide layer partially offsets the reduced clad OD. The “typical” oxide thickness at 20 GWd/MTU is about []^{a,c} [37]. The net fuel clad maximum OD reduction is therefore []^{a,c}.

Criticality Safety Evaluation Report

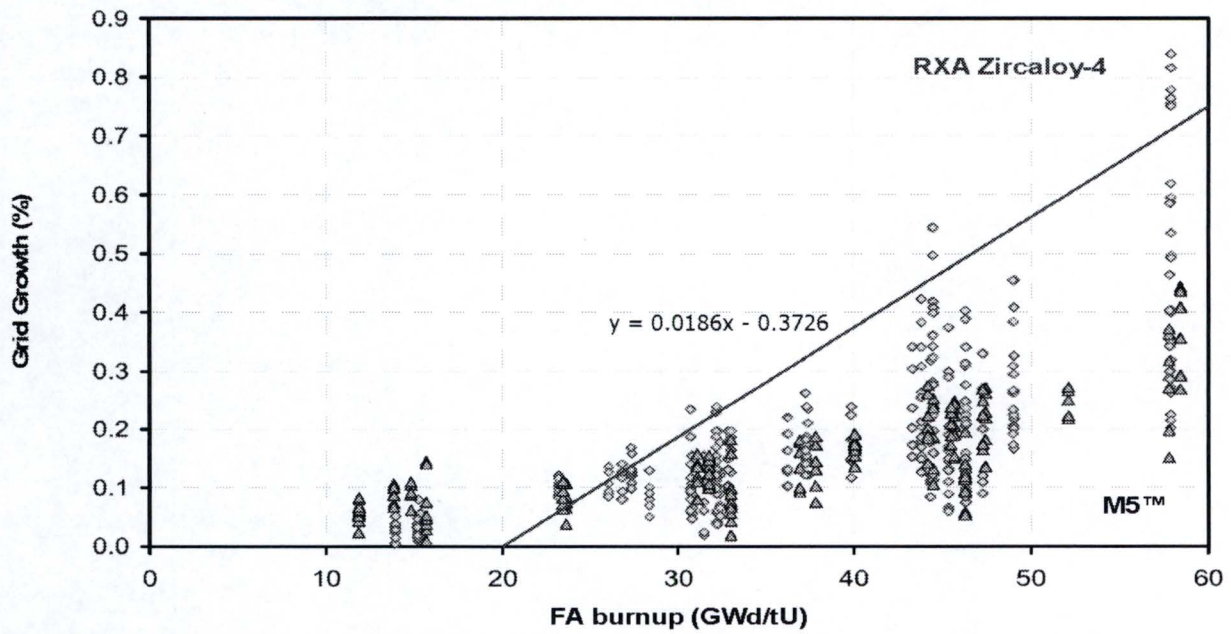
Serial No. 18-039

Docket No. 50-423

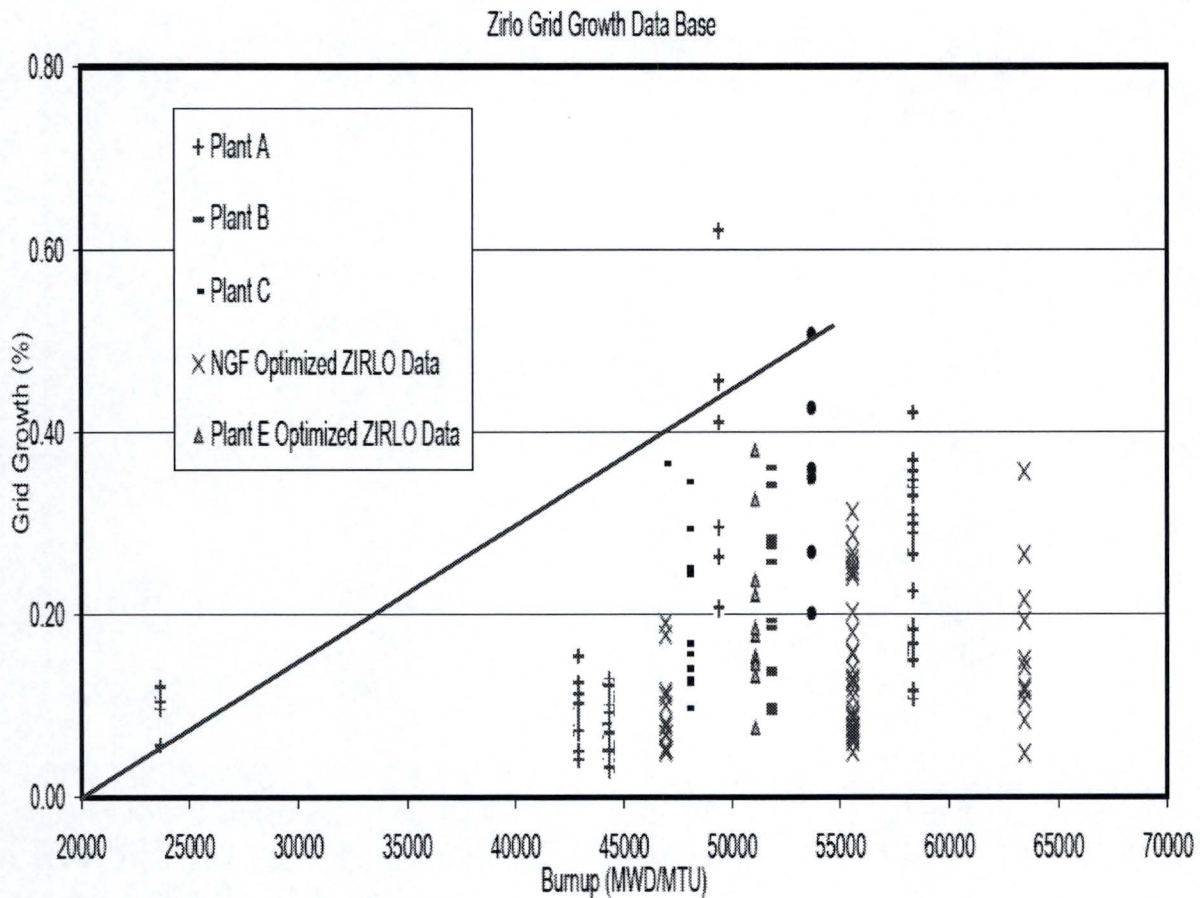
Attachment 6, Page 134 of 300

Fuel clad creep down is evaluated as a bias. Although models and measurements confirm that clad OD rebounds after reaching a maximum reduction, a conservative two segment linear function will be used. The clad OD is assumed to decrease linearly by []^{a,c} at 20 GWd/MTU and then remain at this minimum OD through the rest of the burnup range.

Figure 9.9: Zircaloy-4 Grid Growth [38]



NOTE: Zirc-4 growth model line added.

Figure 9.10: ZIRLO Grid Growth Data [42]

NOTE: ZIRLO growth model line added.

Pellet densification and/or swelling can be ignored because they do not change the fuel to water ratio. Verification that the effect of slightly changing the pellet diameter without changing the fuel mass is neutronically insignificant is provided by comparing the pellet OD and pellet density tolerance case results. One case changes fuel mass and geometry and the other changes only the fuel mass. Adjusted for the amount of fuel mass change represented in each case, the effect on k_{eff} is the same to within one half of the uncertainty of one KENO case.

9.6.6 Horizontal Burnup Tilt

The effect of horizontal burnup tilt is determined at 10, 20, 30, 36, and 40 GWd/MTU using a Region 2 6x6 rack cell model. Analogous to the calculation of asymmetric placement bias in Region 1, a 4x4 region of optimally oriented, maximally "tilted" fuel is surrounded by a buffer row of fuel with no tilt. A 4x4 region is large enough to dominate the SFP k-eff calculation. Periodic X-Y boundary conditions replicate this grouping of fuel across the entire fuel pool.

Each "tilted" fuel assembly is modeled as two burnup zones on diagonally opposite halves of the fuel assembly. All fuel pins in each diagonal half have the same burnup, and the average burnup of the two halves is equal to the assembly average burnup of a reference case un-tilted assembly. A uniform axial burnup is used for simplicity and because the intent of this calculation is to determine the effect of a horizontal burnup tilt.

Two different 4x4 region burnup tilt orientations are modeled (Figures 9.11 and 9.12). The lower burnup portion of each assembly is oriented towards the center of the 4x4 block (Figure 9.11) or the center of each 2x2 cluster (Figure 9.12). A tilt bias is calculated using the KENO rack model k-effs from tilted and uniform burnup cases.

The magnitude of assembly horizontal burnup tilt was determined using SIMULATE-5 model predictions for fuel at the end of each cycle. Figure 9.13 shows the fuel assembly tilt results for 16 cycles of fuel and the bounding value used for the TRITON depletion models.

Figure 9.11: Horizontal Burnup Tilt KENO Model #1

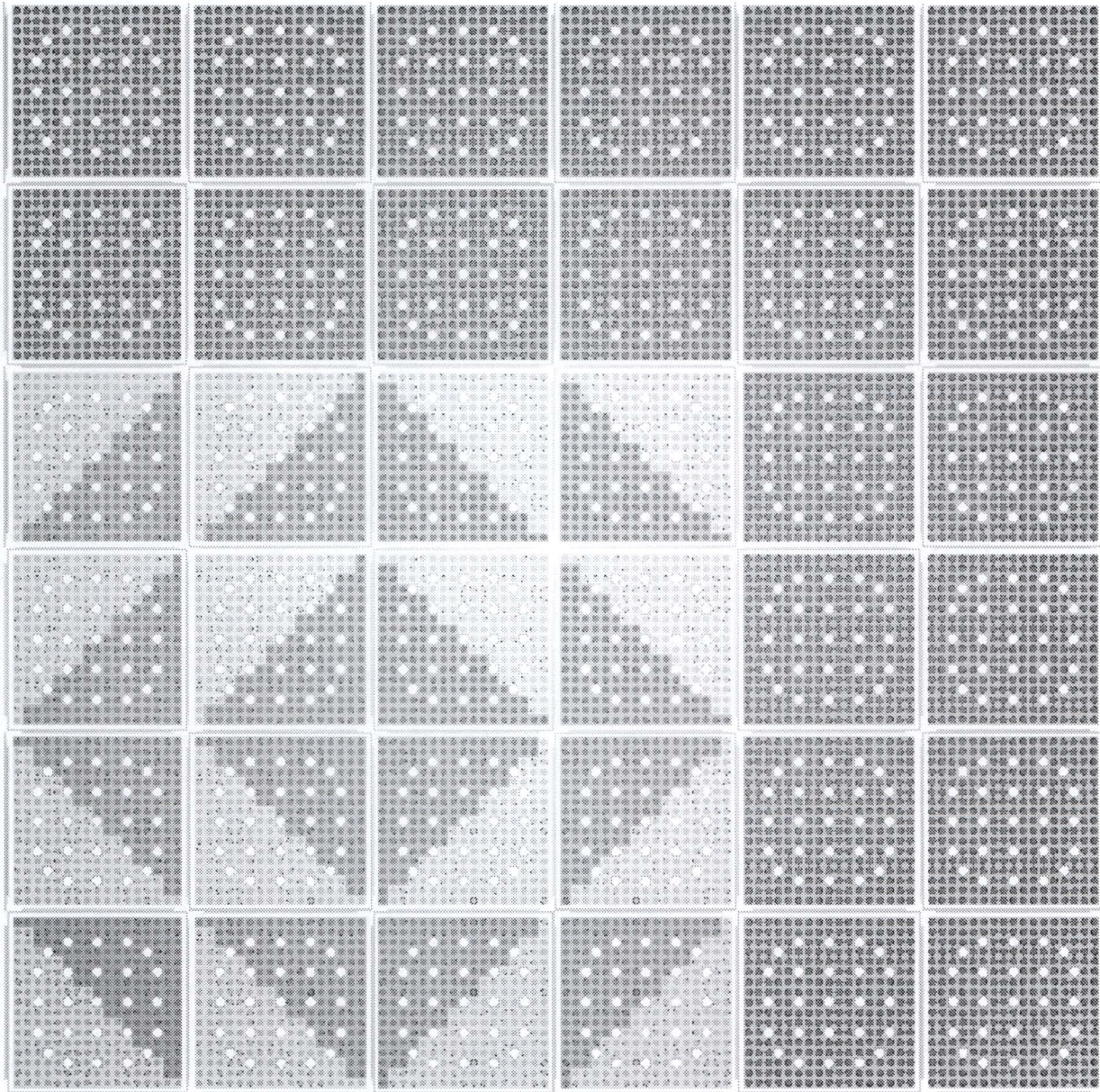


Figure 9.12: Horizontal Burnup Tilt KENO Model #2

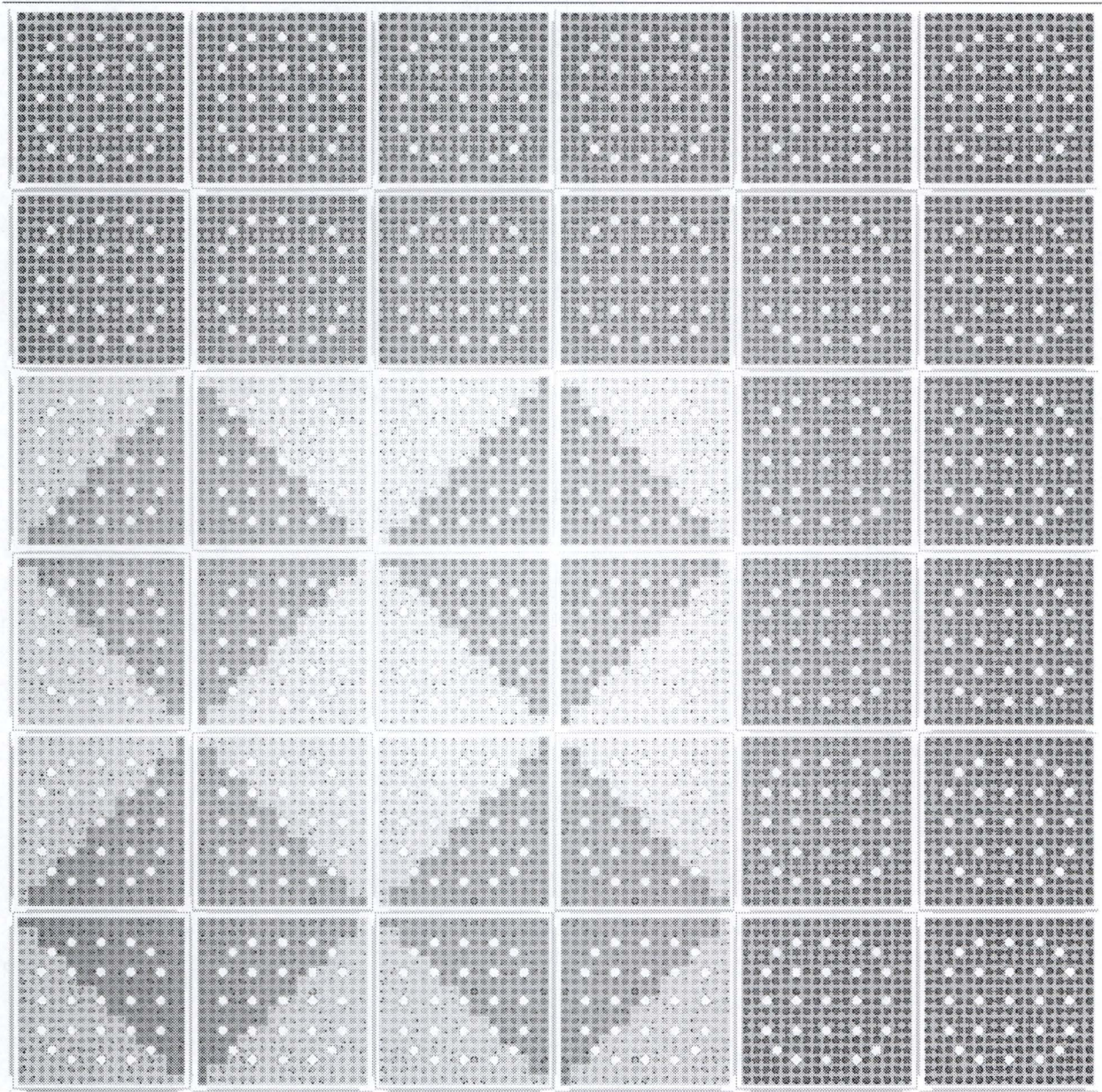


Figure 9.13: Quadrant Tilt vs. Burnup

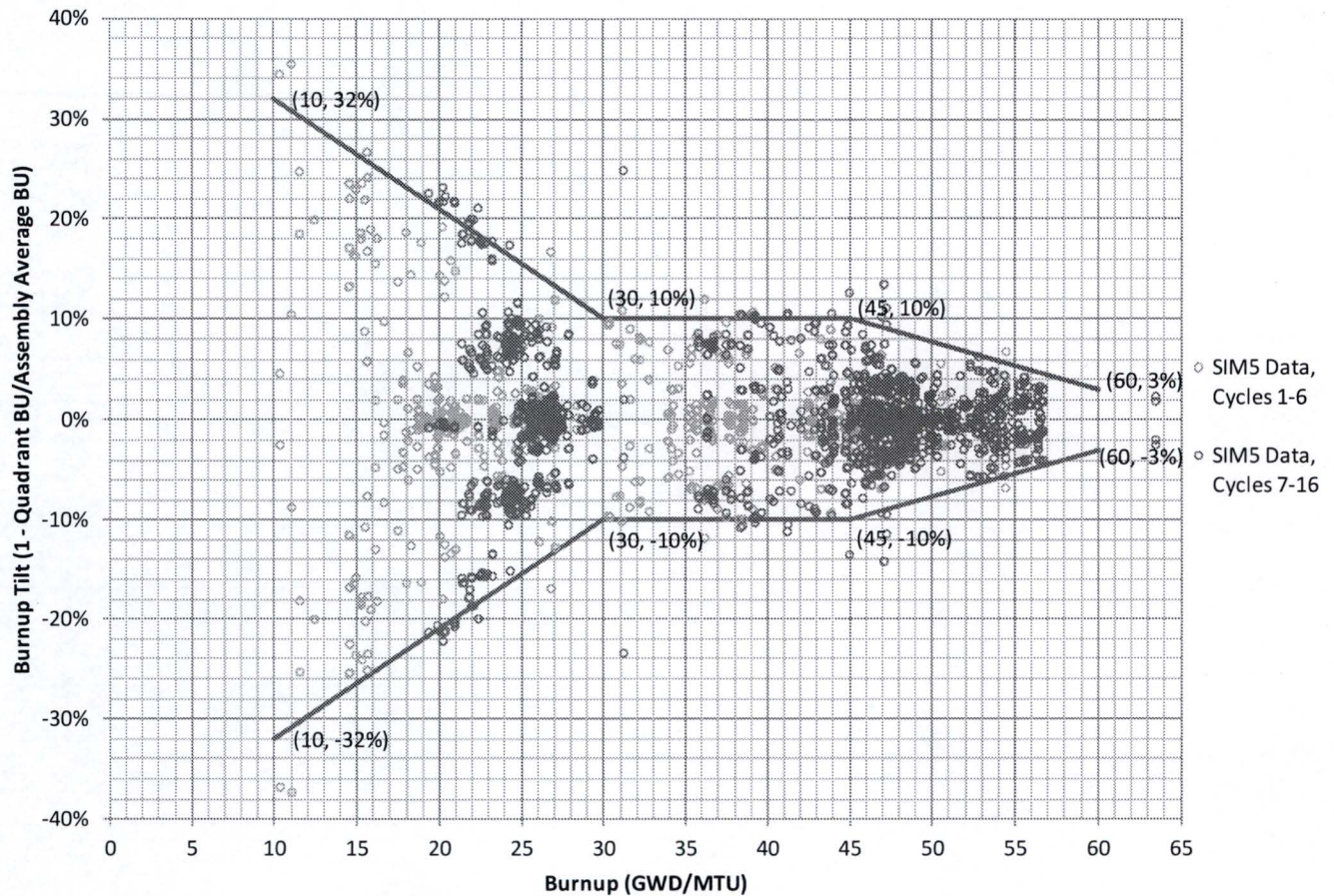


Table 9.10 shows the results of the KENO burnup tilt cases. These tilt bias values are considered conservative, because only a small portion of the fuel population has burnup tilts near the analyzed values and because the Region 2 model represents multiple repeating clusters of optimally oriented fuel assemblies.

Table 9.10: Horizontal Burnup Tilt Bias

Enrich. (wt%)	Assembly Burnup (GWd/MTU)	Tilt	Low Burnup (GWd/MTU)	High Burnup (GWd/MTU)	Case	k-eff	Sigma	EALF	Tilt Bias (dk)*
2.60	10	0	10	10	Base	0.96816	0.00006	0.28	N/A
3.25	20	0	20	20	Base	0.95659	0.00006	0.333	N/A
4.00	30	0	30	30	Base	0.94899	0.00006	0.379	N/A
4.50	36	0	36	36	Base	0.94532	0.00006	0.404	N/A
5.00	40	0	40	40	Base	0.95065	0.00006	0.426	N/A
2.60	10	32%	6.8	13.2	Inward	0.96854	0.00006	0.279	0.0005
3.25	20	21%	15.8	24.2	Inward	0.95695	0.00006	0.332	0.0005
4.00	30	10%	27.0	33.0	Inward	0.94918	0.00006	0.379	0.0004
4.50	36	10%	32.4	39.6	Inward	0.94542	0.00006	0.405	0.0003
5.00	40	10%	36.0	44.0	Inward	0.95096	0.00006	0.426	0.0005
2.60	10	32%	6.8	13.2	4 Clusters	0.96861	0.00006	0.279	0.0006
3.25	20	21%	15.8	24.2	4 Clusters	0.95706	0.00006	0.332	0.0006
4.00	30	10%	27.0	33.0	4 Clusters	0.94924	0.00006	0.378	0.0004
4.50	36	10%	32.4	39.6	4 Clusters	0.94543	0.00006	0.405	0.0003
5.00	40	10%	36.0	44.0	4 Clusters	0.95090	0.00006	0.426	0.0004

*Includes statistical uncertainty

9.6.7 Code Validation Bias and Uncertainty

Validation of SCALE/KENO using ENDF/B-VII 238 group cross sections identified EALF as the primary bias and uncertainty trend. Region 2 KENO cases have $EALF \leq 0.35$ eV with fuel burnup < 30 GWd/MTU $EALF \leq 0.4$ eV with fuel burnup < 40 GWd/MTU and $EALF < 0.45$ with burnup ≥ 40 GWd/MTU. The applicable bias and uncertainty for fresh fuel and depleted fuel (MOX validation cases) are shown in Table 9.11.

As in the Region 1 analysis, the limiting temperature is 32 °F, therefore, the validation temperature bias is not applicable.

To cover gaps in the validation for minor actinides and fission products, a bias of 1.5% of the minor actinide and fission product worth is added. [13]

Table 9.11: Region 2 Validation Bias and Uncertainty

Validation cases	Bias (dk)	Uncertainty (dk)
UO2 (EALF < 0.35)	0.0034	0.0048
MOX (EALF < 0.35)	0.0017	0.0083
UO2 (EALF < 0.4)	0.0034	0.0048
MOX (EALF < 0.4)	0.0018	0.0088
UO2 (EALF < 0.45)	0.0037	0.0048
MOX (EALF < 0.45)	0.0019	0.0094

9.6.8 Summary of Bias and Uncertainty

Table 9.12 summarizes the biases and uncertainties incorporated in the Region 2 analysis.

Table 9.12: Summary of Region 2 Biases, Uncertainties, and Conservatism

Analysis	Item / Type (B=bias, C=conservatism, U=uncertainty)	Explanation
Depletion	Burnable absorbers	C Bounding BPRA (24 WABA bounds IFBA).
Depletion	Soluble boron	C 1050 ppm bounds cycle average boron.
Depletion	Fuel and Water Temperature	C Bounding high node-specific. Section 8.2, 8.4, 8.5.
Depletion	Specific power	C Bounding high power assembly history.
Depletion	Axial burnup shapes	C Uniform and NUREG/CR-6801.
Depletion	Grids	C Maximum Zr grids
Depletion	Volatile fission products	C Reduced based on gap release fractions
Depletion	Fuel density (base case)	C 95.5% net bounds all fuel batches
Depletion	Power history	C Reduced 50% for the last 40 days of depletion
Depletion	Incore thimble	C Included in base depletion
Rack k-eff	Fuel density	U 95.5% [] ^{a,c} net bounds all fuel batches
Rack k-eff	Grids (base case)	C Minimum volume Zr grids
Rack k-eff	Boral B-10	C Minimum ([] ^H)
Rack k-eff	Boral width	U [] ^H
Rack k-eff	Boral thickness	U [] ^H
Rack k-eff	Boral wrapper thickness	U [] ^H
Rack k-eff	Boral wrapper width	U [] ^H (welded end tab removed)
Rack k-eff	Fuel pin pitch	U 0.496 [] ^{a,c}
Rack k-eff	Fuel pellet OD	U 0.3225 [] ^{a,c}
Rack k-eff	Fuel clad ID	U 0.329 [] ^{a,c}
Rack k-eff	Fuel clad OD	U 0.374 [] ^{a,c}
Rack k-eff	Guide tube ID	U 0.442 [] ^{a,c}
Rack k-eff	Guide tube OD	U 0.474 [] ^{a,c}
Rack k-eff	Fuel stack height	U [] ^{a,c}
Rack k-eff	Burnup worth	U 5% of burnup worth
Rack k-eff	Measured burnup	U 4% of burnup worth
Rack k-eff	Enrichment	U [] ^{a,c}
Rack k-eff	Cell wall thickness	U [] ^H
Rack k-eff	Rack cell pitch	U [] ^H
Rack k-eff	Rack cell ID	U [] ^H
Rack k-eff	Code uncertainty	U Section 9.6.7
Rack k-eff	KENO case uncertainty	U 2 standard deviations
Rack k-eff	Minor actinides + FP	B 1.5% of worth.
Rack k-eff	Code bias	B Section 9.6.7
Rack k-eff	Temperature	C Most reactive temp. (32 F) used
Rack k-eff	Creep and grid growth	B Modified pin pitch and clad diameter
Rack k-eff	Horizontal burnup tilt	B Section 9.6.6
Rack k-eff	Eccentric placement	C Most reactive position (centered in cell) used
Rack k-eff	Boral blisters	B Full panel blister 0.009 in. thick
Rack k-eff	NRC admin. margin	B 1% Δk

9.6.9 Unbounded Historical Fuel

Use of 24 WABA in the TRITON depletions leaves two batches of fuel potentially unbounded due to the burnable absorber used. In Cycle 1, some batch B and C fuel assemblies contained 24 Pyrex BPRA. Pyrex BP is somewhat more limiting than WABA because Pyrex rods displace slightly more water than WABA and have stainless steel clad instead of zirconium based clad. The effect of Pyrex BPRA depletion vs WABA depletion is estimated to be <0.0015 dk in Table 8.6 (three cycles of depletion with BP). This small effect is more than offset by these considerations:

- 1) Batch 2 and 3 assemblies have at least 5 GWd/MTU more burnup than required for Region 2 storage (~ 0.025 dk).
- 2) Batch 2 and 3 assemblies were discharged before 1992. Decay time of 25 years provides substantial additional margin for these assemblies since decay time is not credited in Region 2. Decay time of 25 years is worth >0.02 dk at 20 GWd/MTU (based on Region 3 burnup curves and fuel enrichment worth tolerances).

Ten fuel assemblies from Cycles 1 and 4 were identified in Section 8.8 as having control rod insertion history that exceeds 10 steps cycle average insertion that is bounded by the standard 24 WABA depletion. Sensitivity cases in Section 8.8 estimated the reactivity effect of depletion with a control rod inserted 10 steps and with a control rod inserted 2 nodes (bounds all MPS3 cycles) to be ~ 0.02 dk for 3 cycles of depletion. The assemblies are dispositioned for storage in Region 2 because they have over 5 GWd/MTU more burnup than required (~ 0.025 dk) and 25 years of uncredited decay time.

Additional discussion of unbounded fuel assemblies is provided in Section 9.9.8 for the Region 3 analysis.

9.6.10 Region 2 k-eff and Margin Calculation – Fresh fuel

With the exception of temperature changes, tolerance cases for Region 2 (fresh fuel with no control rod credit) are performed at 32 °F (1.00 g/cc) with no soluble boron. The 32 °F base case has the highest k-eff of the four temperature cases (0.011 dk higher than the 150 °F case). Most tolerance cases (cell wall thickness, cell ID, fuel clad OD and ID, pellet OD, BORAL wrapper thickness, guide tube OD and ID, fuel pin pitch, and BORAL thickness) were run bidirectional rather than assume monotonic behavior. Statistically significant tolerances are monotonic and only one-sided analysis is needed. Eccentric positioning cases including fuel skewed toward the model center and skewed into the same cell corner confirmed that the highest base case k-eff results from fuel centered in each rack cell. As examples, a few of the key KENO results are shown in Table 9.13.

Table 9.13: Selected Tolerance Results for Region 2 (Fresh fuel)

Enrichment (w/o U235)	Tolerance	K-eff	Uncert.	Sensitivity (dK)	EALF	Note
2.0	32F Base	0.96421	0.00006	N/A	0.189	Symmetric 2x2 32F
2.05	Enrichment	0.97159	0.00006	0.0076	0.190	Enrichment +0.05
2.0	Clad OD-	0.96544	0.00006	0.0014	0.188	Decrease clad OD
2.0	narrowboral	0.96650	0.00006	0.0025	0.188	Decrease BORAL width
2.0	Pin pitch+	0.96476	0.00006	0.0007	0.188	Increase pin pitch
2.0	150F	0.95227	0.00006	-0.0110	0.214	Temperature
2.0	Asymmetric	0.96054	0.00006	-0.00349	0.191	Eccentric 2x2 Positioning

Bias and uncertainty values resulting from KENO cases include the difference in k-eff between the worth case and the base case plus two times the root sum square of the KENO case uncertainties.

Table 9.14 shows the calculation of 95/95 k-eff margin to the limit. Uncertainty values are combined by root sum square. Bias values are added. NRC administrative margin of 0.01 dk is added to the total bias and uncertainty for the Dominion margin calculation. Dominion margin is 1.0 minus base case k-eff minus total bias and uncertainty.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 146 of 300

Table 9.14: Region 2 Total Bias, Uncertainty, and Margin (0 ppm soluble boron)

Burnup (GWd/MTU)	0
Enrichment (wt%)	2.03
Uncertainties (dk)	
Fuel Stack PTD	0.0009
Enrichment, +0.05	0.0076
Pellet OD	0.0006
Active Fuel Length	0.0003
Clad ID	0.0003
Clad OD	0.0014
GT ID	0.0005
GT OD	0.0006
Pin Pitch	0.0007
Cell Wall Thickness	0.0005
Cell I.D.	0.0003
Cell Pitch	0.0006
Wrapper Thickness	0.0005
Wrapper Width	0.0003
BORAL width	0.0025
BORAL thickness	0.0003
Code Benchmarking Unc.	0.0048
KENO Case Uncertainty	0.0001
RSS OF UNCERTAINTIES	0.0096
Biases (dk)	
BORAL Blisters	0.0004
Code Benchmarking Bias	0.0034
SUM OF BIASES	0.0038
Summary	
Base Case k-eff	0.9687
Total Bias and Uncertainty	0.0133
NRC Administrative Margin	0.0100
Maximum k-eff	0.9920
10CFR50.68 Limit	1.0000
Dominion Margin (dk)	0.0080

9.6.11 Region 2 Control Rod Credit

Control rods in the SFP may have reached their in-core design lifetime and been discharged. Discharged control rods may have experienced some depletion of the absorber material near the bottom of the rod. The amount of depletion is dependent on the at-power residence time in the reactor. Conservative depleted absorber content is required to extend control rod credit to discharged control rods.

There have been three types of control rod used in MP3. Table 9.15 summarizes the relevant design features. The first generation of MPS3 control rods were Hf/Zr. A transition to the second generation (AIC) began when Cycles 3 and 4 had 6 AIC control rods and 55 Hf/Zr control rods. Cycle 5 completed the transition with 61 AIC control rods. A transition to the third generation of control rods (AIC) began in Cycle 13.

RCCAs are depleted for three cycles or fewer in lead bank locations. In all but two cycles, lead bank cycle average insertion is 10 steps or less. Therefore, control rod depletion for three cycles involving the bottom 15.9 cm of absorber will bound all but Cycle 1 and Cycle 4 control rods.

Only one control rod was used in a lead bank location in three cycles and was also present in Cycle 1 or Cycle 4. No RCCA was in a lead bank location in both Cycles 1 and 4. The bounding RCCA history is one cycle of 22 step depletion and 2 additional cycles of 10 step depletion. A bounding depletion history is therefore depletion of the bottom 10 steps (15.9 cm) of absorber for three cycles and depletion of the next 12 steps (total length 34.9 cm) above that region for one cycle.

To bound control rod depletion conditions, the following TRITON depletion model input is used:

- Nominal absorber density and dimensions (TRITON depletion)
- Node 18 depletion conditions (top of core)
- Deplete 3 cycle control rod absorber (bottom 10 steps) using 40 GWd/MTU uniform axial burnup shape at core average RPD.
- Deplete 1 cycle rod absorber (next 12 steps) using 20 GWd/MTU uniform axial burnup shape at core average RPD.
- Fuel enrichment of 4.0 w/o is representative of historical MP3 fuel that experienced the highest average at power D-Bank insertion. Recent cycles have operated with less D-Bank insertion so depleting the control rods with parameters from Cycles 1-4 is bounding.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 148 of 300

Justification for the 3 cycle absorber depletion strategy is as follows. MPS3 cycle length is approximately 20 GWd/MTU. Fuel assembly average power in lead bank locations tends to be above average ($\sim 1.25 \times$ cycle average or less). However, in the top 16 cm of the fuel (midpoint of depletion region), local power even with no control rod inserted relative power tends to be well below average due to leakage ($\sim 0.4 \times$ axial average). Control rod insertion in the lead bank location assembly further reduces the local power substantially. Depletion for 40 GWd/MTU is approximately 33% more than the best estimate burnup ($20 \text{ GWd/MTU} \times 1.25 \times 3 \text{ cycles} \times .4 = 30 \text{ GWd/MTU}$) even before accounting for power depression by the control rod.

Justification for the 1 cycle absorber depletion strategy is similar, except that the basis is the Cycle 1 and Cycle 4 cycles in particular because those are the cycles with greater than 10 step cycle average control rod insertion. The maximum burnup accumulation in Cycle 1 and Cycle 4 lead bank locations is 21.2 GWd/MTU. The maximum cycle average relative un-rodged axial power of the two cycles considered at the 1 cycle absorber elevation is ~ 0.8 . Control rod insertion in the lead bank location assembly further reduces the local power substantially. Depletion for 20 GWd/MTU is approximately 18% more than best estimate burnup ($21.2 \text{ GWd/MTU} \times 0.8 = 16.96 \text{ GWd/MTU}$) even before accounting for power depression by the control rod.

Table 9.15: MPS3 Control Rod Data

CR Type	Item	Value
Hf-Zr	Absorber	95.3 - 95.4 w/o Hf, 4.5 w/o Zr
Ag-In-Cd	Absorber	80 w/o Ag, 15 w/o In, 5 w/o Cd (10.16 g/cc)
All	Absorber OD	0.341 in.
All	Clad ID	0.344 in.
All	Clad OD	0.381 in.
All	Clad Material	SS-304
All	Step size	0.625 in.
All	Fuel below absorber	5.25 in.
All	Lead bank	5 core locations
All	All banks	61 core locations
All	D-bank insertion @ HFP	Figure 8.5
All	Other bank insertion @ HFP	Figure 8.5
All	Modeled D-bank insertion @ HFP	22 steps (1 cycle) 10 steps (3 cycles)

Table 9.16 presents the results of KENO control rod credit cases using fresh 5.0 w/o fuel. Control rods extend from 5.25 (maximum) inches above the bottom of the active fuel to the top of the active fuel. Control rod structure below the bottom of the control rod absorber is modeled as water. Nominal fuel and control rod dimensions are used. The bottom 15.88 cm of control rod absorber is depleted material from the 40 GWd/MTU TRITON node 18 depletion conditions. Control rod absorber from 15.88 cm to 34.93 cm is depleted material from the 20 GWd/MTU TRITON node 18 depletion conditions. The balance of the control rod absorber is fresh material.

Table 9.16: Region 2 KENO Control Rod Credit Cases (5.0 w/o fresh fuel)

Control Rod	K-eff	Uncert	EALF	Note
None	0.96430	0.00006	0.189	Base case 2.0 wt% fresh
AIC	0.93264	0.00008	0.479	5.0 wt% fresh, 32 F
AIC	0.92268	0.00007	0.531	5.0 wt% fresh, 150 F
Hf/Zr	0.93240	0.00007	0.484	5.0 wt% fresh, 32 F

The highest k-eff control rod case is more than 0.03 dk less reactive than the Region 2 fresh fuel base case. A full accounting of bias and uncertainty for this configuration would include tolerances that would increase total bias and uncertainty (absorber OD, control rod clad OD, absorber composition) but no enrichment tolerance. Review of similar SFP control rod credit cases for North Anna [23] indicates that total bias and uncertainty is smaller for the control rod configuration than for no control rod credit.

The North Anna calculations and the large amount of margin indicated by the control rod credit base cases is sufficient to conclude that for Region 2, the control rod credit configuration is significantly non-limiting and there is no need to perform a full set of tolerance and bias cases. Fuel assemblies of enrichment 5.0 wt% U-235 or less, each containing a control rod may be stored in Region 2 with no burnup required. Removal of the control rod must be performed with the assembly in a Region in which it qualifies for storage without the control rod.

9.6.12 Region 2 k-eff and Margin Calculation – Burnup credit

With the exception of temperature changes, tolerance cases for Region 2 (depleted fuel) are performed at 32 °F (1.00 g/cc) with no soluble boron. The 32 °F base case has the highest k-eff. Table 9.17 identifies the NUREG/CR-6801 Table 5 axial burnup shapes used to determine the TRITON depletion input.

Table 9.17: SFP Region 2 Limiting Axial Shapes

Assembly Average Burnup (GWd/MTU)	Limiting Shape (Uniform or NUREG/CR-6801 Table 5 Group)
10	Uniform*
20	8*
30	5
35, 36	4
40, 41	3

*Uniform and NUREG shapes checked. Limiting shape listed.

Not all uncertainty terms are calculated for each burnup/enrichment combination:

- Fuel density uncertainty is calculated for fresh fuel and used for all burnups.
- Clad OD uncertainty is used for fresh fuel and replaced by a bias for clad creep for depleted fuel.
- Some minor uncertainty items are calculated for fresh fuel and used for all burnups (shaded values in Table 9.20)
- Some bias and uncertainty items are calculated at two burnups to confirm no significant trend exists and the maximum value used at other burnups.

Minor uncertainties are not calculated at all burnups because they require significant resources and because the magnitude is small enough that modest variation would not significantly affect the RSS total uncertainty, which is dominated by enrichment uncertainty at low burnup and by burnup uncertainties at high burnup. Bias and uncertainty values resulting from KENO cases include the difference in k-eff between the worth case and the base case plus two times the root sum square of the KENO case uncertainties.

Due to the iterative nature of the burnup credit curve determination, some bias and uncertainty calculations are performed at an enrichment or burnup close to but not identical to the final

enrichment / burnup point. This is acceptable because only the enrichment uncertainty term is a strong function of enrichment. The base case k -eff at 35.65 and 40.25 GWd/MTU is determined by linear interpolation versus burnup using cases 1 GWd/MTU apart. Enrichment sensitivity is calculated with depleted fuel (2 sets of TRITON depletions required) except at 20 GWd/MTU, which uses fresh fuel. For example, two sets of TRITON depletion cases were run at 10 GWD/MTU with either 2.55 wt% or 2.60 wt% while only two fresh fuel cases were run at 3.25 wt% or 3.30 wt% to calculate the 20 GWD/MTU enrichment sensitivity. Table 9.18 provides a comparison of enrichment tolerance calculated using fresh and depleted fuel at two different enrichment / burnup combinations that supports using either method for depleted fuel.

Table 9.18: Enrichment Tolerance Calculation Comparison

Enrichment (wt% U-235)	Assembly Average Burnup (GWd/MTU)	Enrichment Tolerance (dk w/0.05 wt% increase)
2.55	0	0.0057
2.55	10U*	0.0046
4.0	0	0.0031
4.0	30	0.0032

*U indicates uniform burnup profile.

Table 9.19 contains the Region 2 KENO model k-eff base cases performed at 273 K (1.00 g/cc water density) with no soluble boron.

Table 9.19: Region 2 Base Cases (0 ppm boron)

Enrichment (w/o U-235)	Burnup (GWd/MTU)	K-eff	Uncert.	EALF
2.6	10U*	0.96570	0.00006	0.277
3.25	20	0.96323	0.00007	0.314
4.05	30	0.96135	0.00006	0.361
4.47	35	0.96332	0.00007	0.379
4.47	36	0.95940	0.00006	0.379
5.0	40	0.95791	0.00006	0.408
5.0	41	0.95366	0.00007	0.409

*U indicates uniform burnup profile.

Table 9.20 shows the calculation of 95/95 k-eff margin to the limit for depleted fuel in Region 2. Shaded cells are values that were not calculated for that burnup and enrichment combination. The reactivity effect of 4% burnup uncertainty (the k-eff increase resulting from 4% burnup reduction) is determined by linear interpolation of adjacent burnup worth vs. burnup data points that bound the reduced burnup rather than calculating 4% of the total burnup worth.

Uncertainty values are combined by root sum square. Bias values are added. NRC administrative margin of 0.01 dk is added to the total bias and uncertainty for the margin calculation. Margin is 1.0 minus base case k-eff minus total bias and uncertainty. For depleted fuel, margin is calculated using fresh fuel and depleted fuel code validation bias and uncertainty. The option with the least margin is shown. The Region 2 burnup credit curve, which matches or bounds all analyzed points, is shown in Figure 9.14.

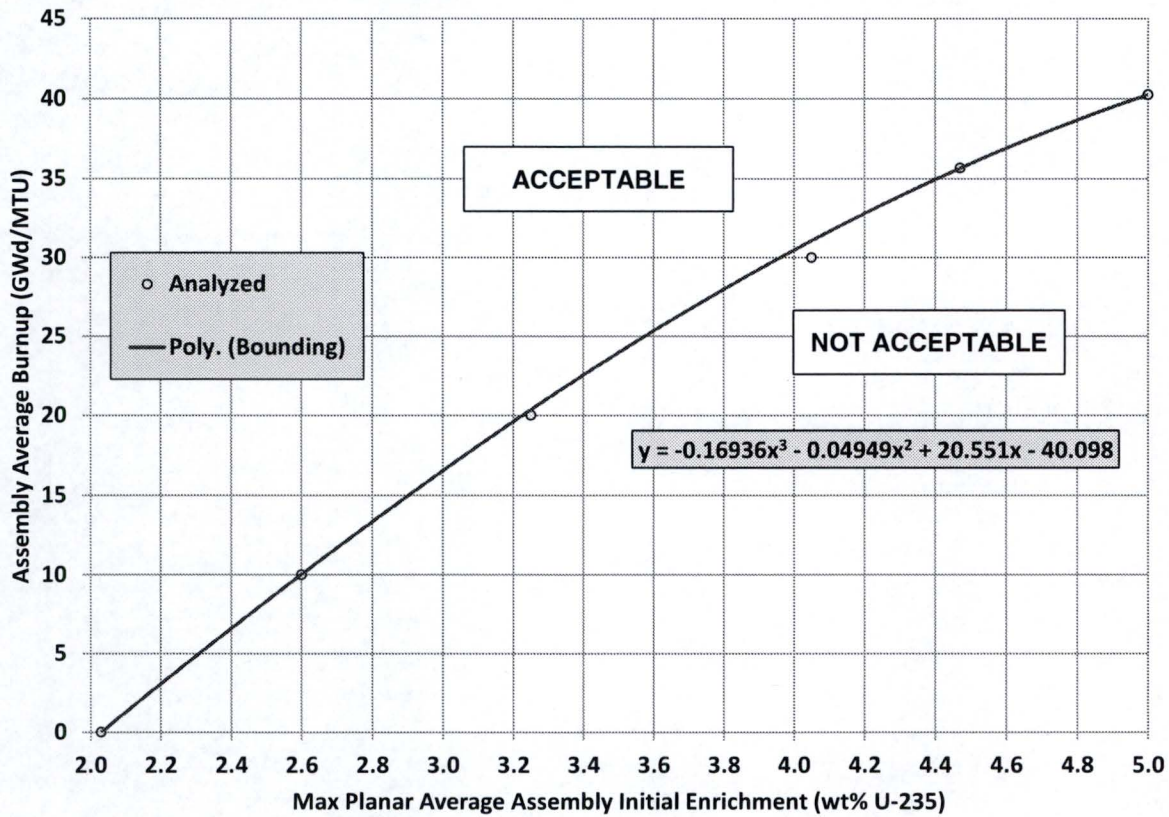
Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 153 of 300

Table 9.20: Region 2 Total Bias, Uncertainty, and Margin (Depleted fuel, 0 ppm boron)

Burnup (GWd/MTU)	10	20	30	35.65	40.25
Enrichment (wt%)	2.60	3.25	4.05	4.47*	5.00*
Worths (dk)					
Minor Actinides and FP	0.0486	0.0663	0.0871	0.0942	0.1079
Burnup	0.0737	0.1367	0.1927	0.2159	0.2441
Uncertainties (dk)					
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009
Enrichment, +0.05	0.0046	0.0043	0.0032	0.0027	0.0000
Pellet OD	0.0002	0.0002	0.0002	0.0002	0.0002
Active Fuel Length	0.0003	0.0003	0.0003	0.0003	0.0003
Clad ID	0.0003	0.0003	0.0003	0.0003	0.0003
Clad OD (replaced by creep bias)	0.0000	0.0000	0.0000	0.0000	0.0000
GT ID	0.0004	0.0004	0.0004	0.0004	0.0004
GT OD	0.0007	0.0007	0.0007	0.0007	0.0007
Pin Pitch	0.0009	0.0009	0.0012	0.0012	0.0012
Cell Wall Thickness	0.0004	0.0004	0.0004	0.0004	0.0004
Cell I.D.	0.0002	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0013	0.0013	0.0015	0.0015	0.0015
Wrapper Thickness	0.0002	0.0002	0.0004	0.0004	0.0004
Wrapper Width	0.0003	0.0003	0.0003	0.0003	0.0003
BORAL Width	0.0024	0.0024	0.0024	0.0024	0.0024
BORAL Thickness	0.0003	0.0003	0.0003	0.0003	0.0003
Depletion Worth Unc.	0.0037	0.0068	0.0096	0.0108	0.0122
Burnup Measurement Unc.	0.0029	0.0050	0.0067	0.0058	0.0099
Code Benchmarking Unc.	0.0083	0.0083	0.0088	0.0088	0.0094
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0001
RSS OF UNCERTAINTIES	0.0111	0.0130	0.0154	0.0157	0.0186
Biases (dk)					
Minor Actinides and Fission Products	0.0007	0.0010	0.0013	0.0014	0.0016
Clad Creep and Grid Growth	0.0011	0.0023	0.0039	0.0044	0.0050
Radial Burnup Tilt	0.0006	0.0006	0.0004	0.0003	0.0005
Blisters	0.0002	0.0002	0.0004	0.0004	0.0004
Code Benchmarking Bias	0.0017	0.0017	0.0018	0.0018	0.0019
SUM OF BIASES	0.0043	0.0058	0.0078	0.0082	0.0093
Summary					
Base Case k-eff	0.9657	0.9632	0.9614	0.9608	0.9568
Total Bias and Uncertainty	0.0154	0.0189	0.0232	0.0239	0.0280
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9911	0.9921	0.9946	0.9947	0.9948
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0089	0.0079	0.0054	0.0053	0.0052

Figure 9.14: Region 2 Bounding Burnup Credit Curve



Satisfaction of the burnup curve requirement shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

9.7 *Region 2 Analysis Summary*

Region 2 has been demonstrated to satisfy the SFP k-eff requirement for spent fuel pool analyses as follows:

- k-eff < 1.0 if flooded with unborated water (95% probability, 95% confidence)

In Region 2 with 4-out-of-4 storage, the requirement is met with either a) a restrictive loading curve (burnup credit curve) of minimum required assembly measured burnup versus initial fuel assembly as-built enrichment or b) credit for a control rod stored in the fuel assembly.

Region 2 analysis includes calculation and application of bias and uncertainty as well as identification of NRC administrative margin and Dominion margin to the k-eff limit. For Region 2, one 3rd order polynomial burnup credit curve is determined for fuel with initial enrichment >2.03 wt% U-235. The coefficients of the curve are shown in Figure 9.14.

9.8 *Region 3 Analysis*

9.8.1 Region 3 Fuel Storage

Analysis of Region 3 includes credit for burnup and decay time. A fuel assembly may be placed in Region 3 if the fuel assembly burnup and enrichment satisfy a burnup and decay time credit curve (Figure 9.20). Analysis is performed using an infinite lattice 6x6 rack cell KENO model.

9.8.2 Region 3 Modeling Assumptions

These modeling simplifications and assumptions are used in the Region 3 analysis:

- a) Boraflex is not credited and is modeled as water.
- b) The plenum, guide tubes above and below the active fuel, and top and bottom nozzles are modeled as water.
- c) The storage cell tubes and shear wall are modeled at the nominal shear wall length (from the bottom of the fuel assembly to ~5 cm above the active fuel. The rack structure below the bottom of the storage cell tube and above the shear wall is modeled as unborated water.
- d) No reduction in fuel enrichment is applied for axial blankets.
- e) Fuel pin pitch growth is assumed to be the same relative magnitude as grid growth. This is reasonable because fuel pins are constrained within the grid lattice.
- f) Clad OD is conservatively assumed to decrease linearly from 0 to 20 GWd/MTU and remain at the minimum value at higher burnup.
- g) Zr-based clad materials are modeled as Zr.
- h) Certain tolerances are only calculated for fresh fuel and not for depleted fuel for efficiency. This practice is justified because results for other tolerances calculated for fresh and depleted fuel that change only modestly and because the tolerances assumed to be constant with burnup are small compared to the total uncertainty and any changes would not be seen in the RSS total.
- i) Water density is determined assuming the pressure is 15 psia.
- j) Grids are represented by homogenizing the grid and the water in the fuel lattice over the length of the fuel. Water inside the guide tubes are modeled as pure water.
- k) Certain tolerances are only calculated for the 0 year decay curve and not for the decay time curves. This was done for efficiency and is justified via the same reasons listed in assumption h above.

9.9 *Region 3 Infinite Lattice Model*

Drawings and dimensions of Region 3 fuel racks are provided in Section 4.2.3. The KENO Region 3 model is a 6x6 model with void axial and periodic X-Y boundary conditions. The axial regions above and below the active fuel are large enough to be effectively neutronically infinite (~ 28 cm of water at the bottom and ~45 cm of water at the top). Each Region 3 rack is a 6x6 module with a stainless steel rack wall around the perimeter of the module. The KENO model represents an actual rack module.

The X-Y boundary conditions represent an infinite number of adjacent modules. The KENO model includes the as-installed minimum rack-to-rack spacing of 1 inch. The nominal rack spacing is 1.125 inch. The rack-to-rack adjacent cell pitch is greater than the nominal storage cell pitch as long as the rack-to-rack spacing is greater than 0.25 inch.

Figure 9.15 shows a 2-D X-Y representation of the Region 3 model. Fuel is stored in all cells of the infinite lattice model. Storage cells on the perimeter of Region 3 (a 3x3 or 3x4 cluster of rack modules as shown in Figure 4.8) do not have Boraflex or wrappers on the outer face. Because no credit is taken for Boraflex, the model is insensitive to the presence or absence of perimeter wrappers. Figure 9.16 shows the axial view.

Figure 9.15: X-Y Representation of Region 3 KENO Model

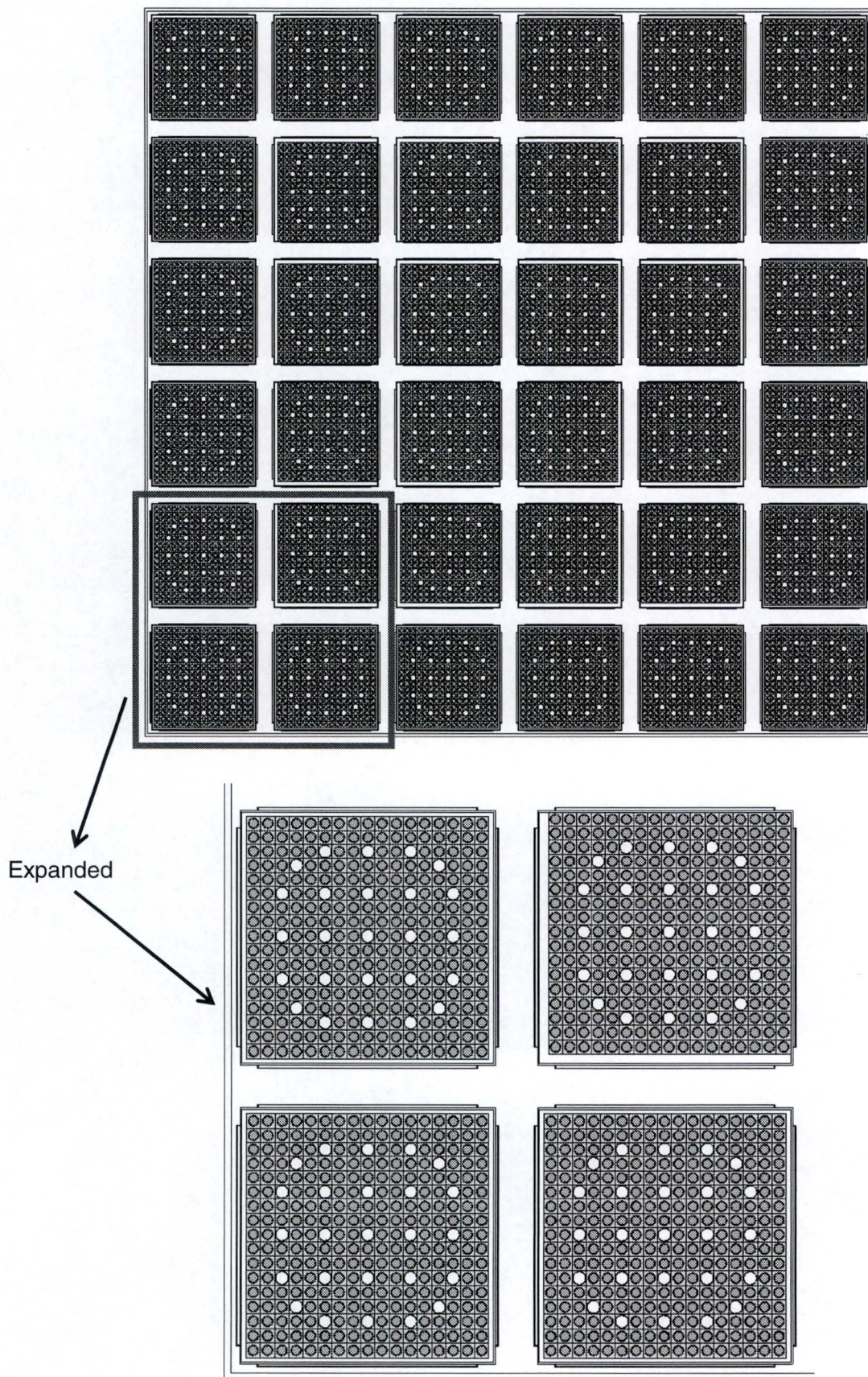
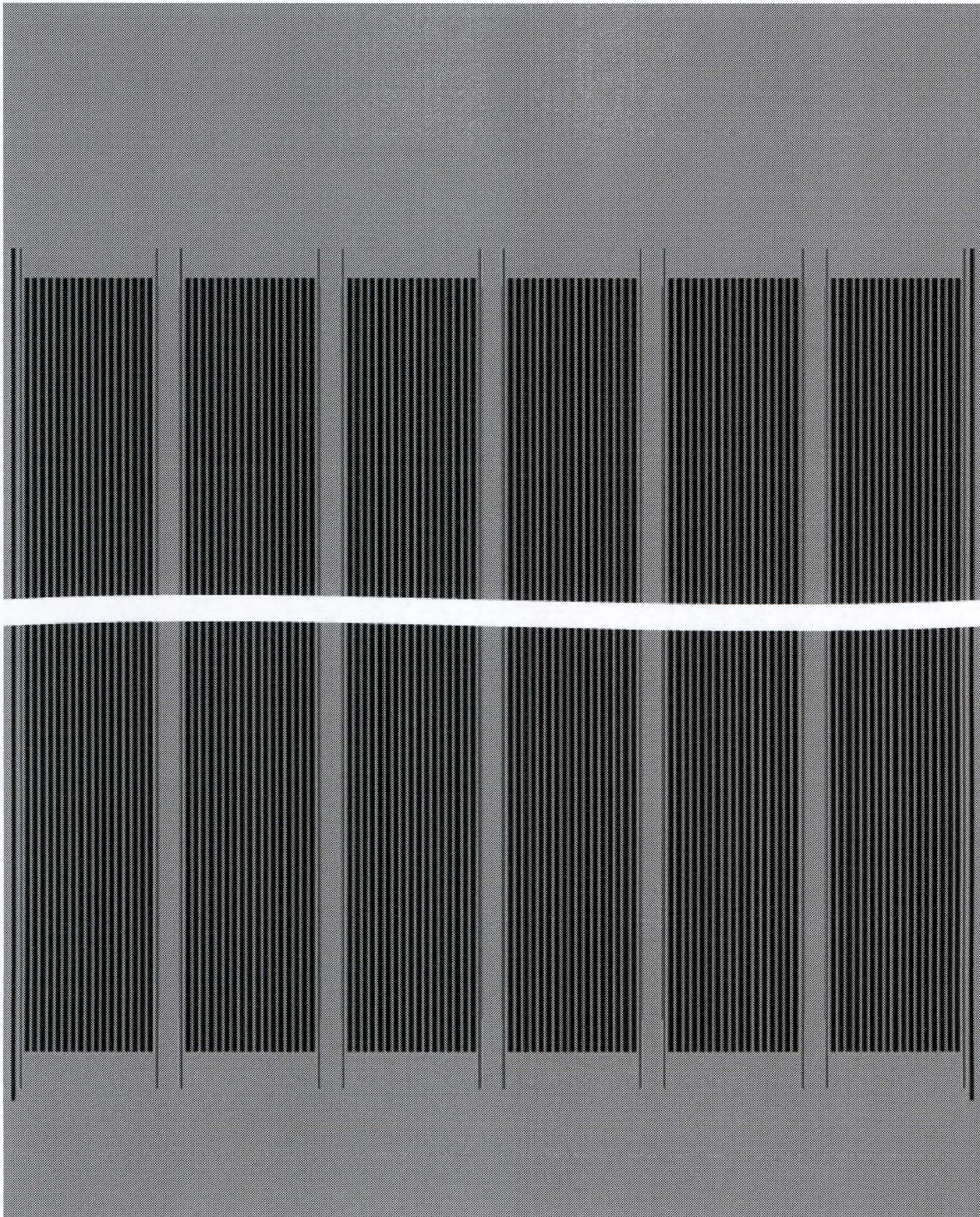


Figure 9.16: Axial Representation of Region 3 KENO Model



9.9.1 SFP Normal Operation Water Temperature and Density

Rack k-eff calculations are performed at three of the four water temperatures in Table 9.1 (32°F, 68°F, and 150°F).

9.9.2 TRITON Depletion Input for Burnup Credit

Depleted fuel content is calculated as described in Section 8. Uniform and NUREG/CR-6698 burnup profiles are used at 10, 20, and 30 GWd/MTU to determine the most reactive profile.

9.9.3 Asymmetric Fuel Placement

Asymmetric placement of fuel in the storage cell is evaluated by co-locating fuel in the center 4x4 group of storage cells into the corner of each cell toward the center of the 6x6 model. The remaining storage locations have fuel centered in each cell. Test cases for Region 3 confirm that asymmetric placement increases rack k-eff (~0.007 dk). The Region 3 KENO base model includes the 4x4 group of asymmetrically placed fuel.

9.9.4 Fuel Geometry Changes with Burnup

Region 3 depleted fuel cases use the same Zirlo-based clad creep and grid growth models as Region 2 cases. Clad creep and grid growth are evaluated as a bias.

One batch of fuel (Batch 6, introduced in Cycle 4 and discharged after Cycle 5) was built with Zircaloy-4 grids. Batch 7 (introduced in Cycle 5 and discharged after Cycle 9) was re-caged into assembly skeletons with Zircaloy-4 grids. For these 4.2 – 4.5 w/o fuel assemblies with 13.5-22.5 years decay time, the Region 3 burnup requirement (see Figure 9.20) is ~40 GWd/MTU or less.

At 40 GWd/MTU, the additional grid growth of Zircaloy-4 vs. ZIRLO is []^{a,c}.

Table 9.23 shows that the pin pitch tolerance is []^{a,c} of the nominal dimension, and Table 9.28 shows that tolerance is worth 0.0010 dk at 40 GWD/MTU. Therefore, based on fuel pitch tolerance cases, the reactivity effect of the grid growth difference is ~0.001 dk. Batch 6 and 7 fuel assemblies have 6 inch natural enrichment axial blankets, which are not credited in the bounding assembly design used to determine the burnup credit requirement. A 40 GWd/MTU comparison case was run with the top and bottom 6 inches of depleted enriched fuel replaced

with depleted natural enrichment fuel. Because the nodes are 8 inches high, the top and bottom nodes were split into 2 inch and 6 inch segments with the 2 inch portion containing enriched depleted fuel. The effect of the natural enrichment blankets in the Batch 6 and 7 fuel at 40 GWd/MTU is approximately -0.03, or about 30 times the positive effect of the additional Zircaloy-4 grid expansion. Crediting the axial blanket is more than sufficient to determine that no special storage requirements are needed for Batch 6 and 7 fuel.

9.9.5 Horizontal Burnup Tilt

The effect of horizontal burnup tilt is determined for Region 3 in the same manner as for Region 2 (Section 9.6.6). The two different 4x4 region burnup tilt orientations are shown in Figures 9.17 and 9.18. The lower burnup portion of each assembly is oriented towards the center of the model (Figure 9.17) or the center of each 2x2 cluster (Figure 9.18). A tilt bias is calculated using the KENO rack model k-effs from tilted and uniform burnup cases.

Figure 9.17: Region 3 Horizontal Burnup Tilt KENO Model #1

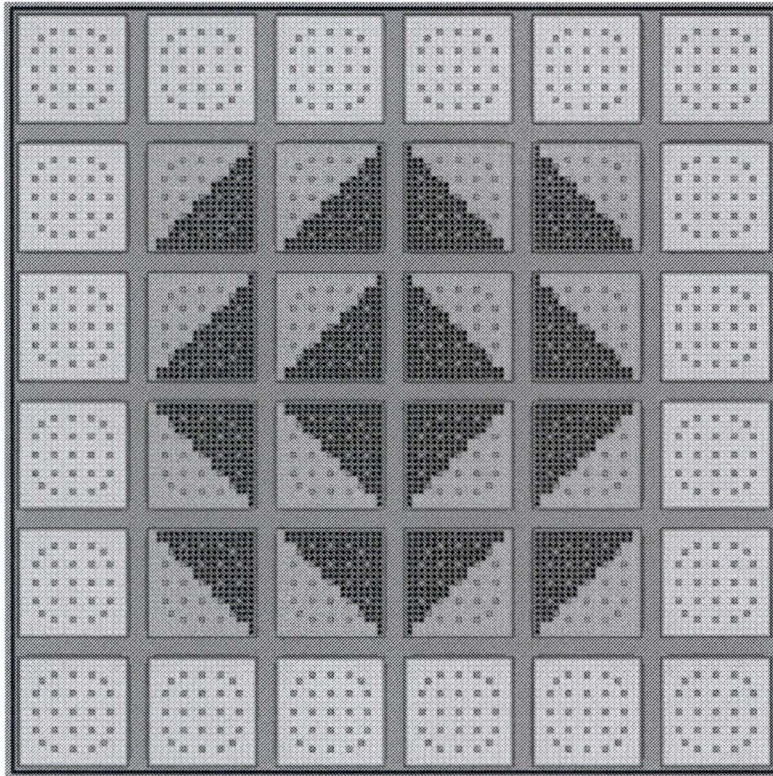
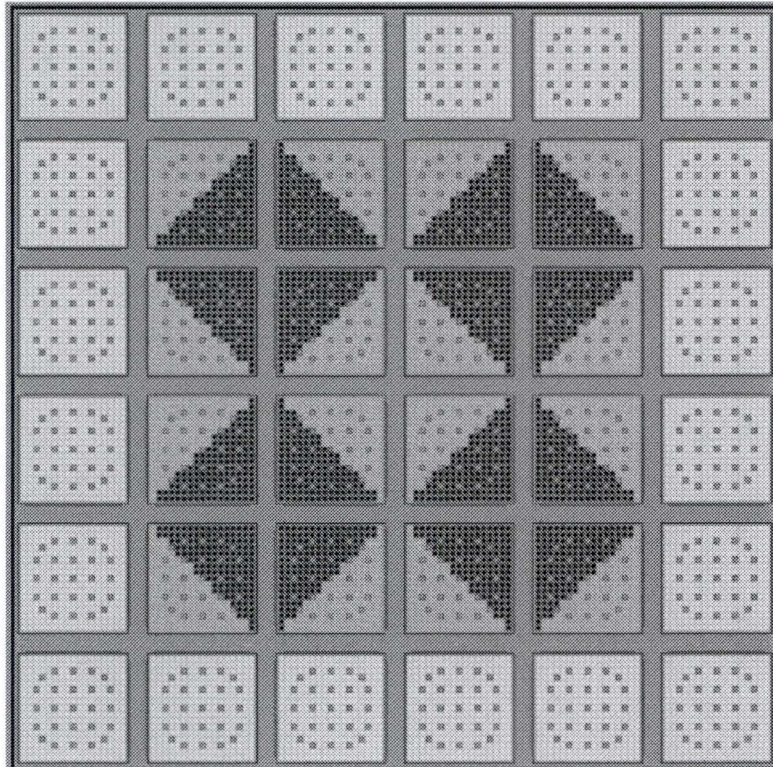


Figure 9.18: Region 3 Horizontal Burnup Tilt KENO Model #2



9.9.6 Code Validation Bias and Uncertainty

Validation of SCALE/KENO using ENDF/B-VII 238 group cross sections identified EALF as the primary bias and uncertainty trend. Because of the flux trap design with no credit for Boraflex, Region 3 KENO cases have $EALF \leq 0.35$ eV for all analyzed fuel burnups. The applicable bias and uncertainty for fresh fuel and depleted fuel (MOX validation cases) are shown in Table 9.22.

Unlike the Region 1 and 2 analyses, the limiting temperature in Region 3 is 150 °F, therefore, the validation temperature bias is applicable. The validation temperature bias (dk) is calculated by multiplying the ΔT above room temperature (K) by 1.7E-05 dk/K. At 150 °F the bias is 0.00078 dk.

To cover gaps in the validation for minor actinides and fission products, a bias of 1.5% of the minor actinide and fission product worth is added. [13]

Table 9.22: Region 3 Validation Bias and Uncertainty

Validation cases	Bias (dk)	Uncertainty (dk)
UO2 (EALF < 0.35)	0.0034	0.0048
MOX (EALF < 0.35)	0.0017	0.0083

9.9.7 Summary of Bias and Uncertainty

Table 9.23 summarizes the biases and uncertainties incorporated in the Region 3 analysis.

Table 9.23: Summary of Region 3 Biases, Uncertainties, and Conservatism

Analysis	Item / Type (B=bias, C=conservatism, U=uncertainty)	Explanation
Depletion	Burnable absorbers	C Bounding BPRA (24 WABA bounds IFBA).
Depletion	Soluble boron	C 1050 ppm bounds cycle average boron.
Depletion	Fuel and Water Temperature	C Bounding high node-specific. Section 8.2, 8.4, 8.5.
Depletion	Specific power	C Bounding high power assembly history.
Depletion	Axial burnup shapes	C Uniform and NUREG/CR-6801.
Depletion	Grids	C Maximum Zr grids
Depletion	Volatile fission products	C Reduced based on gap release fractions
Depletion	Fuel density (base case)	C 95.5% net bounds all fuel batches
Depletion	Power history	C Reduced 50% for the last 40 days of depletion
Depletion	Incore thimble	C Included in base depletion
Rack k-eff	Fuel density	U 95.5% [] ^{a,c} net bounds all fuel batches
Rack k-eff	Grids (base case)	C Minimum volume Zr grids
Rack k-eff	Wrapper thickness	U [] ^{a,c}
Rack k-eff	Rack wall thickness	U [] ^{a,c}
Rack k-eff	Fuel pin pitch	U 0.496 [] ^{a,c}
Rack k-eff	Fuel pellet OD	U 0.3225 [] ^{a,c}
Rack k-eff	Fuel clad ID	U 0.329 [] ^{a,c}
Rack k-eff	Fuel clad OD	U 0.374 [] ^{a,c}
Rack k-eff	Guide tube ID	U 0.442 [] ^{a,c}
Rack k-eff	Guide tube OD	U 0.474 [] ^{a,c}
Rack k-eff	Fuel stack height	U [] ^{a,c}
Rack k-eff	Burnup worth	U 5% of burnup worth
Rack k-eff	Measured burnup	U 4% of burnup worth
Rack k-eff	Enrichment	U [] ^{a,c}
Rack k-eff	Cell wall thickness	U [] ^{a,c}
Rack k-eff	Rack cell pitch	U [] ^{a,c}
Rack k-eff	Rack cell ID	U [] ^{a,c}
Rack k-eff	Rack pitch	U [] ^{a,c}
Rack k-eff	Code uncertainty	U Section 9.9.6
Rack k-eff	KENO case uncertainty	U 2 standard deviations
Rack k-eff	Minor actinides + FP	B 1.5% of worth.
Rack k-eff	Code bias	B Section 9.9.6; separate bias for temperature
Rack k-eff	Fuel axial position	B [] ^{a,c}
Rack k-eff	Temperature	C Most reactive temp. used (150 F)
Rack k-eff	Creep and grid growth	B Modified pin pitch and clad diameter
Rack k-eff	Horizontal burnup tilt	B Section 9.9.5
Rack k-eff	Eccentric placement	C Most reactive position (eccentric) used
Rack k-eff	NRC admin. margin	B 1% Δk

9.9.8 Unbounded Historical Fuel

Use of 24 WABA in the TRITON depletions leaves two batches of fuel potentially unbounded due to the burnable absorber used. In Cycle 1, some batch B and C fuel assemblies contained 24 Pyrex BPRA. Pyrex BP is somewhat more limiting than WABA because Pyrex rods displace slightly more water than WABA and have stainless steel clad instead of zirconium based clad. The effect of Pyrex BPRA depletion vs WABA depletion is estimated to be 0.00144 dk in Table 8.6 (three cycles of depletion with BP).

The assemblies in question have more than 25 years of decay time and have an enrichment which requires between 20 and 30 GWD/MTU on the burnup curve (Figure 9.20). The Region 3 calculated differential burnup worth between 20 and 30 GWd/MTU with 25 years decay time (Table 9.33) is 0.0068 Δk / GWD/MTU. Using this sensitivity, excess burnup of 0.21 GWD/MTU is sufficient to accommodate the Pyrex BPRA depletion reactivity effect. The minimum excess burnup for Batch B and C assemblies is 1.128 GWD/MTU, therefore no special requirement is needed for storage in Region 3 (or Region 2, which requires less burnup credit).

Ten fuel assemblies from Cycles 1 and 4 were identified in Section 8.8 as having control rod insertion history that exceeds 10 steps cycle average insertion that is bounded by the standard 24 WABA depletion. Sensitivity cases in Section 8.8 estimated the reactivity effect of depletion with a control rod inserted 10 steps vs. a control rod inserted 2 nodes (bounds all MPS3 cycles) to be ~ 0.02 dk for 3 cycles of depletion to 54 GWd/MTU. Cycle 4 lead bank assemblies have over 10 GWd/MTU more burnup than required by the burnup credit curves (Figure 9.20). The assemblies in question have more than 18 years of decay time and have an enrichment which requires between 30 and 40 GWD/MTU on the burnup curve (Figure 9.20). The Region 3 calculated differential burnup worth between 30 and 40 GWd/MTU with 18 years decay time (Table 9.32) is 0.0058 Δk / GWD/MTU. Using this sensitivity, excess burnup of 3.42 GWD/MTU is sufficient to accommodate the Cycle 4 assemblies with unbounded control rod insertion. Since all five Cycle 4 assemblies have more than 10 GWD/MTU of excess burnup, they can be stored in Region 3 with no restrictions.

The Cycle 1 assemblies were only depleted one cycle (~22 GWd/MTU), therefore the control rod effect cited is likely significantly overstated for them. Cycle 1 lead bank assemblies have ~30 years cooling time and a minimum of 1.6 GWd/MTU excess burnup (~-0.01 dk reactivity effect) as compared to the Region 3 25 year decay burnup curve. Rather than re-calculate the

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 166 of 300

control rod effect over one cycle of depletion, the assembly with the minimum excess burnup (B24) is explicitly modeled to directly confirm the as-depleted Cycle 1 fuel meets the requirements for storage in Region 3.

Table 9.24 shows the actual operating history of assembly B24 compared to the bounding model used to create the Region 3 burnup curves. To determine if the standard Region 3 modeling bounds B24, the in-rack k-eff of B24 is compared to the bounding model's 2.9 wt%, 20 GWD/T, 25 year decay time case (Table 9.33) which is used, in part, to determine the burnup curve. If assembly B24 has a lower best-estimate rack k-eff then it can be concluded that it is bounded by the standard model, because assembly B24 biases for quadrant burnup tilt and grid growth are lower than the standard model and other uncertainties (rack and fuel geometry tolerances) are the same.

Table 9.24: Depletion Conditions Comparison for Assembly B24

Parameter	Bounding Model	Assembly B24
Average RPD	1.44	1.160
Average Soluble Boron	1050 ppm	628 ppm
Core Average Moderator Temperature	592.3°F	586.9°F
Inlet Temperature	556.6°F	558.4°F
Core Power	3725 MWt	3411 MWt
RCS Flow	391,358 gpm	424,937 gpm
Bypass Flow	5.6%	5.6%
Axial Shape	NUREG Shapes	CMS5 Shapes
Axial Fuel Blankets	None	None
Fuel Density	0.955	[] ^{a,c}
Grids	Zirconium Grids	Inconel Grids
Poison Type	Full Length WABA	Control Rod (2 Node Insertion)
Instrument Tube	In-Core Thimble	Water Filled
G.T. and I.T. Dimensions	V5H	Standard
Quadrant BU Tilt ¹	-19.09%	-2.17%
Decay Time	25 Years	29.5 Years ³
Low Power EOL	40 days at 50% reduced power	40 days at 50% reduced power
Clad Creep Down ¹	Ignored in Depletion 0.7% Clad Creep in SFP	Ignored in Depletion 0.7% Clad Creep in SFP
Grid Growth ^{1,2}	Ignored in Depletion 0.025% Growth in SFP	Ignored in Depletion 0% Growth in SFP

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 167 of 300

¹These effects will not be modeled differently for B24 but are highlighted to show that B24 is more benign or equal to what is modeled for the burnup curves.

²Assembly B24 has Inconel grids. It is generally accepted that Inconel grids expand much less than zirconium based grids. At 21.736 GWD/MTU, ZIRLO experiences little growth so it is assumed that Inconel does not experience any growth.

³Calculated 4/26/17

Table 9.25 shows that Assembly B24 with 2 nodes of control rod insertion is substantially less reactive than the standard bounding assembly. Therefore, Cycle 1 lead bank assemblies do not require special treatment and may be stored normally in Region 3.

Table 9.25: Modeling Assembly B24 with Measured Design Inputs

Case Description	Burnup (GWD/MTU)	Enrich. (wt%)	Decay Time (Yr)	k-eff	σ	Δk
Bounding Model	20	2.90	25	0.96008	0.00007	N/A
Assembly B24 Model	21.739	2.90	29.5	0.95176	0.00007	-0.00833

9.9.9 Region 3 k-eff and Margin Calculation Without Decay Time Credit

With the exception of temperature changes, tolerance cases for Region 3 are performed at 150 °F with no soluble boron. The 150 °F base case has the highest k-eff of the three temperature cases. Table 9.26 shows the temperature cases as well as uniform and NUREG/CR-6801 axial profile cases for determination of the bounding profile at each burnup.

Table 9.26: Temperature and Burnup Shape Test Cases

Case Description	Burnup (GWD/MTU)	Enrich. (wt%)	k-eff	σ	Δk
Temperature - 32°F	0	1.70	0.96247	0.00007	-0.00612
Temperature - 68°F	0	1.70	0.96271	0.00007	-0.00587
Temperature - 150°F	0	1.70	0.96859	0.00007	N/A
Burnup Shape - NUREG	10	2.20	0.97399	0.00006	-0.00371
Burnup Shape - Uniform	10	2.20	0.97770	0.00007	N/A
Burnup Shape - NUREG	20	2.70	0.97130	0.00007	N/A
Burnup Shape - Uniform	20	2.70	0.96014	0.00006	-0.01116
Burnup Shape - NUREG	30	3.30	0.96414	0.00006	N/A
Burnup Shape - Uniform	30	3.30	0.94629	0.00006	-0.01785
Temperature - 32°F	54	5.00	0.94337	0.00007	-0.00744
Temperature - 150°F	54	5.00	0.95081	0.00006	N/A

Table 9.27 identifies the NUREG/CR-6801 Table 5 axial burnup shapes used to determine the TRITON depletion input.

Table 9.27: SFP Region 3 Limiting Axial Shapes

Assembly Average Burnup (GWd/MTU)	Limiting Shape (Uniform or NUREG/CR-6801 Table 5 Group)
0	Uniform
10	Uniform*
20	8*
30	5*
40	3
47.8	1
53,54	1

*Uniform and NUREG shapes checked. Limiting shape listed.

Not all uncertainty terms are calculated for each burnup/enrichment combination. A full set of uncertainty cases were run for the 0 GWD/MTU and 54 GWD/MTU burnup steps. To save computer run time, not all uncertainty cases were run for the intermediate burnup steps (10, 20, 30, 40 and 47.8 GWD/MTU). For the uncertainty cases not specifically run for a particular burnup, the maximum uncertainty value of the next lower or higher burnup is applied to the intermediate burnup steps. Bias terms are calculated for each burnup step.

Table 9.28 shows the results of base case, uncertainty, bias, and margin calculations for Region 3 with no credit for decay time beyond 5 days. Calculated uncertainty and bias values include 2 times the root sum square of the KENO case uncertainties. Shaded values were not calculated at that burnup.

Due to the iterative nature of determining points on the burnup curve with equal margin to the limit, the final base case k-eff at each burnup (excluding 47.8 GWD/MTU) are determined by interpolating between two base cases 0.05 wt% apart. This is acceptable because only the enrichment uncertainty term is a strong function of enrichment. These two base cases (0.05 wt% enrichment difference with their own set of TRITON depletion cases) are used to calculate enrichment sensitivity. At 40 GWd/MTU the final base case k-eff is a slight extrapolation (0.004 wt%). Depletion worth was multiplied by 5% to calculate the depletion worth uncertainty. The burnup measurement uncertainty was calculated by calculating the difference between the worth of the nominal burnup and the worth of 96% of the nominal burnup. The worth of 96% of the nominal burnup is calculated by linear interpolation between adjacent calculated burnup worths.

Uncertainty values are combined by root sum square. Bias values are added. NRC administrative margin of 0.01 dk is added to the total bias and uncertainty for the margin calculation. Dominion margin is 1.0 minus base case k-eff minus total bias and uncertainty. For depleted fuel, margin is calculated using fresh fuel and depleted fuel code validation bias and uncertainty. The option with the least margin is shown. The Region 3 burnup credit curve with no decay time credit, which matches or bounds all analyzed points, is shown in Figure 9.19.

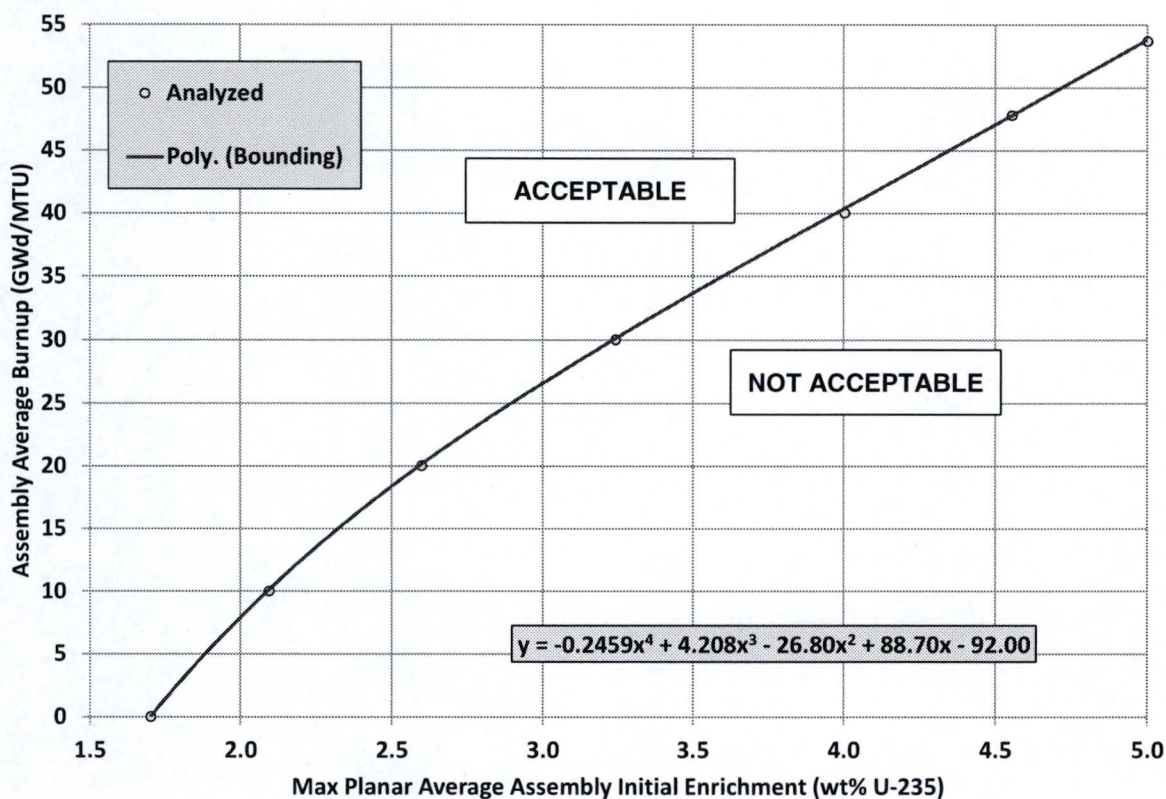
Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 171 of 300

Table 9.28: Region 3 Total Bias, Uncertainty, and Margin (0 years decay)

Burnup (GWd/MTU)	0	10	20	30	40	47.8	53.654
Enrichment (wt%)	1.703	2.095	2.599	3.244	4.004	4.555	5.00
Worths (dk)							
Minor Actinides and FP	0.0000	0.0515	0.0702	0.0914	0.1120	0.1218	0.1281
Burnup	0.0000	0.0676	0.1340	0.1949	0.2511	0.2811	0.3035
Uncertainties (dk)							
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009	0.0005
Enrichment, +0.05	0.0094	0.0055	0.0043	0.0036	0.0032	0.0027	0.0000
Pellet OD	0.0005	0.0005	0.0005	0.0005	0.0005	0.0005	0.0003
Active Fuel Length	0.0002	0.0002	0.0002	0.0002	0.0002	0.0003	0.0002
Clad ID	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Clad OD	0.0004	N/A	N/A	N/A	N/A	N/A	N/A
GT ID	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
GT OD	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Pin Pitch	0.0007	0.0009	0.0009	0.0010	0.0010	0.0011	0.0011
Cell Wall Thickness	0.0029	0.0027	0.0025	0.0026	0.0026	0.0026	0.0023
Cell I.D.	0.0000	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0031	0.0033	0.0031	0.0029	0.0030	0.0030	0.0029
Wrapper Thickness	0.0027	0.0026	0.0027	0.0025	0.0025	0.0026	0.0026
Wrapper Channel Thickness	0.0002	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003
Rack Wall Thickness	0.0004	0.0006	0.0006	0.0006	0.0006	0.0006	0.0006
Rack Pitch	0.0005	0.0006	0.0006	0.0006	0.0006	0.0006	0.0006
Depletion Worth Unc.	N/A	0.0034	0.0067	0.0097	0.0125	0.0140	0.0152
Burnup Measurement Unc.	N/A	0.0027	0.0053	0.0073	0.0090	0.0073	0.0081
Code Benchmarking Unc.	0.0048	0.0083	0.0083	0.0048	0.0048	0.0048	0.0048
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
RSS OF UNCERTAINTIES	0.0118	0.0121	0.0136	0.0145	0.0172	0.0175	0.0185
Biases (dk)							
Minor Actinides and Fission	N/A	0.0008	0.0011	0.0014	0.0017	0.0018	0.0019
Clad Creep and Grid Growth	N/A	0.0006	0.0011	0.0024	0.0041	0.0053	0.0061
Fuel Axial Position	0.0000	0.0000	0.0004	0.0004	0.0004	0.0004	0.0005
Radial Burnup Tilt	N/A	0.0018	0.0029	0.0016	0.0020	0.0020	0.0014
Code Temperature Bias	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008
Code Benchmarking Bias	0.0034	0.0017	0.0017	0.0034	0.0034	0.0034	0.0034
SUM OF BIASES	0.0042	0.0056	0.0079	0.0099	0.0123	0.0137	0.0141
Summary							
Base Case k-eff	0.9691	0.9665	0.9629	0.9606	0.9554	0.9539	0.9523
Total Bias and Uncertainty	0.0159	0.0177	0.0216	0.0244	0.0296	0.0312	0.0326
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9951	0.9942	0.9945	0.9950	0.9949	0.9951	0.9949
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0049	0.0058	0.0055	0.0050	0.0051	0.0049	0.0051

Figure 9.19: Region 3 Bounding Burnup Credit Curve (0 Years Decay)



Satisfaction of the burnup curve requirement shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.

9.9.10 Region 3 k-eff and Margin Calculation With Decay Time Credit

The same methodology is used to calculate the burnup curves with decay time that was used to calculate the burnup curve without decay time credit. TRITON depletions are performed with a larger decay time step at the end of the depletion. Decay times of 3, 9, 18, and 25 years are used.

Most of the uncertainty and bias values are taken from the no decay time credit analysis. However, four of the larger uncertainty and bias terms are calculated for each specific burnup: enrichment uncertainty, depletion worth uncertainty, burnup measurement uncertainty, and 1.5% of minor actinide and fission product worth bias. The other large uncertainty and bias terms are determined from code validation (Appendix A) and are not a function of decay time. Enrichment uncertainty is a function of enrichment. Allowable enrichment increases with increased decay time for a given burnup on the burnup curve. Burnup worth, which includes the effects of depletion and decay time, increases with increasing decay time. When uncertainties are root sum squared together, the four large terms (enrichment, depletion worth, burnup measurement, and code benchmarking) dominate the total uncertainty.

Table 9.29 shows the base and uncertainty cases run for fuel with the four decay times. Total uncertainty, bias, and margin calculations for Region 3 with decay time credit are provided in Table 9.30 (3 years decay), Table 9.31 (9 years decay), Table 9.32 (18 years decay), and Table 9.33 (25 years decay). The calculations follow the same methodology used for Region 3 with no decay time. Shaded values are retained from the no decay time analysis.

The 9, 18, and 25 year decay time burnup curves do not have an analyzed burnup point at >40 GWd/MTU like the 5 day and 3 year burnup curves. The 5 day and 3 year curves show that the points at >40 GWd/MTU are more limiting for curve shape determination than the 40 GWd/MTU point. To ensure that the 9, 18 and 25 year burnup curves are conservative for burnup >40 GWd/MTU, the curvature of the 5 day decay burnup curve will be used for all of the curves.

The decay time credit burnup curves are generated by multiplying the 5 day decay coefficients by a common multiple until all the data points are bounded. This method ensures that the shape of the decay curves is the same as the 5 day decay curve. The conservatism of this approach is confirmed in two ways.

First, as shown in Figure 9.20, using this method introduces progressively higher conservatism (curve burnup higher than analyzed burnup) at 10 and 40 GWd/MTU as decay time is

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 174 of 300

increased. The second confirmation is provided directly on the 3 year burnup curve, which has a 50 GWd/MTU analyzed point that is 0.09 GWd/MTU below the curve.

Fit coefficients are provided in Table 9.34.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 175 of 300

Table 9.29: Uncertainty Cases for Region 3 Decay Time Burnup Curves

Case Description	Burnup (GWD/MTU)	Enrich. (wt%)	Decay Time (Yrs)	k-eff	σ	Δk
Base Case	10	2.15	3	0.96777	0.00006	N/A
Interpolation Case	10	2.10	3	0.96243	0.00006	-0.00533
Burnup Worth	0	2.15	N/A	1.04166	0.00008	0.07409
Base Case	20	2.65	3	0.96097	0.00006	N/A
Interpolation Case	20	2.70	3	0.96523	0.00007	0.00426
Burnup Worth	0	2.65	N/A	1.10218	0.00007	0.14140
Base Case	30	3.35	3	0.95933	0.00006	N/A
Interpolation Case	30	3.40	3	0.96277	0.00006	0.00344
Burnup Worth	0	3.35	N/A	1.16391	0.00007	0.20476
Base Case	40	4.15	3	0.95300	0.00008	N/A
Interpolation Case	40	4.20	3	0.95594	0.00006	0.00293
Burnup Worth	0	4.15	N/A	1.21441	0.00007	0.26163
Base Case	50	4.90	3	0.95160	0.00006	N/A
Interpolation Case	50	4.95	3	0.95407	0.00006	0.00247
Burnup Worth	0	4.90	N/A	1.25019	0.00008	0.29879
Base Case	10	2.20	9	0.96644	0.00006	N/A
Interpolation Case	10	2.15	9	0.96082	0.00006	-0.00562
Burnup Worth	0	2.20	N/A	1.04858	0.00007	0.08234
Base Case	20	2.75	9	0.95994	0.00006	N/A
Interpolation Case	20	2.80	9	0.96421	0.00006	0.00427
Burnup Worth	0	2.75	N/A	1.11253	0.00006	0.15277
Base Case	30	3.55	9	0.95982	0.00007	N/A
Interpolation Case	30	3.50	9	0.95630	0.00008	-0.00352
Burnup Worth	0	3.55	N/A	1.17811	0.00008	0.21850
Base Case	40	4.45	9	0.95401	0.00007	N/A
Interpolation Case	40	4.40	9	0.95086	0.00007	-0.00314
Burnup Worth	0	4.45	N/A	1.22998	0.00007	0.27617
Base Case	10	2.25	18	0.96498	0.00007	N/A
Interpolation Case	10	2.20	18	0.95937	0.00006	-0.00561
Burnup Worth	0	2.25	N/A	1.05510	0.00006	0.09030
Base Case	20	2.85	18	0.95979	0.00006	N/A
Interpolation Case	20	2.90	18	0.96425	0.00007	0.00446
Burnup Worth	0	2.85	N/A	1.12216	0.00007	0.16256
Base Case	30	3.70	18	0.95820	0.00007	N/A
Interpolation Case	30	3.75	18	0.96172	0.00007	0.00352
Burnup Worth	0	3.70	N/A	1.18807	0.00007	0.23006

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 176 of 300

Case Description	Burnup (GWD/MTU)	Enrich. (wt%)	Decay Time (Yrs)	k-eff	σ	Δk
Base Case	40	4.70	18	0.95358	0.00007	N/A
Interpolation Case	40	4.65	18	0.95075	0.00007	-0.00283
Burnup Worth	0	4.70	N/A	1.24152	0.00007	0.28814
Base Case	10	2.30	25	0.96720	0.00006	N/A
Interpolation Case	10	2.25	25	0.96170	0.00006	-0.00550
Burnup Worth	0	2.30	N/A	1.06179	0.00006	0.09476
Base Case	20	2.90	25	0.96008	0.00007	N/A
Interpolation Case	20	2.95	25	0.96452	0.00006	0.00443
Burnup Worth	0	2.90	N/A	1.12660	0.00007	0.16671
Base Case	30	3.80	25	0.95947	0.00006	N/A
Interpolation Case	30	3.85	25	0.96294	0.00007	0.00347
Burnup Worth	0	3.80	N/A	1.19441	0.00007	0.23512
Base Case	40	4.80	25	0.95243	0.00007	N/A
Interpolation Case	40	4.85	25	0.95528	0.00007	0.00285
Burnup Worth	0	4.80	N/A	1.24573	0.00007	0.29350

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 177 of 300

Table 9.30: Region 3 Total Bias, Uncertainty, and Margin (3 years decay)

Burnup (GWd/MTU)	0	10	20	30	40	50
Enrichment (wt%)	1.703	2.137	2.670	3.363	4.183	4.911
Worths (dk)						
Minor Actinides and FP	0.0000	0.0525	0.0685	0.0890	0.1096	0.1213
Burnup	0.0000	0.0741	0.1414	0.2048	0.2616	0.2988
Uncertainties (dk)						
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009	0.0005
Enrichment, +0.05	0.0094	0.0055	0.0044	0.0036	0.0031	0.0026
Pellet OD	0.0005	0.0005	0.0005	0.0005	0.0005	0.0003
Active Fuel Length	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Clad ID	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Clad OD	0.0004	N/A	N/A	N/A	N/A	N/A
GT ID	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
GT OD	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002
Pin Pitch	0.0007	0.0009	0.0009	0.0010	0.0010	0.0011
Cell Wall Thickness	0.0029	0.0027	0.0025	0.0026	0.0026	0.0023
Cell I.D.	0.0000	0.0002	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0031	0.0033	0.0031	0.0029	0.0030	0.0029
Wrapper Thickness	0.0027	0.0026	0.0027	0.0025	0.0025	0.0026
Wrapper Channel Thickness	0.0002	0.0003	0.0003	0.0003	0.0003	0.0003
Rack Wall Thickness	0.0004	0.0006	0.0006	0.0006	0.0006	0.0006
Rack Pitch	0.0005	0.0006	0.0006	0.0006	0.0006	0.0006
Depletion Worth Unc.	N/A	0.0037	0.0071	0.0102	0.0131	0.0149
Burnup Measurement Unc.	N/A	0.0030	0.0054	0.0076	0.0091	0.0074
Code Benchmarking Unc.	0.0048	0.0083	0.0083	0.0048	0.0048	0.0048
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0002	0.0001
RSS OF UNCERTAINTIES	0.0118	0.0122	0.0139	0.0149	0.0177	0.0182
Biases (dk)						
Minor Actinides and Fission Products	N/A	0.0008	0.0010	0.0013	0.0016	0.0018
Clad Creep and Grid Growth	N/A	0.0006	0.0011	0.0024	0.0041	0.0061
Fuel Axial Position	0.0000	0.0000	0.0004	0.0004	0.0004	0.0005
Radial Burnup Tilt	N/A	0.0018	0.0029	0.0016	0.0020	0.0021
Code Temperature Bias	0.0008	0.0008	0.0008	0.0008	0.0008	0.0008
Code Benchmarking Bias	0.0034	0.0017	0.0017	0.0034	0.0034	0.0034
SUM OF BIASES	0.0042	0.0056	0.0079	0.0099	0.0123	0.0145
Summary						
Base Case k-eff	0.9691	0.9664	0.9627	0.9602	0.9549	0.9521
Total Bias and Uncertainty	0.0159	0.0179	0.0218	0.0248	0.0300	0.0327
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9951	0.9942	0.9945	0.9950	0.9949	0.9949
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0049	0.0058	0.0055	0.0050	0.0051	0.0051

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 178 of 300

Table 9.31: Region 3 Total Bias, Uncertainty, and Margin (9 years decay)

Burnup (GWd/MTU)	0	10	20	30	40
Enrichment (wt%)	1.703	2.196	2.778	3.547	4.456
Worths (dk)					
Minor Actinides and FP	0.0000	0.0526	0.0663	0.0860	0.1061
Burnup	0.0000	0.0823	0.1528	0.2185	0.2762
Uncertainties (dk)					
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009
Enrichment, +0.05	0.0094	0.0058	0.0044	0.0037	0.0033
Pellet OD	0.0005	0.0005	0.0005	0.0005	0.0005
Active Fuel Length	0.0002	0.0002	0.0002	0.0002	0.0002
Clad ID	0.0002	0.0002	0.0002	0.0002	0.0002
Clad OD	0.0004	N/A	N/A	N/A	N/A
GT ID	0.0002	0.0002	0.0002	0.0002	0.0002
GT OD	0.0002	0.0002	0.0002	0.0002	0.0002
Pin Pitch	0.0007	0.0009	0.0009	0.0010	0.0010
Cell Wall Thickness	0.0029	0.0027	0.0025	0.0026	0.0026
Cell I.D.	0.0000	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0031	0.0033	0.0031	0.0029	0.0030
Wrapper Thickness	0.0027	0.0026	0.0027	0.0025	0.0025
Wrapper Channel Thickness	0.0002	0.0003	0.0003	0.0003	0.0003
Rack Wall Thickness	0.0004	0.0006	0.0006	0.0006	0.0006
Rack Pitch	0.0005	0.0006	0.0006	0.0006	0.0006
Depletion Worth Unc.	N/A	0.0041	0.0076	0.0109	0.0138
Burnup Measurement Unc.	N/A	0.0033	0.0056	0.0079	0.0092
Code Benchmarking Unc.	0.0048	0.0083	0.0083	0.0048	0.0048
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0001
RSS OF UNCERTAINTIES	0.0118	0.0126	0.0143	0.0156	0.0183
Biases (dk)					
Minor Actinides and Fission Products	N/A	0.0008	0.0010	0.0013	0.0016
Clad Creep and Grid Growth	N/A	0.0006	0.0011	0.0024	0.0041
Fuel Axial Position	0.0000	0.0000	0.0004	0.0004	0.0004
Radial Burnup Tilt	N/A	0.0018	0.0029	0.0016	0.0020
Code Temperature Bias	0.0008	0.0008	0.0008	0.0008	0.0008
Code Benchmarking Bias	0.0034	0.0017	0.0017	0.0034	0.0034
SUM OF BIASES	0.0042	0.0056	0.0079	0.0098	0.0123
Summary					
Base Case k-eff	0.9691	0.9660	0.9623	0.9596	0.9544
Total Bias and Uncertainty	0.0159	0.0182	0.0222	0.0254	0.0306
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9951	0.9942	0.9945	0.9950	0.9949
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0049	0.0058	0.0055	0.0050	0.0051

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 179 of 300

Table 9.32: Region 3 Total Bias, Uncertainty, and Margin (18 years decay)

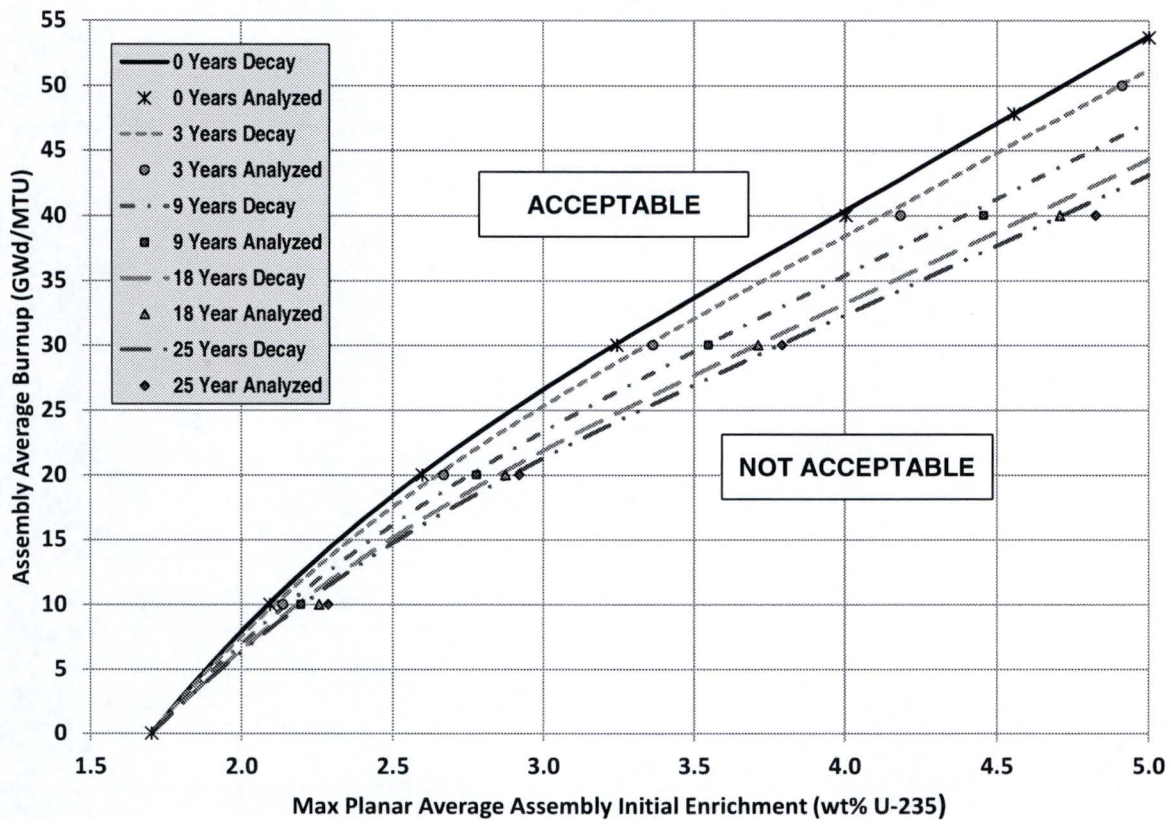
Burnup (GWd/MTU)	0	10	20	30	40
Enrichment (wt%)	1.703	2.257	2.874	3.713	4.707
Worths (dk)					
Minor Actinides and FP	0.0000	0.0529	0.0640	0.0830	0.1027
Burnup	0.0000	0.0903	0.1626	0.2301	0.2881
Uncertainties (dk)					
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009
Enrichment, +0.05	0.0094	0.0058	0.0047	0.0037	0.0030
Pellet OD	0.0005	0.0005	0.0005	0.0005	0.0005
Active Fuel Length	0.0002	0.0002	0.0002	0.0002	0.0002
Clad ID	0.0002	0.0002	0.0002	0.0002	0.0002
Clad OD	0.0004	N/A	N/A	N/A	N/A
GT ID	0.0002	0.0002	0.0002	0.0002	0.0002
GT OD	0.0002	0.0002	0.0002	0.0002	0.0002
Pin Pitch	0.0007	0.0009	0.0009	0.0010	0.0010
Cell Wall Thickness	0.0029	0.0027	0.0025	0.0026	0.0026
Cell I.D.	0.0000	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0031	0.0033	0.0031	0.0029	0.0030
Wrapper Thickness	0.0027	0.0026	0.0027	0.0025	0.0025
Wrapper Channel Thickness	0.0002	0.0003	0.0003	0.0003	0.0003
Rack Wall Thickness	0.0004	0.0006	0.0006	0.0006	0.0006
Rack Pitch	0.0005	0.0006	0.0006	0.0006	0.0006
Depletion Worth Unc.	N/A	0.0045	0.0081	0.0115	0.0144
Burnup Measurement Unc.	N/A	0.0036	0.0058	0.0081	0.0093
Code Benchmarking Unc.	0.0048	0.0083	0.0083	0.0048	0.0048
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0001
RSS OF UNCERTAINTIES	0.0118	0.0128	0.0147	0.0161	0.0187
Biases (dk)					
Minor Actinides and Fission Products	N/A	0.0008	0.0010	0.0012	0.0015
Clad Creep and Grid Growth	N/A	0.0006	0.0011	0.0024	0.0041
Fuel Axial Position	0.0000	0.0000	0.0004	0.0004	0.0004
Radial Burnup Tilt	N/A	0.0018	0.0029	0.0016	0.0020
Code Temperature Bias	0.0008	0.0008	0.0008	0.0008	0.0008
Code Benchmarking Bias	0.0034	0.0017	0.0017	0.0034	0.0034
SUM OF BIASES	0.0042	0.0056	0.0078	0.0098	0.0122
Summary					
Base Case k-eff	0.9691	0.9658	0.9619	0.9591	0.9540
Total Bias and Uncertainty	0.0159	0.0184	0.0225	0.0259	0.0309
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9951	0.9942	0.9945	0.9950	0.9949
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0049	0.0058	0.0055	0.0050	0.0051

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 180 of 300

Table 9.33: Region 3 Total Bias, Uncertainty, and Margin (25 years decay)

Burnup (GWd/MTU)	0	10	20	30	40
Enrichment (wt%)	1.703	2.287	2.920	3.792	4.824
Worths (dk)					
Minor Actinides and FP	0.0000	0.0528	0.0629	0.0815	0.1007
Burnup	0.0000	0.0948	0.1667	0.2351	0.2935
Uncertainties (dk)					
Fuel Stack PTD	0.0009	0.0009	0.0009	0.0009	0.0009
Enrichment, +0.05	0.0094	0.0057	0.0046	0.0037	0.0030
Pellet OD	0.0005	0.0005	0.0005	0.0005	0.0005
Active Fuel Length	0.0002	0.0002	0.0002	0.0002	0.0002
Clad ID	0.0002	0.0002	0.0002	0.0002	0.0002
Clad OD	0.0004	N/A	N/A	N/A	N/A
GT ID	0.0002	0.0002	0.0002	0.0002	0.0002
GT OD	0.0002	0.0002	0.0002	0.0002	0.0002
Pin Pitch	0.0007	0.0009	0.0009	0.0010	0.0010
Cell Wall Thickness	0.0029	0.0027	0.0025	0.0026	0.0026
Cell I.D.	0.0000	0.0002	0.0002	0.0002	0.0002
Cell Pitch	0.0031	0.0033	0.0031	0.0029	0.0030
Wrapper Thickness	0.0027	0.0026	0.0027	0.0025	0.0025
Wrapper Channel Thickness	0.0002	0.0003	0.0003	0.0003	0.0003
Rack Wall Thickness	0.0004	0.0006	0.0006	0.0006	0.0006
Rack Pitch	0.0005	0.0006	0.0006	0.0006	0.0006
Depletion Worth Unc.	N/A	0.0047	0.0083	0.0118	0.0147
Burnup Measurement Unc.	N/A	0.0038	0.0058	0.0082	0.0093
Code Benchmarking Unc.	0.0048	0.0083	0.0083	0.0048	0.0048
KENO Case Uncertainty	0.0001	0.0001	0.0001	0.0001	0.0001
RSS OF UNCERTAINTIES	0.0118	0.0129	0.0148	0.0163	0.0190
Biases (dk)					
Minor Actinides and Fission Products	N/A	0.0008	0.0009	0.0012	0.0015
Clad Creep and Grid Growth	N/A	0.0006	0.0011	0.0024	0.0041
Fuel Axial Position	0.0000	0.0000	0.0004	0.0004	0.0004
Radial Burnup Tilt	N/A	0.0018	0.0029	0.0016	0.0020
Code Temperature Bias	0.0008	0.0008	0.0008	0.0008	0.0008
Code Benchmarking Bias	0.0034	0.0017	0.0017	0.0034	0.0034
SUM OF BIASES	0.0042	0.0056	0.0078	0.0097	0.0122
Summary					
Base Case k-eff	0.9691	0.9658	0.9619	0.9589	0.9538
Total Bias and Uncertainty	0.0159	0.0185	0.0226	0.0261	0.0311
NRC Administrative Margin	0.0100	0.0100	0.0100	0.0100	0.0100
Maximum k-eff	0.9951	0.9943	0.9945	0.9950	0.9949
10CFR50.68 Limit	1.0000	1.0000	1.0000	1.0000	1.0000
Dominion Margin (dk)	0.0049	0.0057	0.0055	0.0050	0.0051

Figure 9.20: Region 3 Bounding Burnup Credit Curves

The burnup curve equations have the following polynomial format:

$$BU \text{ [GWD/MTU]} = a_4 * \text{wt}\%^4 + a_3 * \text{wt}\%^3 + a_2 * \text{wt}\%^2 + a_1 * \text{wt}\%^1 + a_0 * \text{wt}\%^0$$

wt%: maximum planar average, as-built U-235 enrichment

Table 9.34: Region 3 Burnup Credit Curve Coefficients

Decay Time Credit	a_4	a_3	a_2	a_1	a_0
No Credit	-0.2459	4.208	-26.80	88.70	-92.00
3 Years	-0.2338	4.001	-25.48	84.34	-87.47
9 Years	-0.2153	3.684	-23.46	77.66	-80.54
18 Years	-0.2020	3.458	-22.02	72.88	-75.59
25 Years	-0.1964	3.361	-21.40	70.84	-73.47

Satisfaction of the burnup curve requirement shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.
- 3) The fuel assembly decay time (time elapsed since last use at power in the reactor core) is greater than or equal to the applicable burnup curve.

9.10 *Region 3 Analysis Summary*

Region 3 has been demonstrated to satisfy the SFP k-eff requirement for spent fuel pool analyses as follows:

- $k\text{-eff} < 1.0$ if flooded with unborated water (95% probability, 95% confidence)

In Region 3 with 4-out-of-4 storage, the requirement is met with either a restrictive loading curve (burnup credit curve) of minimum required assembly measured burnup versus initial fuel assembly as-built enrichment (maximum planar volume averaged as-built initial enrichment in the assembly) and with minimum decay time.

Region 3 analysis includes calculation and application of bias and uncertainty as well as identification of NRC administrative margin and Dominion margin to the k-eff limit. For Region 3, five 4th order polynomial burnup credit curves are determined for fuel with initial enrichment >1.703 wt% U-235 (Table 9.34).

10 Interface Analysis

10.1 Interfaces Between Storage Configurations Within a Rack

Only Region 1 has two different sub-regions with different storage requirements within the same rack. The Region 1A / 1B interface is directly modeled and dispositioned in Section 9.3.

10.2 Interfaces Between Dissimilar Racks

Analysis of Region interfaces will determine if the proximity of one Region to another affects the conclusions of the individual Region infinite lattice analyses. The MPS3 interface analysis confirms that interface effects do not reduce the minimum margin to the k-eff limit determined for Regions 1, 2, and 3. Part of that confirmation is justification of appropriate interface model bias and uncertainty. Reference 4 provides this guidance for analysis of storage configuration interfaces:

Interfaces: *For applications that contain more than a single storage configuration, in order to ensure that the regulatory requirement for keff to be known with a 95 percent probability at a 95 percent confidence level is met the NCS analysis should consider the interface between storage configurations. Given all the combinations that are in existence, it is impossible to predict all of the combinations that could be proposed. Therefore, the staff should verify that each application includes a portion of the analysis that demonstrates that the interface analysis used is appropriate for its specific conditions.*

- i. *Absent a determination of a set of biases and uncertainties specifically for the combined interface model, use of the maximum biases and uncertainties from the individual storage configurations should be acceptable in determining whether the keff of the combined interface model meets the regulatory requirements.*

For some interfaces, use of the maximum total bias and uncertainty from the individual Regions is adequate. This is the simplest approach. For some interfaces, a set of biases and uncertainties are developed specifically for the interface configuration.

In the interface analysis, the KENO best estimate k-eff for a model representing the nominal fuel and rack design is referred to as "base case k-eff". The k-eff that is compared to the regulatory limit that includes allowance for total bias and uncertainty is referred to as the 95/95 k-eff. KENO rack models are developed using Region 1, 2, and 3 infinite lattice models. Depleted fuel isotopic content is the same as used for Region 1, 2, and 3 infinite lattice analyses.

10.2.1 MPS3 Region Interfaces

Region interfaces can be seen in Figure 4.3.

- 1) Region 2 meets Region 1
- 2) Region 2 meets Region 3

10.2.2 Interface Analysis Method

Interface analysis methodology is summarized in the steps below. Refer to Figures 10.1 and 10.3 for an illustration of the KENO interface models.

- Develop a KENO interface model with two Regions (one on each side of the interface).
- Choose the number of storage cells in each Region along the interface to minimize model gaps at the model boundary due to different cell pitch on either side of the interface. This is intended to maintain fidelity to the infinite lattice analysis as closely as possible.
- Use reflective or periodic boundary conditions as appropriate to maintain fidelity with infinite lattice analysis.
- Compare the interface model k-eff to the infinite lattice analysis "base case" k-eff for each Region.
- For a Region with burnup credit, perform interface calculations with fresh fuel and with maximally depleted fuel to capture potential axial shape effects at the interface.
- For two Regions with different bounding temperature, model the interface at both bounding temperatures.

There are several possible results in the interface analysis:

- 1) Interface model k-eff is greater than both infinite lattice Region base case k-effs. The interface is "adverse" because it increases k-eff as compared to both infinite lattice analyses. The infinite lattice 95/95 k-eff margin calculations are not bounding for the interface model. The MPS3 analysis has no adverse interfaces.
- 2) Interface model k-eff is less than both infinite lattice Region base case k-effs. The interface is "benign", meaning the interface reduces k-eff as compared to both infinite lattice analyses. The infinite lattice margin calculations are bounding. No additional analysis is needed. If the highest Region total bias and uncertainty is added to the interface model k-eff, the 95/95 k-eff is lower than the infinite lattice Region 95/95 k-eff.
- 3) Interface model k-eff is less than one infinite lattice Region base case k-eff and greater than the other. This is an "indeterminate" interface.

With an indeterminate interface, neutron leakage across the interface from the higher reactivity Region (Region "A") into the lower reactivity Region (Region "B") results in a base case k-eff less than Region A and greater than Region B. It is not clear what total bias and uncertainty value should be applied to the interface model to determine the interface model 95/95 k-eff.

One way to determine the 95/95 k-eff for an indeterminate interface is to use the higher total bias and uncertainty of Region A or Region B for the interface model. This typically results in an interface model 95/95 k-eff higher than either infinite lattice Region analysis. This approach may be unnecessarily conservative and result in apparent margin to the limit that may be insufficient.

Alternatively, the total bias and uncertainty of the interface model can be directly calculated. In the calculation of total bias and uncertainty, there are typically a few dominant terms. For the purpose of determining interface model bias and uncertainty, it is not necessary to calculate all bias and uncertainty items. MPS3 indeterminate interfaces are evaluated by calculating dominant k-eff tolerance values for each Region in the interface model independently and comparing them to the same tolerance values calculated for the infinite lattice Region analyses. This comparison is used to determine Region weighting factors, which are then used to calculate a weighted total bias and uncertainty value for the interface model.

10.3 *Region 1-2 Interface*

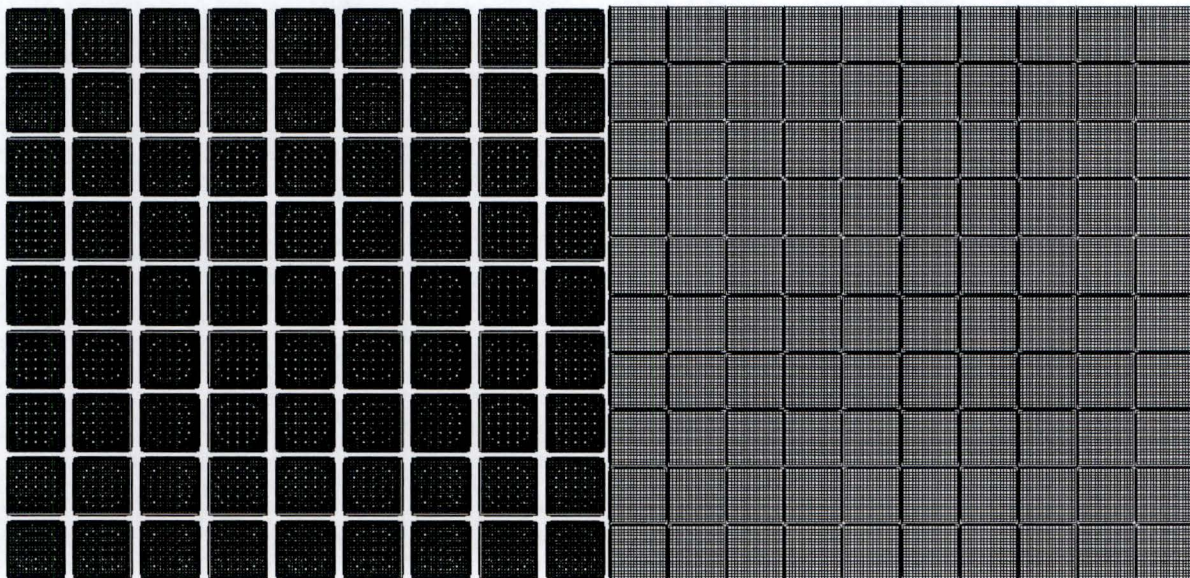
Region 1 and Region 2 are both most reactive at 32 °F. Interface cases are run at 32 °F.

Region 1 is represented by the infinite lattice model (4.75 wt% fresh fuel with no SFP wall neutron leakage boundary), which bounds the k-eff of the wall credit model. Due to the rack placement in the SFP, only the 4.75 w/o fresh fuel side of Region 1 has an interface with Region 2. Region 1 with 5.0 wt% fuel (Table 9.6) requires burnup credit of 2 GWd/MTU, which is well below the burnup at which non-uniform axial burnup shapes become more reactive than uniform shapes. The 5.0 wt% 2 GWd/MTU condition has over 0.01 dk more margin than the fresh fuel case and the 12 IFBA rods condition has over 0.02 dk more margin than the fresh fuel case. Because of these features, only fresh 4.75 wt% fuel is used to represent Region 1 in the interface models.

Region 2 burnup credit varies from 0 MWd/MTU at 2.03 wt% to 40.25 MWd/MTU at 5.0 wt%. To investigate possible interface axial burnup shape interference effects, Region 1-2 interface models include 2.03 wt% fresh fuel and 40 GWd/MTU 5.0 wt% depleted fuel (NUREG axial burnup shape).

The Region 1-2 interface model is formed from a 9x9 cell Region 1 adjacent to a 10x10 cell Region 2 (Figure 10.1). These array sizes are chosen to closely match the overall dimension of each Region along the common interface (prevents non-physical water gaps across the periodic boundary of the model). Note that the Region 1 cell pitch in the x-coordinate direction is different than cell pitch in the y-coordinate direction. The longer pitch is perpendicular to the interface.

Figure 10.1: X-Y Representation of the Region 1-2 Interface Model

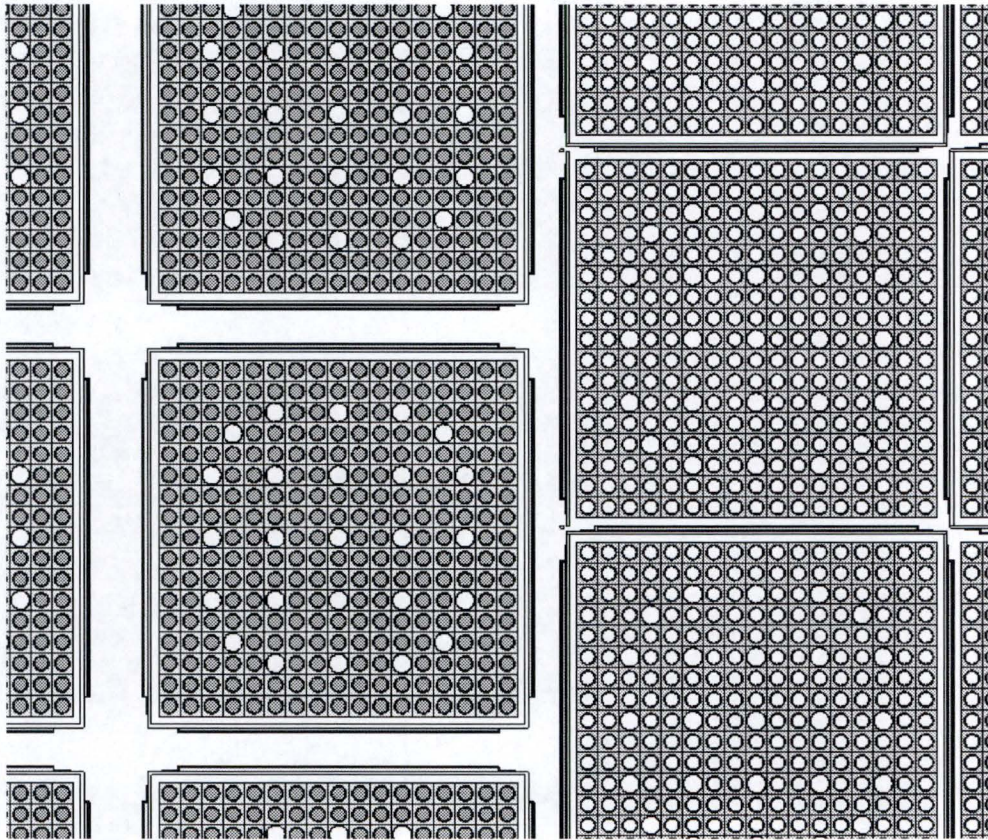


With periodic boundary conditions, the Region 1-2 model represents infinite “columns” of Region 1 and 2 cells in the Y dimension with a very small 0.17 inch water gap between groupings of 9 Region 1 cells (this occurs because the Region 2 10x10 group is 0.17 inches longer in the y-direction than the Region 1 9x9 group). The gap is kept as small as possible to maintain fidelity between the infinite lattice Region analysis and the interface models. Only the intra-rack gap resulting from the water at the edge of the infinite lattice models (a result of modeling normal cell pitch) is credited. No additional intra-rack spacing is added for the nominal 1.5 inch spacing between Region 1 and 2.

The interface model has 0.62 inches between the Region 1 and Region 2 wrappers (Figure 10.2). The rack baseplates extend a total of 1.5 inches outside the faces of the outer storage cell walls. Accounting for the extension of the Boral wrappers beyond the cell wall leads to a minimum design spacing between adjacent wrappers across the Region 1-2 interface of 1.2 inches. The model is conservative because it has rack cells ~0.6 inch closer together than is physically possible and the flux trap between Region 1 and 2 has less water gap than actual. Periodic boundary conditions are used in the X and Y direction. Reflective boundary conditions are used in the Z direction.

Note that depleted fuel base case k-eff for the Region 2 infinite lattice model in this section does not match the k-eff in Table 9.19. Region 2 interface model depleted fuel cases are modeled with Zirc2 rather than Zr clad material. To maintain modeling consistency in the comparison of infinite lattice k-eff and interface model k-eff, the Region 2 infinite lattice k-eff with Zirc2 clad modeling is reported. Region 2 sensitivity cases were also run with Zirc-2 cladding, therefore the sensitivity comparison is also consistent.

Figure 10.2: X-Y Representation of the Region 1-2 Interface Showing Flux Traps



10.3.1 Region 1-2 Interface with Region 2 Fresh Fuel

Table 10.1 summarizes infinite lattice and interface model results with fresh fuel in both Regions. All cases have fuel centered in the rack cells which maximizes k-eff in Regions 1 and 2. The interface model k-eff is bounded by the infinite lattice analysis k-eff of Region 1 and Region 2. The final case in Table 10.1 confirms that modestly increasing intra-rack spacing reduces k-eff. No additional Region 1-2 interface analysis is needed for this condition.

Table 10.1: Region 1-2 Interface with Fresh Fuel

Enrichment (wt% U235)	Region	k-eff	Uncert.	Note
4.75 / N/A	1	0.96745	0.00008	Inifinite lattice Reg. 1
N/A / 2.03	2	0.96872	0.00006	Inifinite lattice Reg. 2
4.75 / 2.03	1-2	0.96661	0.00007	Region 1-2 interface
4.75 / 2.03	1-2	0.96426	0.00007	Increase intra-rack spacing 2 cm

10.3.2 Region 1-2 Interface with Region 2 Depleted Fuel

Table 10.2 summarizes infinite lattice and interface model results with depleted fuel (5.0 wt%, 40 GWd/MTU) in Region 2. All cases have fuel centered in the rack cells which maximizes k-eff in Regions 1 and 2. The first two cases are infinite lattice cases for comparison. The interface model k-eff is not bounded by the Region 2 infinite lattice k-eff. This result is expected because of the relatively large infinite lattice reactivity difference between Region 1 and 2.

Table 10.2: Region 1-2 Interface (Region 1 Fresh Fuel and Region 2 Depleted Fuel)

Enrichment and Burnup	Region	k-eff	Uncert.	Note
4.75 0G / N/A	1	0.96745	0.00008	Inifinite lattice Reg. 1
N/A / 5.0 40G	2	0.95678	0.00006	Inifinite lattice Reg. 2
4.75 / 5.0 40G	1-2	0.96384	0.00007	Region 1-2 interface

Because the interface result is not bounded by the infinite lattice analysis k-eff of Region 2, there is a possibility that the interface model has less margin to the k-eff limit than indicated in the Region 2 infinite lattice analysis. Table 10.3 documents interface model tolerance cases.

These will be used to compare geometric and fuel content sensitivity of the interface model to analogous infinite lattice Region sensitivities in Table 10.4. The tolerances in Table 10.3 are selected from the most significant tolerances identified in the infinite lattice analysis.

Table 10.3: Region 1-2 Interface Model Selected Tolerances (Fresh/Depleted)

Enrichment and Burnup	Region	k-eff	Uncert.	dk	Note
4.75 / 5.0 40G	1-2	0.96384	0.00007	N/A	Base case
4.75 / 5.0 41G	1-2	0.96366	0.00008	-0.0002	Reg. 2 burnup increase
4.75 / 5.0 40G	1-2	0.96377	0.00007	-0.0001	Reg. 2 narrow BORAL
4.75 / 5.0 40G	1-2	0.96377	0.00007	-0.0001	Reg. 2 growth and creep
4.80 / 5.0 40G	1-2	0.96561	0.00008	0.0018	Reg. 1 enrichment increase +0.05 wt%
4.75 / 5.0 40G	1-2	0.95794	0.00008	-0.0059	Reg. 1 cell ID change
4.75 / 5.0 40G	1-2	0.96563	0.00008	0.0018	Region 1 clad OD change

Table 10.4 compares Region 1 and Region 2 infinite lattice tolerance sensitivity and tolerance sensitivity of the interface model. All tolerance values are best estimate (Δk -eff). The relative sensitivity of the interface model to perturbations in each Region is calculated as (Interface tolerance Δk / Infinite lattice Δk , %). The weighted average sensitivity of the interface model for each Region is calculated using as follows (infinite lattice tolerance is the absolute value):

$$\text{Weighted Importance} = \frac{\sum(\text{infinite lattice tolerance} \times \text{relative sensitivity})_i}{\sum \text{infinite lattice tolerance}_i}$$

It is clear for this interface condition (fresh fuel Region 1, depleted fuel Region 2) that the interface model is dominated by Region 1 and is insensitive to perturbations in Region 2. Therefore, the Region 1 total bias and uncertainty may appropriately be applied to the interface model. Table 10.5 presents the interface model margin calculation.

Table 10.4: Region 1-2 Interface Model Tolerance Comparison (Fresh / Depleted)

Tolerance case	Model	Reg. 1 Enrich. / Burnup**	Reg. 2 Enrich. / Burnup**	Tolerance (k - k base)*	Interface tol. / Inf. lattice tol.
Increase enrich 0.05 Reg1	Interface 1-2	4.80 / 0	5.0 / 40	0.0018	89%
	Region 1	4.80 / 0	N/A	0.0020	
Reduce cell ID Reg1	Interface 1-2	4.75 / 0	5.0 / 40	-0.0059	100%
	Region 1	4.75 / 0	N/A	-0.0059	
Reduce clad OD Reg1	Interface 1-2	4.75 / 0	5.0 / 40	0.0018	97%
	Region 1	4.75 / 0	N/A	0.0019	
Weighted average Region 1 importance					97%
Increase BU 1 GWd/MTU Reg2	Interface 1-2	4.75 / 0	5.0 / 41	-0.0002	4%
	Region 2	N/A	5.0 / 41	-0.0041	
Narrow BORAL Reg 2	Interface 1-2	4.75 / 0	5.0 / 40	-0.0001	-3%
	Region 2	N/A	5.0 / 40	0.0022	
Growth Reg 2	Interface 1-2	4.75 / 0	5.0 / 40	-0.0001	-1%
	Region 2	N/A	5.0 / 40	0.0048	
Weighted average Region 2 importance					0%

*Best estimate dk (does not include statistical uncertainty)

**Enrichment is wt% U-235, Burnup is GWD/MTU

Table 10.5: Region 1-2 Interface Model Margin (Fresh / Depleted)

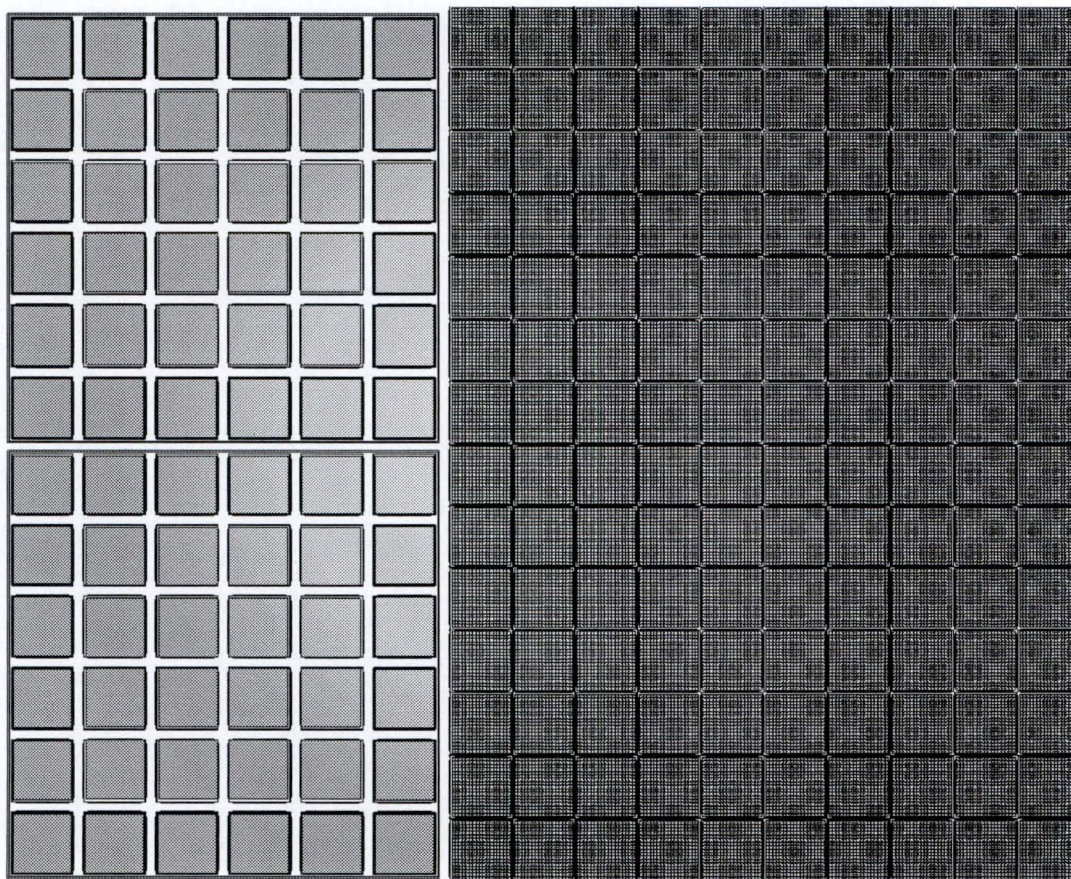
Region 1 total bias and uncertainty	0.0247	(includes 0.01 dk NRC admin margin)
Region 2 total bias and uncertainty	0.0380	(includes 0.01 dk NRC admin margin)
Region 1 importance	100%	
Region 2 importance	0%	
Interface model base k-eff	0.9638	
Interface model bias and uncertainty	0.0247	(weighted total bias and uncertainty)
Interface model 95/95 k-eff	0.9885	(interface k-eff + wtd. total bias + uncert.)
Interface model margin to k-eff limit	0.0115	(greater than infinite lattice Reg 1 and Reg 2)
Adequate margin to k-eff limit?	YES	(bounded by infinite lattice analysis)

10.4 Region 2-3 Interface

Region 2 is most reactive at 32 F. Region 3 is most reactive at 150 F. Interface cases are run at both temperatures. Both Regions have substantial burnup requirements for all but very low enrichment fuel, so fresh fuel and high burnup fuel is considered in each Region.

The Region 2-3 interface models are formed from a 10x14 or 14x14 cell Region 2 and two 6x6 cell Region 3 racks (Figure 10.3). Region 3 racks are 6x6 enclosed by stainless steel on all four sides and are modeled as installed with a small (~1 inch) gap between adjacent modules. The 14 Region 2 cells match the 2 Region 3 racks along the interface within 0.5 inch.

Figure 10.3: X-Y Representation of the Region 2-3 Interface Model



With periodic X-Y boundary conditions, the Region 2-3 model represents infinite columns of Region 3 racks and Region 2 cells in the Y dimension.

In the X dimension at the rack interface, only the intra-rack gap resulting from the water at the edge of the infinite lattice models (a result of modeling normal cell pitch) is credited, resulting in

a 0.565 inch gap. No additional intra-rack spacing is added to credit the 1.28 inch spacing between Region 2 and 3 racks. Reflective boundary conditions are used in the Z direction.

Note that the depleted fuel base case k-eff for the Region 2 infinite lattice model in this section does not match the k-eff in Table 9.19. Region 2 interface model depleted fuel cases are modeled with Zirc2 rather than Zr clad material. To maintain modeling consistency in the comparison of infinite lattice k-eff and interface model k-eff, the Region 2 infinite lattice k-eff with Zirc2 clad modeling is reported. Region 2 sensitivity cases were also run with Zirc-2 cladding, therefore the sensitivity comparison is also consistent.

10.4.1 Region 2-3 Interface (Fresh / Fresh)

Table 10.6 summarizes infinite lattice and interface model results with fresh fuel in both Regions. Region 2 (10x14) has fuel centered in the rack cells to maximize k-eff. Region 3 has a 4x4 cell section of each rack module asymmetrically loaded towards the rack center to maximize k-eff. The interface model k-eff is bounded by the maximum infinite lattice analysis k-eff of Region 1 and Region 2. The final case in Table 10.6 confirms that modestly increasing intra-rack spacing reduces k-eff. No additional Region 2-3 interface analysis is needed for this condition.

Table 10.6: Region 2-3 Interface (Fresh / Fresh)

Enrichment (wt% U235)	Region	k-eff	Uncert.	Note
2.03 / N/A	32F Reg 2	0.96872	0.00006	Infinite lattice Reg. 2
2.03 / N/A	150F Reg 2	0.95676	0.00006	Infinite lattice Reg. 2
N/A / 1.70	32F Reg 3	0.96247	0.00007	Infinite lattice Reg. 3
N/A / 1.70	150F Reg 3	0.96859	0.00007	Infinite lattice Reg. 3
2.03 / 1.70	32F Reg 2-3	0.96420	0.00006	Region 2-3 Interface
2.03 / 1.70	150F Reg 2-3	0.96686	0.00007	Region 2-3 Interface
2.03 / 1.70	150F Reg 2-3	0.96628	0.00007	2cm larger intra-rack Reg 2-3 gap

10.4.2 Region 2-3 Interface (Fresh / Depleted)

Table 10.7 summarizes infinite lattice and interface model results with depleted fuel (5.0 w/o, 54 GWd/MTU) in Region 3. Region 2 (14x14) has fuel centered in the rack cells to maximize k-eff. Region 3 has a 4x4 cell section of each rack module asymmetrically loaded to maximize k-eff. The first two cases are infinite lattice cases for comparison. The interface model k-eff is higher than the Region 3 infinite lattice k-eff. This result is expected because of the relatively large infinite lattice reactivity difference between Region 2 and 3.

Table 10.7: Region 2-3 Interface (Fresh / Depleted)

Enrich. and Burnup*	Region	k-eff	Uncert.	Note
2.03 0G / N/A	32F Reg 2	0.96872	0.00006	Infinite lattice Reg. 2
2.03 0G / N/A	150F Reg 2	0.95676	0.00006	Infinite lattice Reg. 2
N/A / 5.0 54G	32F Reg 3	0.94337	0.00007	Infinite lattice Reg. 3
N/A / 5.0 54G	150F Reg 3	0.95081	0.00006	Infinite lattice Reg. 3
2.03 0G / 5.0 54G	32F Reg 2-3	0.96589	0.00007	Fresh Reg. 2 / Max BU Reg 3
2.03 0G / 5.0 54G	150F Reg 2-3	0.95386	0.00006	Fresh Reg. 2 / Max BU Reg 3

*Enrichment is w/o U-235; burnup is GWd/MTU

The interface model is most reactive at 32 F. Because the interface k-eff (0.96589) is greater than the infinite lattice analysis k-eff of Region 3 (0.95081), there is a possibility that the interface model has less margin to the k-eff limit than indicated in the Region 3 infinite lattice analysis. Table 10.8 documents interface model tolerance cases. These will be used to compare geometric and fuel content sensitivity of the interface model to infinite lattice Region sensitivities in Table 10.9. The tolerances in Table 10.8 are selected from the most significant tolerances identified in the infinite lattice analysis. All cases are run at the limiting interface model temperature (32 °F).

Table 10.8: Region 2-3 Interface Model Selected Tolerances (Fresh / Depleted)

Enrichment and Burnup*	Region	k-eff	Uncert.	dk	Note
2.03 0G / 5.0 54G	2-3	0.96589	0.00006	N/A	Base case
2.03 0G / 5.0 53G	2-3	0.96584	0.00007	-0.0001	Reg 3 reduce burnup 1 GWd/MTU
2.03 0G / 5.0 54G	2-3	0.96588	0.00006	0.0000	Reg 3 creep and grid growth
2.0 0G / 5.0 54G	2-3	0.96153	0.00006	-0.0044	Reg 2 reduce enrichment 0.03 wt%

*Enrichment is w/o U-235; burnup is GWd/MTU

Table 10.9 compares infinite lattice tolerance sensitivity from the infinite lattice analyses (Region 2 and Region 3) and tolerance sensitivity of the interface model. All infinite lattice values are calculated at the bounding temperature for that Region. Interface model tolerances are calculated at the limiting interface model temperature. All tolerance values are best estimate ($\Delta k\text{-eff}$).

Table 10.9: Region 2-3 Interface Model Tolerance Comparison (Fresh / Depleted)

Tolerance case	Model	Reg. 2 Enrich. / Burnup**	Reg. 3 Enrich. / Burnup**	Tolerance (k - k base)*	Interface tol. / Inf. lattice tol.
Reduce enrich 0.03	Interface 2-3	2.0 / 0	5.0 / 54	-0.0044	99%
	Region 2	2.0 / 0	N/A	-0.0044	
Weighted average Region 2 importance					99%
Reduce BU 1 GWd/MTU	Interface 2-3	2.03 / 0	5.0 / 53	-0.0001	-1%
	Region 3	N/A	5.0 / 53	0.0044	
Growth	Interface 2-3	2.03 / 0	5.0 / 54	0.0000	0%
	Region 3	N/A	5.0 / 54	0.0061	
Weighted average Region 3 importance					-1%

*Best estimate dk (does not include statistical uncertainty)

**Enrichment is wt% U-235, Burnup is GWD/MTU

At this interface condition (fresh fuel Region 2, depleted fuel Region 3) the interface model is dominated by Region 2 and is insensitive to perturbations in Region 3. Therefore, it is appropriate to apply the Region 2 infinite lattice analysis total bias and uncertainty to the interface model. Table 10.10 presents the interface model margin calculation.

Table 10.10: Region 2-3 Interface Model Margin (Fresh / Depleted)

Region 2 total bias and uncertainty	0.0233	(includes 0.01 dk NRC admin margin)
Region 3 total bias and uncertainty	0.0426	(includes 0.01 dk NRC admin margin)
Region 2 importance	100%	
Region 3 importance	0%	
Interface model base k-eff	0.9659	
Interface model bias and uncertainty	0.0233	(weighted total bias and uncertainty)
Interface model 95/95 k-eff	0.9892	(interface k-eff + wtd. total bias + uncert.)
Interface model margin to k-eff limit	0.0108	(greater than infinite lattice Reg 2 and Reg 3)
Adequate margin to k-eff limit?	YES	(bounded by infinite lattice analysis)

10.4.3 Region 2-3 Interface (Depleted / Fresh)

Table 10.11 summarizes infinite lattice and interface model results with depleted fuel (5.0 w/o, 40 GWd/MTU) in Region 2. Region 2 (14x14) has fuel centered in the rack cells to maximize k-eff. Region 3 has a 4x4 cell section of each rack module asymmetrically loaded to maximize k-eff. The first four cases are infinite lattice cases for comparison. The interface model k-eff is not bounded by the Region 2 infinite lattice k-eff. This result is expected because of the relatively large infinite lattice reactivity difference between Region 2 and 3.

Table 10.11: Region 2-3 Interface (Depleted / Fresh)

Enrich. and Burnup*	Region	k-eff	Uncert.	Note
5.0 40G / N/A	32F Reg 2	0.95678	0.00006	Infinite lattice Reg. 2
5.0 40G / N/A	150F Reg 2	0.94838	0.00006	Infinite lattice Reg. 2
N/A / 1.70 0G	32F Reg 3	0.96247	0.00007	infinite lattice Reg. 3
N/A / 1.70 0G	150F Reg 3	0.96859	0.00007	infinite lattice Reg. 3
5.0 40G / 1.70 0G	32F Reg 2-3	0.96081	0.00007	Interface Max BU Reg. 2 / Fresh Reg 3
5.0 40G / 1.70 0G	150F Reg 2-3	0.96650	0.00006	Interface Max BU Reg. 2 / Fresh Reg 3

*Enrichment is w/o U-235; burnup is GWd/MTU

The interface model is most reactive at 150 F. Because the interface k-eff (0.96650) is greater than the infinite lattice analysis k-eff of Region 2 (0.95678), there is a possibility that the interface model has less margin to the k-eff limit than indicated in the Region 2 infinite lattice analysis. Table 10.12 documents interface model tolerance cases. They will be used to compare geometric and fuel content sensitivity of the interface model to infinite lattice Region sensitivities in Table 10.13. The tolerances in Table 10.14 are selected from the most significant tolerances identified in the infinite lattice analysis. All cases are run at the limiting interface model temperature (150 F).

Table 10.12: Region 2-3 Interface Model Selected Tolerances (Depleted / Fresh)

Enrichment and Burnup*	Region	k-eff	Uncert.	dk	Note
5.0 40G / 1.70 0G	150F Reg 2-3	0.96650	0.00006	N/A	Base case
5.0 40G / 1.75 0G	150F Reg 2-3	0.97574	0.00006	0.0092	Reg 3 increase enrichment 0.05 wt%
5.0 41G / 1.70 0G	150F Reg 2-3	0.96644	0.00007	-0.0001	Reg 2 Increase burnup 1 GWd/MTU
5.0 40G / 1.70 0G	150F Reg 2-3	0.96661	0.00007	0.0001	Reg 2 Creep and grid growth

*Enrichment is w/o U-235; burnup is GWd/MTU

Table 10.13 compares infinite lattice tolerance sensitivity from the infinite lattice analyses (Region 2 and Region 3) and tolerance sensitivity of the interface model. All infinite lattice values are calculated at the bounding temperature for that Region. Interface model tolerances are calculated at the limiting interface model temperature. All tolerance values are best estimate (Δk_{eff}).

Table 10.13: Region 2-3 Interface Model Tolerance Comparison (Depleted / Fresh)

Tolerance case	Model	Reg. 2 Enrich. / Burnup**	Reg. 3 Enrich. / Burnup**	Tolerance (k - k base)*	Interface tol. / Inf. lattice tol.
Increase BU 1 GWd/MTU	Interface 2-3	5.0 / 41	1.70 / 0	-0.0001	1%
	Region 2	5.0 / 41	N/A	-0.0041	
Grid growth and clad creep	Interface 2-3	5.0 / 40	1.70 / 0	0.0001	2%
	Region 2	5.0 / 40	N/A	0.0048	
Weighted average Region 2 importance					2%
Increase enrich 0.05	Interface 2-3	5.0 / 40	1.75 / 0	0.0092	101%
	Region 3	N/A	1.75 / 0	0.0092	
Weighted average Region 3 importance					101%

*Best estimate Δk (does not include statistical uncertainty)

**Enrichment is wt% U-235, Burnup is GWD/MTU

At this interface condition (depleted fuel Region 2, fresh fuel Region 3) the interface model is dominated by Region 3 and is insensitive to perturbations in Region 2. The Region 2 tolerance values (Δk) are essentially zero (approximately one RSS sigma). Therefore, it is appropriate to apply the Region 3 infinite lattice analysis total bias and uncertainty to the interface model.

Table 10.14 presents the interface model margin calculation.

Table 10.14: Region 2-3 Interface Model Margin (Depleted / Fresh)

Region 2 total bias and uncertainty	0.0380	(includes 0.01 dk NRC admin margin)
Region 3 total bias and uncertainty	0.0259	(includes 0.01 dk NRC admin margin)
Region 2 importance	0%	
Region 3 importance	100%	
Interface model base k-eff	0.9665*	
Interface model bias and uncertainty	0.0259	(weighted total bias and uncertainty)
Interface model 95/95 k-eff	0.9924	(interface k-eff + wtd. total bias + uncert.)
Interface model margin to k-eff limit	0.0076*	(greater than infinite lattice Reg 2 and Reg 3)
Adequate margin to k-eff limit?	YES	(bounded by infinite lattice analysis)

*Region 3 Infinite lattice analysis uses 1.703 wt%, which is 0.0006 more reactive than the 1.70 wt% used for the interface analysis. Interface model margin is 0.007 dk accounting for 0.003 wt% enrichment increase.

10.4.4 Region 2-3 Interface (Depleted / Depleted)

Table 10.15 summarizes infinite lattice and interface model results with depleted fuel in both Regions. Region 2 (14x14) has fuel centered in the rack cells to maximize k-eff. Region 3 has a 4x4 cell section of each rack module asymmetrically loaded to maximize k-eff. The first four cases are infinite lattice cases for comparison. The interface model k-eff is not bounded by the Region 3 infinite lattice k-eff. This result is expected because of the relatively large infinite lattice reactivity difference between Region 2 and 3.

Table 10.15: Region 2-3 Interface (Depleted / Depleted)

Enrich. and Burnup*	Region	k-eff	Uncert.	Note
5.0 40G / N/A	32F Reg 2	0.95678	0.00006	infinite lattice Reg. 2
5.0 40G / N/A	150F Reg 2	0.94838	0.00006	infinite lattice Reg. 2
N/A / 5.0 54G	32F Reg 3	0.94337	0.00007	infinite lattice Reg. 3
N/A / 5.0 54G	150F Reg 3	0.95081	0.00006	infinite lattice Reg. 3
5.0 40G / 5.0 54G	32F Reg 2-3	0.95423	0.00006	Interface Max BU Reg. 2 / Max BU Reg 3
5.0 40G / 5.0 54G	150F Reg 2-3	0.94939	0.00007	Interface Max BU Reg. 2 / Max BU Reg 3

*Enrichment is w/o U-235; burnup is GWd/MTU

The interface model is most reactive at 32 °F. Because the interface k-eff (0.95423) is greater than the infinite lattice analysis k-eff of Region 3 (0.95081), there is a possibility that the interface model has less margin to the k-eff limit than indicated in the Region 3 infinite lattice analysis. Table 10.16 documents interface model tolerance cases. They will be used to compare geometric and fuel content sensitivity of the interface model to infinite lattice Region sensitivities in Table 10.17. All cases are run at the limiting interface model temperature (32 °F).

Table 10.16: Region 2-3 Interface Model Selected Tolerances (Depleted / Depleted)

Enrichment and Burnup*	Region	k-eff	Uncert.	dk	Note
5.0 40G / 5.0 54G	32F Reg 2-3	0.95423	0.00006	N/A	Base case
5.0 41G / 5.0 54G	32F Reg 2-3	0.95023	0.00006	-0.0040	Region 2 increase burnup 1 GWd/MTU
5.0 40G / 5.0 54G	32F Reg 2-3	0.95903	0.00005	0.0048	Region 2 creep and growth
5.0 40G / 5.0 53G	32F Reg 2-3	0.95429	0.00005	0.0001	Region 3 reduce burnup 1 GWd/MTU
5.0 40G / 5.0 54G	32F Reg 2-3	0.95434	0.00006	0.0001	Region 3 creep and growth

*Enrichment is w/o U-235; burnup is GWd/MTU

Table 10.17 compares infinite lattice tolerance sensitivity from the infinite lattice analyses (Region 2 and Region 3) and tolerance sensitivity of the interface model. All infinite lattice values are calculated at the bounding temperature for that Region. Interface model tolerances are calculated at the limiting interface model temperature. All tolerance values are best estimate (Δk -eff).

Table 10.17: Region 2-3 Interface Model Tolerance Comparison (Depleted / Depleted)

Tolerance case	Model	Reg. 1 Enrich. / Burnup**	Reg. 2 Enrich. / Burnup**	Tolerance (k - k base)*	Interface tol. / Inf. lattice tol.
Increase BU 1 GWd/MTU	Interface 2-3	5.0 / 41	5.0 / 54	-0.0040	98%
	Region 2	5.0 / 41	N/A	-0.0041	
Creep and grid growth	Interface 2-3	5.0 / 40	5.0 / 54	0.0048	100%
	Region 2	5.0 / 40	N/A	0.0048	
Weighted avg. Region 2 importance					99%
Reduce BU 1 GWd/MTU	Interface 2-3	5.0 / 40	5.0 / 53	0.0001	1%
	Region 3	N/A	5.0 / 53	0.0044	
Creep and grid growth	Interface 2-3	5.0 / 40	5.0 / 54	0.0001	2%
	Region 3	N/A	5.0 / 54	0.0061	
Weighted avg. Region 3 importance					2%

*Best estimate dk (does not include statistical uncertainty)

**Enrichment is wt% U-235, Burnup is GWD/MTU

At this interface condition (depleted fuel Region 2, depleted fuel Region 3) the interface model is dominated by Region 2 and is less sensitive to perturbations in Region 3. A blended total bias

and uncertainty may be applied to the interface model composed of 99% Region 2 and 2% Region 3.

Although this interface condition is clearly dominated by Region 2, it is used here as an example of a mixed importance interface calculation. For cases with mixed importance (significant importance in both Regions), a more detailed total bias and uncertainty calculation is appropriate for the following reasons:

- 1) The NRC administrative margin (0.01 dk) should be removed from the total prior to the weighting and added back to the final result. If not, for a case with 101% total importance, the NRC administrative margin will have inadvertently increased from 0.01 dk to 0.0101 dk.
- 2) The code bias and code uncertainty should be removed from the total prior to the weighting and incorporated in the final result (bias added back and uncertainty included by RSS).
- 3) Weighted uncertainty components for each Region could be statistically combined to obtain the total rather than being added together because they are independent (different fuel in different racks, different rack manufacturers, etc).

For simplicity, only the first item will be included in the interface margin calculation. Table 10.18 presents the interface model margin calculation.

Table 10.18: Region 2-3 Interface Model Margin (Depleted / Depleted)

Region 2 total bias and uncertainty	0.0380	(includes 0.01 dk NRC admin margin)
Region 3 total bias and uncertainty	0.0426	(includes 0.01 dk NRC admin margin)
Region 2 importance	99%	
Region 3 importance	2%	
Interface model base k-eff	0.9542	
Interface model bias and uncertainty	0.0383	(weighted total bias and uncertainty)*
Interface model 95/95 k-eff	0.9925	(interface k-eff + wtd. total bias + uncert.)
Interface model margin to k-eff limit	0.0075**	(greater than infinite lattice Reg 2 and Reg 3)
Adequate margin to k-eff limit?	YES	(bounded by infinite lattice analysis)

*NRC administrative margin removed prior to weighting and added back to final result.

**Region 3 infinite lattice margin is calculated with a burnup 0.35 GWd/MTU lower than the interface model (0.0015 dk effect). The interface model Region 3 burnup tolerance case shows that the interface model is insensitive to small Region 3 burnup changes.

10.5 *Interface Analysis Summary*

There are two MPS3 SFP Region interfaces:

- 1) Region 2 meets Region 1
- 2) Region 2 meets Region 3

The Region 2 interfaces with Regions 1 and 3 are either benign or indeterminate. For each indeterminate interface, interface model margin calculations demonstrate that the interface model has adequate margin to the k-eff limit. In addition, the interface analysis margin to the limit is greater than the comparable infinite lattice Region margin.

11 Normal Conditions

Normal operation pool water temperature ranges up to 150 °F. Temperatures ranging from 32 °F to 150 °F are considered in Section 9 Region analyses. Normal fuel storage may include spent burnable absorbers, source rods, or control rods. Inserting these in the assembly decreases k-eff. Tolerance calculations for each Region (fuel pin pitch, guide tube OD, grid size) confirm that displacing water in the fuel lattice reduces k-eff with no soluble boron present, so the normal operation of storing and moving these inserts is acceptable.

11.1 Fuel Handling

MPS3 fuel handling procedures for fresh and spent fuel place the following limits (among others) on fuel movement:

- 1) Only one fuel assembly at a time is allowed in the transfer canal
- 2) At least 12 inches of separation is maintained between any two fuel assemblies outside of the racks
- 3) Fuel may be lowered into an approved storage location
- 4) Fuel being lowered must remain at least 12 inches away from the nearest fuel assembly if being lowered outside an approved storage location
- 5) Fuel may be lowered into a Dry Shielded Canister in the Cask Pit

Two fresh 5.0 wt% fuel assemblies 30.48 cm apart (12 inches) produce a k-eff less than 0.94 (Figure 11.1), which is less than the base case k-eff for Regions 1-3 (Section 9). Closer proximity of fuel being normally handled outside the storage racks is precluded by handling procedures. Figure 11.2 confirms that the fuel elevator, transfer track, and upender are not in close proximity to fuel racks. Normal fuel handling procedures preclude lowering fuel adjacent to a fuel rack (not an approved storage location), so proximity of a fuel assembly outside the SFP racks within 12 inches of fuel in SFP racks does not need to be considered a normal configuration.

Figure 11.1: k-eff of Two Fresh 5 wt% Fuel Assemblies in Unborated Water

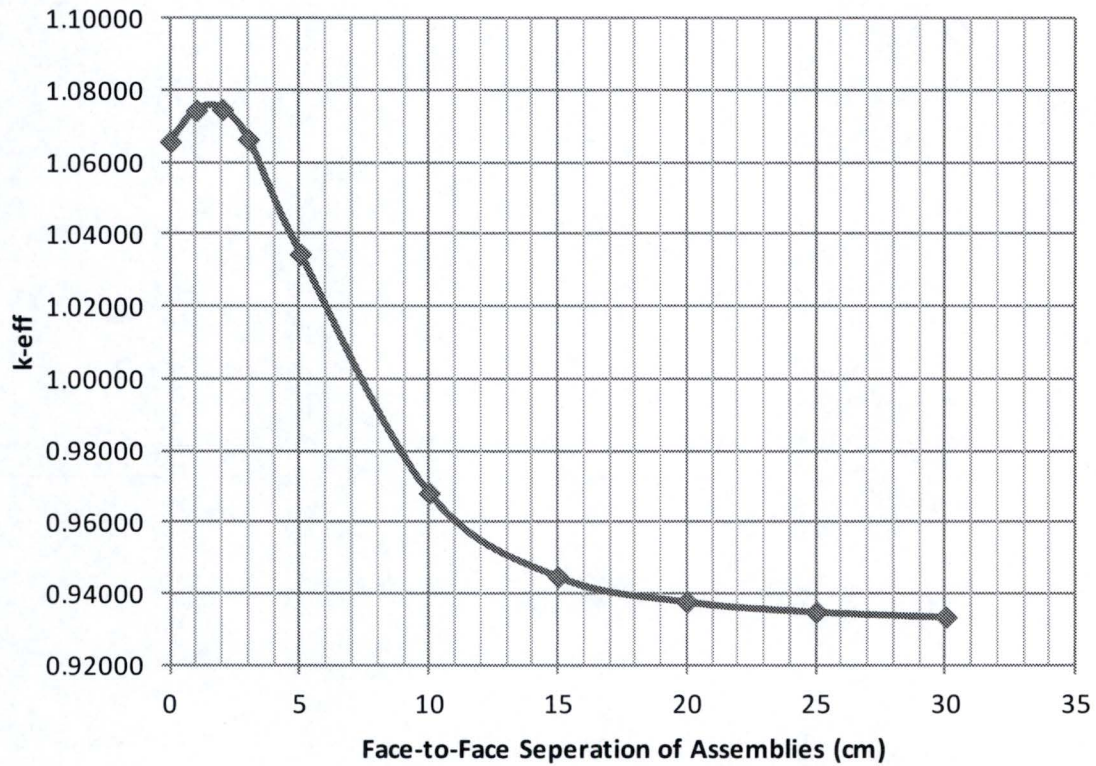
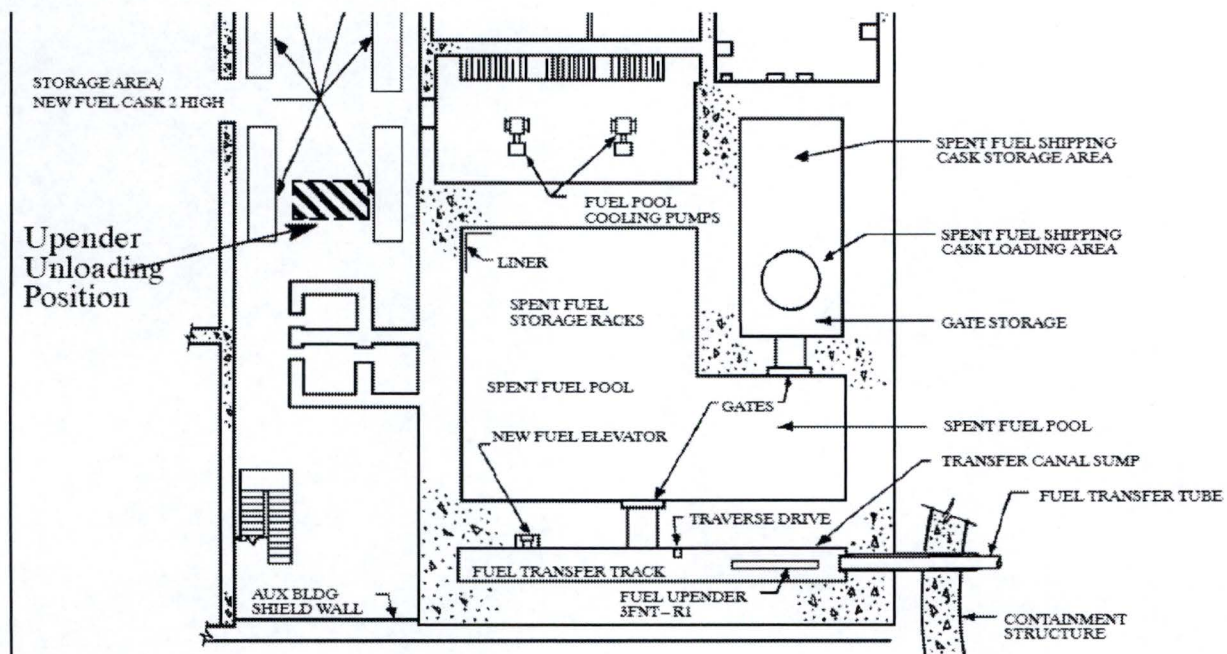


Figure 11.2: Location of MPS3 Fuel Handling Equipment



11.2 Fuel Inspection

Per fuel handling procedure, fuel inspections are performed on only one fuel assembly at a time and is governed by normal fuel handling restrictions. No additional analysis is needed.

11.3 Non-standard Fuel

11.3.1 Fuel Rod Storage Canister

A fuel rod storage canister (FRSC) in the MPS3 SFP allows for storage of up to 52 fuel rods.

The key design features of the FRSC are:

- Square lattice
- 0.937 inch storage tube pitch
- 52 fuel rod storage tubes
- Storage tube OD 0.625 inch (48 tubes) and 0.750 inch (4 tubes)
- Storage tube wall thickness 0.035 inch (48 tubes) and 0.049 inch (4 tubes)

The FRSC is analyzed assuming it contains 52 fresh 5.0 w/o fuel rods. Figure 11.3 depicts the FRSC in a Region 2 KENO model. Figure 11.4 shows the Region 3 FRSC model. Table 11.1 documents Region 2 and Region 3 infinite lattice KENO model results. The FRSC was not modeled in Region 1 because Regions 2 and 3 are far more reactive rack designs. Uncertainty and bias items are not modeled due to the large amount of margin to the k-eff limit. Cases include centered and eccentric placement of the FRSC in the storage cells.

The cross section treatment for the FRSC assumes that it is an infinite array of fuel pins using the FRSC fuel pin pitch. Several attempts of improved modeling were made but none of the various modeling attempts increased k-eff above 0.7 with 0 ppm soluble boron. Special modeling is not needed.

The FRSC may be stored in any storage location in the MP3 SFP in which a fuel assembly is permitted.

Figure 11.3: Region 2 Fuel Rod Storage Canister Model

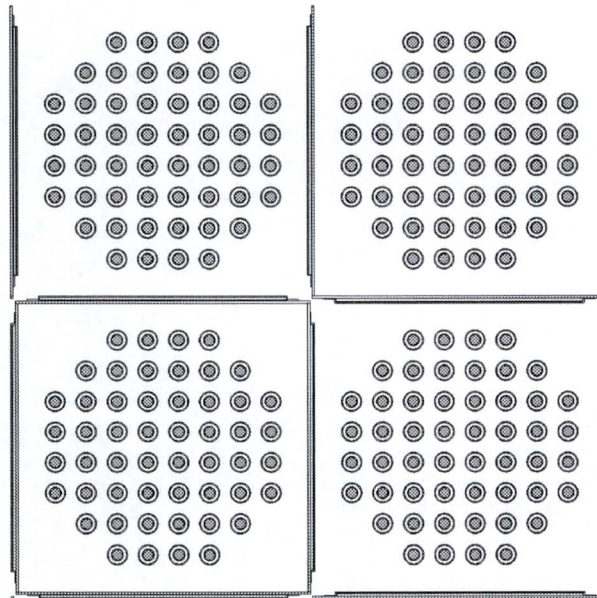


Figure 11.4: Region 3 Fuel Rod Storage Canister Model

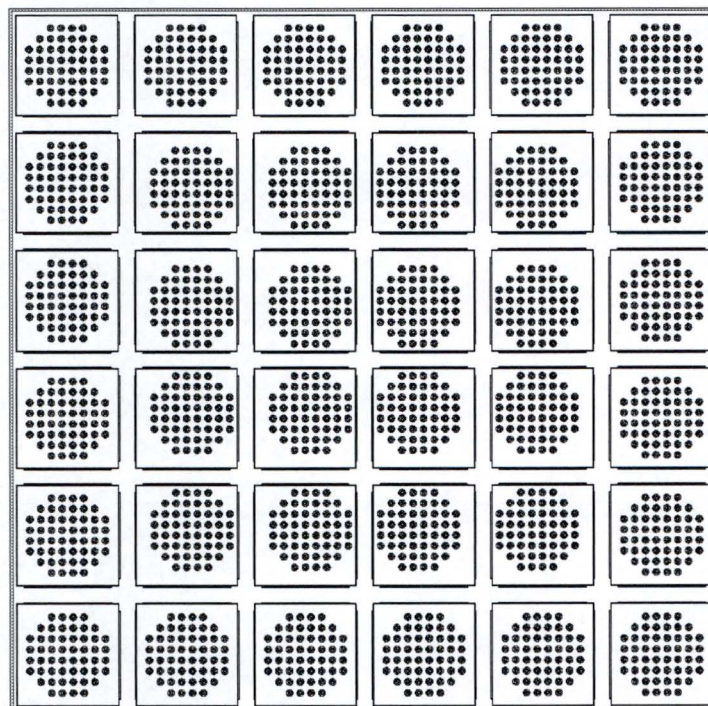


Table 11.1: Region 2 and 3 Fuel Rod Storage Canister Model Results

Region	Temperature (F)	K-eff	Uncert.	EALF
2	32	0.68699	0.00007	0.087
3	150	0.68923	0.00006	0.088

11.3.2 Reconstituted Fuel

Fuel pins in assemblies are sometimes removed and replaced to create reconstituted assemblies. There are two reconstitution scenarios evaluated:

- Remove a fuel pin(s) and replace it with a stainless steel pin(s).
- Remove a fuel pin(s) and do not replace the missing fuel pin(s) with anything. This scenario leaves an open water hole in the fuel lattice.

Because the fuel assembly pin lattice is usually under-moderated, an open water hole in a fuel assembly can increase the reactivity of the assembly. Therefore if an assembly is being reconstituted, storage requirements are established to accommodate fuel pin removal. Figures 11.5 (Region 2) and 11.6 (Region 3) represent modified versions of the infinite lattice Region 2 and 3 KENO models in which the assembly to be reconstituted has empty cells on all four faces. Region 1 is not modeled because Regions 2 and 3 are far more reactive rack designs.

Table 11.2 documents the results of three fuel reconstitution cases for each Region with no soluble boron. The first case has low enrichment fresh fuel in centered in each storage location (2.0 w/o in Region 2, 1.7 w/o in Region 3). The second case has a fresh 5.0 w/o fuel assembly with four empty face neighbors. The third case replaces the 5.0 w/o assembly with a low enrichment assembly. These cases show that the model k-eff is insensitive to the enrichment of the reconstitution assembly (up to 5.0 wt%) and that the small amount of enrichment sensitivity exhibited is two orders of magnitude less than the k-eff reduction vs the base case. These results indicate that with four face-adjacent empty cells, the reconstitution assembly is effectively neutronically isolated and does not need further analysis to support the reconstitution process in Region 1, 2, or 3 in the MP3 SFP.

Replacement of a fuel rod with a stainless steel rod reduces k-eff because there is no change in the amount of moderator present and there is a replacement of fissile material with a weak neutron absorber. Region 2 and Region 3 stainless steel replacement rod cases in Table 11.3

confirm this reduction in fuel reactivity regardless of fuel pin position. Cases in Table 11.3 have no soluble boron.

MP3 fuel assembly MR71, shown in Figure 11.7, has two peripheral fuel pin locations that are empty (locations H01 and J01). Region 2 and Region 3 cases in Table 11.3 confirm that removal of fuel pins H01 and J01 reduces k_{eff} . MR71 may be stored as a normal fuel assembly.

Figure 11.5: Region 2 Reconstitution Model

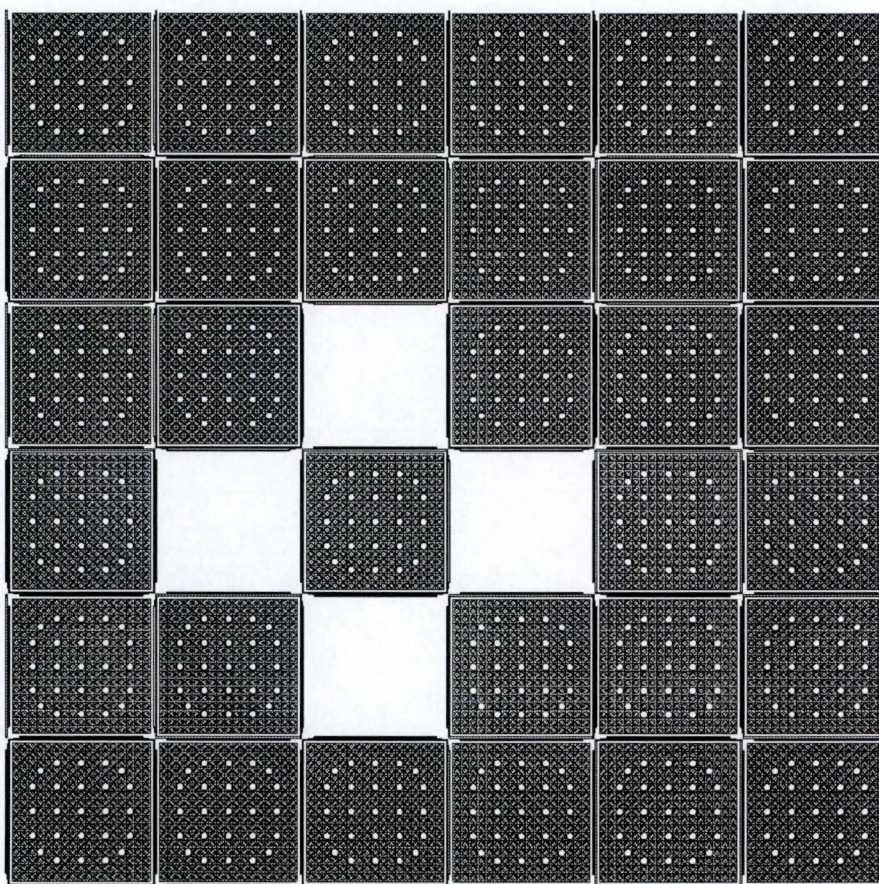
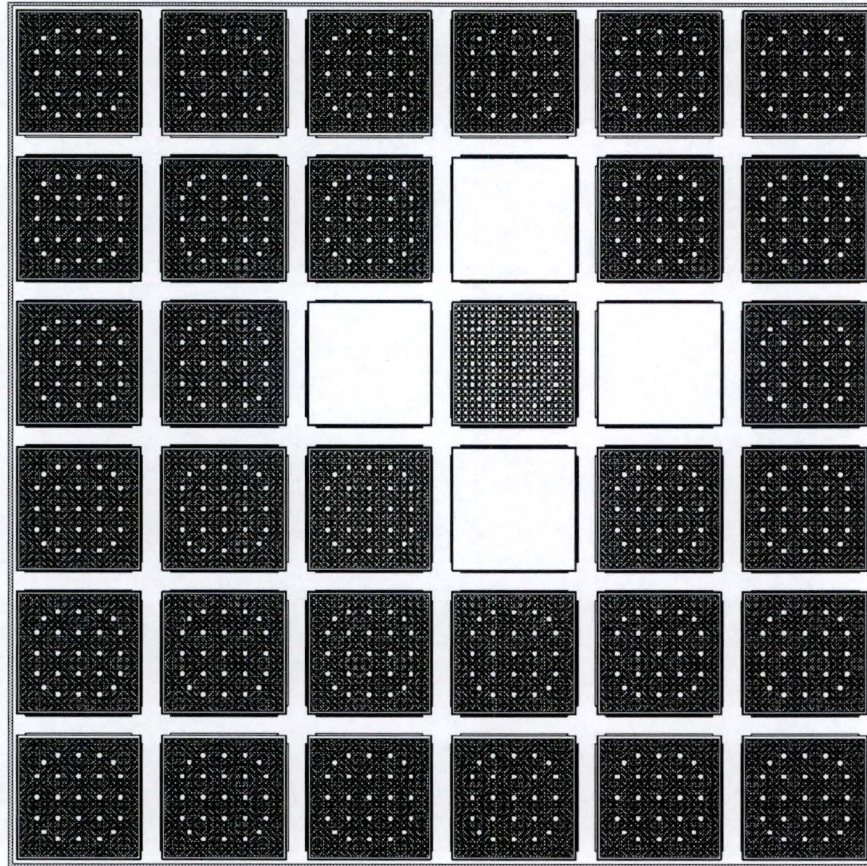


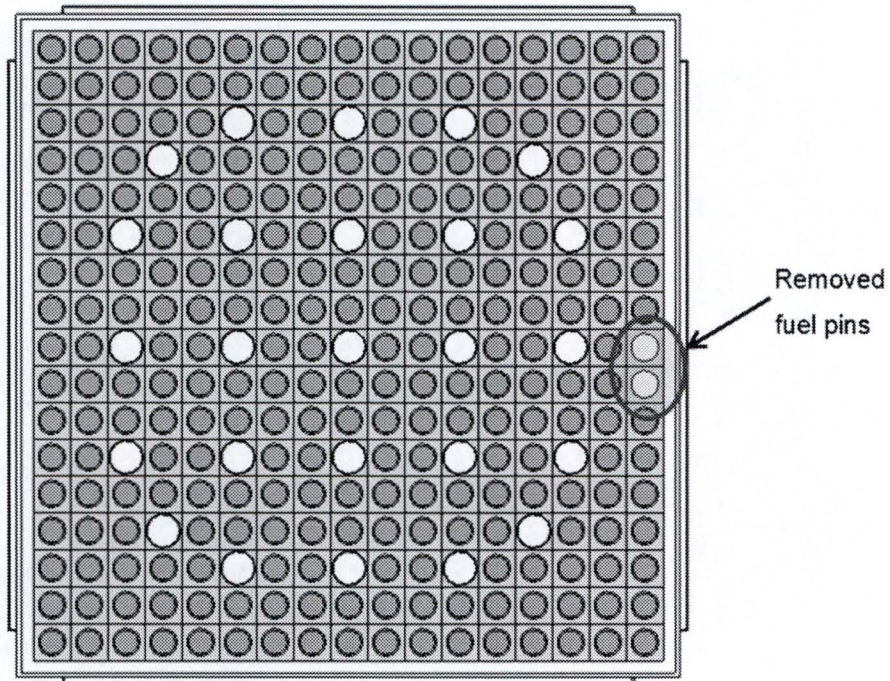
Figure 11.6: Region 3 Reconstitution Model**Table 11.2: Fuel Reconstitution Assembly Isolation**

Region	Temp. (F)	k-eff	Uncert.	dk	Note
2	32	0.96422	0.00006	Base	Base case all 2.0 w/o
2	32	0.94661	0.00007	-0.0176	Base case with 1 5.0 w/o fresh and face neighbors empty
2	32	0.94648	0.00006	-0.0177	Base case with 1 2.0 w/o fresh and face neighbors empty
3	150	0.96154	0.00006	Base	Base case all 1.7 w/o
3	150	0.93573	0.00006	-0.0258	Base case with 1 5.0 w/o fresh and face neighbors empty
3	150	0.93559	0.00006	-0.0260	Base case with 1 1.7 w/o fresh and face neighbors empty

Table 11.3: Fuel Reconstitution Fuel Rod Replacement

Region	Enrich. (w/o)	Temp. (F)	k-eff	Uncert.	EALF	Note
2	2.0	32	0.96422	0.00006	0.189	Base case all 2.0 w/o
2	2.0	32	0.96238	0.00006	0.189	Replace 1 perimeter rod with SS rod
2	2.0	32	0.96001	0.00007	0.189	Replace 1 center rod with SS rod
2	2.0	32	0.96074	0.00006	0.189	Replace 1 interior rod with SS rod
2	2.0	32	0.96213	0.00006	0.189	Replace 1 corner rod with SS rod
2	2.0	32	0.96194	0.00006	0.187	Replace 2 rods on 1 face with water
2	5.0	32	1.20103	0.00007	0.312	Base case all 5.0 w/o
2	5.0	32	1.19923	0.00007	0.311	Replace 1 perimeter rod with SS rod
3	1.7	150	0.96154	0.00006	0.134	Base case all 1.7 w/o
3	1.7	150	0.95752	0.00006	0.134	Replace 1 perimeter rod with SS rod
3	1.7	150	0.95798	0.00006	0.134	Replace 1 center rod with SS rod
3	1.7	150	0.95816	0.00006	0.134	Replace 2 rods on 1 face with water

Figure 11.7: Region 3 Fuel Assembly MR71 Model



11.4 Normal Condition Boron Credit

The Code of Federal Regulations Title 10 Part 50 Section 68 (b).4 states:

"If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. "

The MPS3 SFP criticality analysis credits soluble boron. Rather than searching for the soluble boron required for $k < 0.95$ for all normal configurations, KENO model calculations are performed with 600 ppm soluble boron, which is shown to be more than sufficient to meet the criticality requirement. The normal storage soluble boron calculation scope includes:

- Region 1 6x6 infinite lattice model (4 out of 4 storage)
 - 4.75 w/o, 0 GWd/MTU
 - Important bias and uncertainty items calculated
 - 5.0 w/o, 2 GWd/MTU
- Region 2 2x2 infinite lattice model (4 out of 4 storage)
 - 2.03 w/o, 0 GWd/MTU
 - Bias and uncertainty calculated
 - 5.0 w/o, 40 GWd/MTU
 - Important bias and uncertainty items calculated
 - 5.0 w/o with a control rod
- Region 3 6x6 infinite lattice model (4 out of 4 storage, 4x4 asymmetric placement)
 - 1.7 w/o, 0 GWd/MTU
 - 5.0 w/o 53 GWd/MTU

The Region 1 IFBA credit scenario is not evaluated because it is the least reactive of the Region 1 storage options and because Region 1 is not limiting with 600 ppm soluble boron. No borated Region interface cases are required because the interface configuration reactivity is bounded by infinite lattice model reactivity. No borated non-standard fuel cases are required because non-standard fuel bearing components in the MP3 SFP are less reactive than normal fuel assemblies. No specific analysis is needed for normal fuel handling because fuel handling k-eff is lower than infinite lattice Region k-eff.

Infinite lattice KENO models from Section 9 analyses are modified to include 600 ppm soluble boron. The bounding Region is determined by adding the appropriate total bias and uncertainty

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 216 of 300

for the un-borated condition to the 600 ppm k-eff. To conservatively account for fuel assembly structure that displaces borated water, a portion of the top and bottom reflectors are conservatively modeled as unborated water.

Table 11.4 contains normal condition base case results for Regions 1, 2, and 3 with 600 ppm soluble boron.

For Region 1, the most limiting normal condition is fresh 4.75 w/o fuel at 32 °F. Table 11.5 shows the calculation of total bias and uncertainty for the Region 1 600 ppm condition along with no soluble boron values for comparison. Shaded values (low significance tolerances) are retained from the no soluble boron calculation. Total bias and uncertainty with 600 ppm boron is slightly lower than with no boron. The margin calculation shows that 600 ppm soluble boron is more than enough to meet the requirement of $k < 0.95$.

A full set of 600 ppm tolerance cases (bias and uncertainty) are run for Region 2 for the calculation of Dominion administrative margin for the normal storage condition. Table 11.6 shows the calculation of fresh fuel total bias and uncertainty for the Region 2 600 ppm condition along with no boron values for comparison. Total bias and uncertainty with 600 ppm boron is slightly higher than with no boron. The margin calculation shows that 600 ppm soluble boron is more than enough to meet the requirement of $k < 0.95$. A water displacement tolerance case (Table 11.4) confirms that displacement of water by inert (Zr) rods decreases fuel reactivity. This case allows for storage of depleted BPRA or similar water displacing materials in the fuel assembly guide tubes.

For Region 2, the most limiting normal condition is 5.0 w/o fuel with 40 GWd/MTU burnup at 32 °F. Table 11.7 shows the calculation of depleted fuel total bias and uncertainty for the Region 2 600 ppm condition along with no boron values for comparison. Burnup measurement tolerance is approximated by multiplying the 0 ppm tolerance by the relative total burnup worth (600 ppm burnup worth divided by the 0 ppm burnup worth). This approximation is used because the measurement uncertainty was calculated for the no boron condition by interpolating between burnup steps, but for the 600 ppm condition only one burnup step is available. Code uncertainty and bias values are based on MOX results using a maximum EALF of 0.5 eV. The MOX values result in less margin than 0.5 eV fresh fuel values. Shaded values (low significance tolerances) are retained from the no soluble boron calculation. Total bias and uncertainty with 600 ppm

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 217 of 300

boron is slightly lower than with no boron. The margin calculation shows that 600 ppm soluble boron is more than enough to meet the requirement of $k < 0.95$.

The highest Region 3 600 ppm base case k-eff is 0.844. Region 1 and 2 bias and uncertainty results show that total bias and uncertainty with 600 ppm boron is similar to total bias and uncertainty with 0 ppm. Maximum no boron Region 3 total bias and uncertainty (5 day decay time) is 0.043 including 0.01 NRC administrative margin. The estimated Region 3 95/95 k-eff with 600 ppm soluble boron is 0.887, which is clearly non-limiting.

Table 11.4: Region 1-3 Normal Condition Boron Requirement Cases (600 ppm)

Region	Enrich. (w/o)	Burnup (GWd/MTU)	Temp. (° F)	k-eff	Uncert.	EALF	Tolerance (dk)*	Notes
1	4.75	0	32	0.90300	0.00007	0.372	N/A	Base case
1	5.0	2	32	0.89452	0.00007	0.412	N/A	Base case
1	4.75	0	150	0.89680	0.00007	0.411	-0.0060	Increase temp
2	2.03	0	32	0.86858	0.00006	0.236	N/A	Base case
2	2.03	0	32	0.85816	0.00006	0.258	-0.0103	Zr rods in guide tubes
2	5.0	0	32	0.87641	0.00007	0.569	N/A	Control rod credit base
2	5.0	40	32	0.90059	0.00006	0.478	N/A	Burnup credit base case
3	1.7	0	150	0.80516	0.00005	0.180	N/A	Base case
3	5.0	53	150	0.84428	0.00007	0.334	N/A	Base case

*Includes 2xRSS uncertainty for comparison with 0 ppm tolerance

Table 11.5: Region 1 Bias, Uncertainty, and Margin Comparison

Soluble Boron (ppm)	0	600
Burnup (GWd/MTU)	0	0
Enrichment (wt%)	4.75	4.75
Uncertainties (dk)		
Rack wall thickness	0.0023	0.0021
Rack cell pitch	0.0054	0.0050
Rack cell ID	0.0061	0.0054
Fuel density	0.0007	0.0007
Enrichment	0.0022	0.0022
Wrapper thickness	0.0005	0.0005
Wrapper width	0.0004	0.0004
Fuel OD	0.0004	0.0004
Clad OD	0.0021	0.0015
Clad ID	0.0002	0.0002
Fuel pin pitch	0.0009	0.0009
GT OD	0.0005	0.0005
GT ID	0.0006	0.0006
Poison width	0.0013	0.0013
Poison thickness	0.0010	0.0010
Fuel stack length	0.0002	0.0002
2 x KENO std. dev.	0.0002	0.0002
Code uncertainty	0.0048	0.0048
RSS OF UNCERTAINTIES	0.0104	0.0097
Biases (dk)		
Code bias	0.0034	0.0034
Blisters	0.0009	0.0009
SUM OF BIASES	0.0043	0.0043
Summary		
Base Case k-eff	0.9675	0.9030
Total Bias and Uncertainty	0.0147	0.0139
NRC Administrative Margin	0.0100	0.0100
Maximum k-eff	0.9921	0.9269
10CFR50.68 Limit	1.0000	0.9500
Dominion Margin (dk)	0.0079	0.0231

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 219 of 300

Table 11.6: Region 2 Bias, Uncertainty, and Margin Comparison (fresh fuel)

Soluble Boron (ppm)	0	600
Burnup (GWd/MTU)	0	0
Enrichment (wt%)	2.03	2.03
Uncertainties (dk)		
Rack wall thickness	0.0005	0.0005
Rack cell pitch	0.0006	0.0011
Rack cell ID	0.0003	0.0002
Fuel density	0.0009	0.0011
Enrichment	0.0076	0.0079
Wrapper thickness	0.0005	0.0002
Wrapper width	0.0003	0.0002
Fuel OD	0.0006	0.0007
Clad OD	0.0014	0.0008
Clad ID	0.0003	0.0004
Fuel pin pitch	0.0007	0.0005
GT OD	0.0006	0.0003
GT ID	0.0005	0.0003
Poison width	0.0025	0.0018
Poison thickness	0.0003	0.0003
Fuel stack length	0.0003	0.0002
2 x KENO std. dev.	0.0001	0.0001
Code uncertainty	0.0048	0.0048
RSS OF UNCERTAINTIES	0.0096	0.0097
Biases (dk)		
Code bias	0.0034	0.0034
Blisters	0.0004	0.0004
SUM OF BIASES	0.0038	0.0038
Summary		
Base Case k-eff	0.9687	0.8686
Total Bias and Uncertainty	0.0133	0.0134
NRC Administrative Margin	0.0100	0.0100
Maximum k-eff	0.9920	0.8920
10CFR50.68 Limit	1.0000	0.9500
Dominion Margin (dk)	0.0080	0.0580

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 220 of 300

Table 11.7: Region 2 Bias, Uncertainty, and Margin Comparison (depleted fuel)

Soluble Boron (ppm)	0	600
Burnup (GWd/MTU)	40.25	40
Enrichment (wt%)	5.0	5.0
Worths (dk)		
Minor Actinides and FP	0.108	0.095
Burnup worth	0.244	0.226
Uncertainties (dk)		
Rack wall thickness	0.0004	0.0005
Rack cell pitch	0.0015	0.0015
Rack cell ID	0.0002	0.0002
Fuel density	0.0009	0.0011
Enrichment	0.0000	0.0000
Wrapper thickness	0.0004	0.0004
Wrapper width	0.0003	0.0003
Burnup worth (5%)	0.0122	0.0113
Measured Burnup (4.0%)	0.0099	0.0091
Fuel OD	0.0002	0.0007
Clad OD	0.0000	0.0000
Clad ID	0.0003	0.0004
Fuel pin pitch	0.0012	0.0012
GT OD	0.0007	0.0007
GT ID	0.0004	0.0004
Poison width	0.0024	0.0024
Poison thickness	0.0003	0.0003
Fuel stack length	0.0003	0.0003
2 x KENO std. dev.	0.0001	0.0001
Code uncertainty	0.0094	0.0099
RSS OF UNCERTAINTIES	0.0186	0.0179
Biases (dk)		
Code bias	0.0019	0.0021
Blisters	0.0004	0.0004
Growth, creep	0.0050	0.0038
Tilt	0.0005	0.0005
1.5% minor actinides and FP	0.0016	0.0014
SUM OF BIASES	0.0093	0.0082
Summary		
Base Case k-eff	0.9568	0.9006
Total Bias and Uncertainty	0.0280	0.0261
NRC Administrative Margin	0.0100	0.0100
Maximum k-eff	0.9948	0.9367
10CFR50.68 Limit	1.0000	0.9500
Dominion Margin (dk)	0.0052	0.0133

12 Accident Analysis

Accident scenarios include boron dilution to a minimum boron with normal storage, single fuel assembly mis-load in the fuel racks, multiple fuel assembly mis-load, loss of cooling (including partial voiding), dropped assembly into the racks with grid damage (optimum fuel pin pitch), fuel handling error (two fresh fuel assemblies out of rack in close proximity), and mis-placement of an assembly between fuel racks. An evaluation of boron dilution time and mitigation is included in Attachment 7.

The multiple mis-load accident is the bounding accident scenario. Multiple mis-load cases for non-limiting Regions 1 and 3 use infinite lattice KENO models with fresh 5.0 w/o U-235 fuel in all storage locations. The Region 2 multiple mis-load uses a larger KENO model to simulate the most reactive batch of fuel expected to be present in the MPS3 SFP. A full set of tolerance and bias cases are calculated for the multiple mis-load condition with 2550 ppm soluble boron.

12.1 *Accident Condition Soluble Boron Requirement*

12.1.1 Loss of Cooling

A postulated loss of cooling event is simulated for each Region at 212 °F with 2550 ppm soluble boron and with optimum voiding. Infinite lattice Region models are used.

Region 1 loss of cooling models have un-poisoned 5.0 w/o U-235 fresh fuel. This model is conservative because 5.0 w/o fresh fuel requires 12 IFBA rods to be stored in Region 1's infinite lattice model. Region 2 and 3 cases are run with high enrichment depleted fuel. As shown in the normal storage 600 ppm cases, boron worth is lower for depleted high enrichment fuel than for fresh low enrichment fuel. Isotopic content in the Region 2 and 3 infinite lattice analysis credits all TRITON isotopes except for partial removal of volatile fission products. However, because the loss of cooling is not the limiting accident condition, the fuel is conservatively modeled for convenience using only major actinides.

Results of the loss of cooling cases are shown in Table 12.1. The highest k-eff for the loss of cooling event is 0.885 (Region 3) without credit for minor actinides and fission products (~0.128 dk at 5.0 wt%, 54 GWD/MTU). Compared to the multiple misload accident, the loss of cooling event is non-limiting.

Table 12.1: Loss of Cooling Event

Region	Enrich. (wt%)	Burnup (GWd/MTU)	Boron	Temp. (F)	K-eff	Uncert.	Notes
1	5.0	0	2550	150	0.76221	0.00006	Max temp nominal
1	5.0	0	2550	212	0.76067	0.00006	0% void
1	5.0	0	2550	212	0.75624	0.00006	10% void
2	5.0	40*	2550	150	0.83291	0.00006	Max temp nominal
2	5.0	40*	2550	212	0.83350	0.00006	0% void
2	5.0	40*	2550	212	0.83295	0.00006	10% void
3	5.0	53*	2550	150	0.72689	0.00006	Max temp nominal
3	5.0	53*	2550	212	0.73413	0.00006	0% void
3	5.0	53*	2550	212	0.74945	0.00006	10% void
3	5.0	53*	2550	212	0.76735	0.00006	20% void
3	5.0	53*	2550	212	0.83934	0.00006	50% void
3	5.0	53*	2550	212	0.86568	0.00006	60% void
3	5.0	53*	2550	212	0.88504	0.00006	70% void
3	5.0	53*	2550	212	0.87920	0.00005	80% void

*Major actinides

12.1.2 Single Mis-placement

A single assembly mis-placement between fuel racks could produce a higher k-eff than an in-rack mis-load. The between rack mis-load will be considered using modified Region interface models. Rather than including soluble boron in those models, the un-borated between-rack mis-load k-eff will be compared to the un-borated in-rack mis-load k-eff. If the between rack k-eff is less than the in-rack k-eff, the between-rack mis-load event is non-limiting. Because only the relative magnitude of the mis-load effect is of interest, fresh fuel of the enrichment analyzed in Section 9 will be used to represent the normally loaded fuel in the racks.

Figure 12.1 shows a portion of the modified KENO Region 1-2 interface model with a mis-loaded assembly where the racks meet. The full model includes a 10x10 section of Region 2 rack cells and an 18x9 section of Region 1 rack cells. Periodic X-Y boundary conditions are used.

Figure 12.2 shows an X-Y view of the modified KENO Region 2-3 interface model with a mis-loaded assembly where the racks meet. The actual water space between racks is much larger than modeled (see Figure 4.3 for comparison). Periodic X-Y boundary conditions are used.

Results in Table 12.2 show that mis-placement of a single fresh 5.0 w/o fuel assembly between fuel racks increases k_{eff} less than a mis-load in the fuel rack. Single assembly mis-load in a fuel rack is bounded by multiple mis-load in a fuel rack, therefore the single mis-placement accident is not limiting.

Figure 12.1: Region 1-2 Interface Model for Between-rack Mis-placement

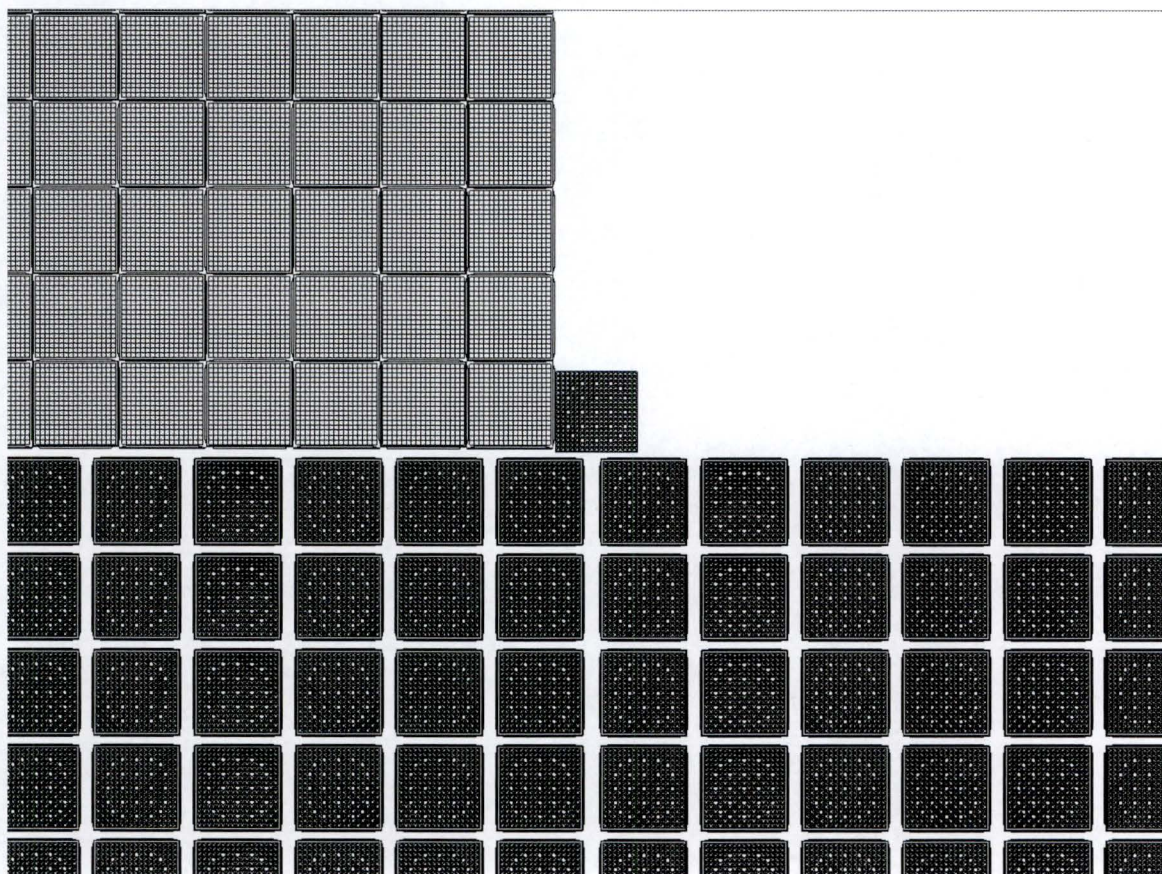
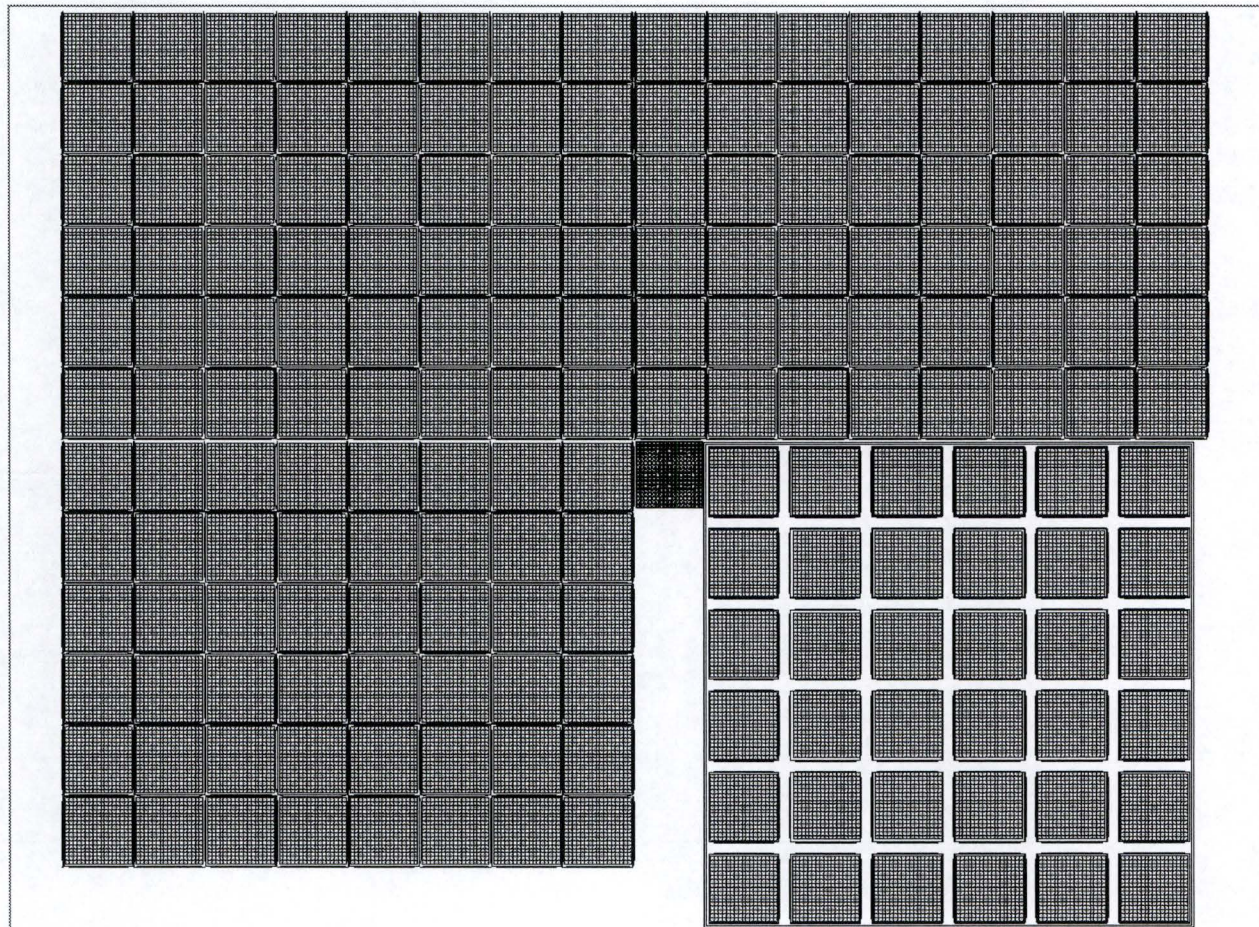


Figure 12.2: Region 2-3 Interface Model for Between-rack Mis-placement



Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 225 of 300

Table 12.2: Single Fuel Assembly Mis-placement

Region	Enrich. (wt%)*	Burnup* (GWd/MTU)	Boron (ppm)	Temp. (°F)	K-eff	Uncert. (dk)	EALF	Notes
1/2	4.75/2.03	0/0	0	32	0.96696	0.00010	0.274	Modified interface model with one fresh 5.0 w/o assembly between racks
1/2	4.75/2.03	0/0	0	32	0.98425	0.00009	0.203	Modified interface model with one fresh 5.0 w/o in Region 2 rack
2/3	2.03/1.70	0/0	0	150	0.97878	0.00010	0.193	Modified interface model with one fresh 5.0 w/o assembly between racks
2/3	2.03/1.70	0/0	0	150	1.04389	0.00010	0.151	Modified interface model with one fresh 5.0 w/o in Region 3 rack

*Case has 2 Regions of fuel

12.1.3 Multiple Mis-load

Analysis of the multiple mis-load is performed with KENO Region 1, 2, and 3 infinite lattice models with 2550 or 2600 ppm soluble boron. A very conservative bounding approach is to presume an un-poisoned fresh fuel batch so large that all storage locations in the model are loaded with un-poisoned fresh 5.0 w/o fuel. This approach is sufficient to demonstrate that k-eff is much less than 0.95 for Regions 1 and 3 and that Region 2 is the most limiting Region for the multiple mis-load condition. Region 2 has the smallest cell pitch of the three rack designs, has the least room for borated water, and contains borated panels which reduced soluble boron efficiency by raising EALF. These features would require more soluble boron using this bounding approach, therefore a more realistic approach is used.

Table 12.3 shows the fuel batch history of MPS3. Batch characteristics of interest are as follows:

- The largest batch size to date is 88 fuel assemblies (maximum enrichment 4.5 w/o).
- The highest batch average enrichment is 4.9 w/o (72 assemblies).
- Removable burnable poisons were used in Cycles 1 and 2. Thereafter, the minimum assembly average IFBA loading in any sub-batch has been 55 rods.
- IFBA loadings in Cycles 5 and 6 used "1.0x" IFBA enrichment. All cycles thereafter have used "1.5x".
- For all IFBA cycles, the largest number of assemblies with no IFBA is 20. Since Cycle 7, every fuel assembly has contained at least 32 1.5x IFBA rods with 12 inch maximum cutback.

Based on this history, a composite most reactive anticipated fuel batch can be developed:

- 24 fresh un-poisoned 5.0 w/o fuel assemblies
 - Maximum allowable fuel enrichment
 - More un-poisoned fuel assemblies than in any IFBA cycle
- All other fuel assemblies 5.0 w/o with 32 1.5x IFBA rods and 12 inch cutback. IFBA B-10 content is conservatively reduced 10%.
 - Higher fuel enrichment than any historical batch average
 - Minimum number of IFBA rods used in any assembly with IFBA to date
 - Lower number of IFBA rods than in any IFBA fuel batch to date
 - More fuel assemblies than any batch

A modified infinite lattice 20x20 storage cell KENO model is used for the Region 2 multiple mis-load evaluation. Figure 12.4 is a KENO3D representation of the model. The most reactive fuel

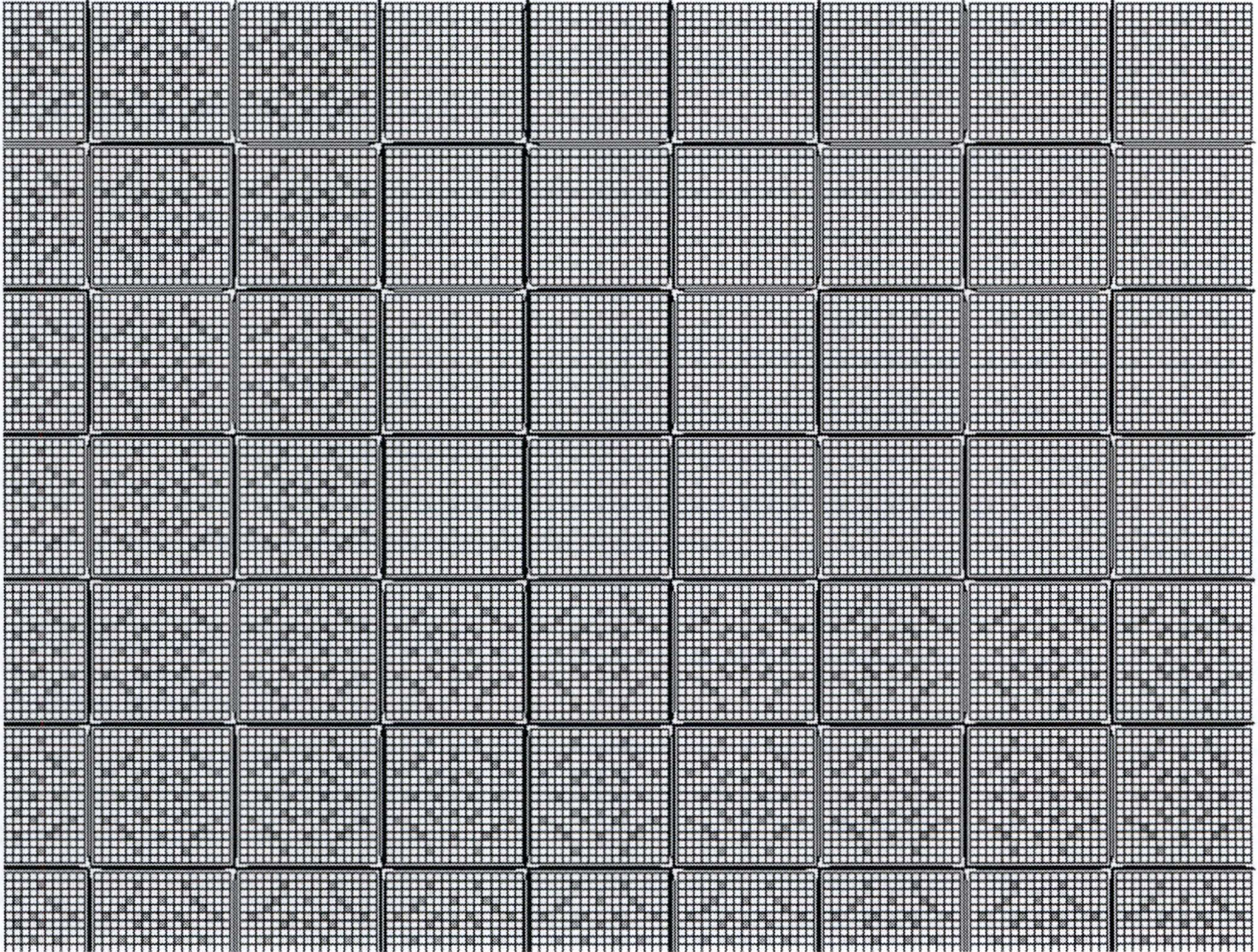
Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 227 of 300

(un-poisoned) is conservatively grouped together in a 6x4 block. Periodic boundary conditions are used.

Table 12.3: MPS3 Fuel Batch History

Cycle	Batch	Enrich. w/o	# FA	Avg. Enrich	# IFBA	Average IFBA / FA	FA w/o IFBA	Minimum IFBA/FA
1	1	2.4	65	2.90	0*	0	65	0
	2	2.9	64		0*	0	64	0
	3	3.4	64		0*	0	64	0
2	4A	3.50	56	3.60	0*	0	56	0
	4B	3.80	28		0*	0	28	0
3	5A	4.10	32	4.33	2448	77	0	32
	5B	4.50	44		2656	60	12	0
4	6A	4.20	32	4.39	4096	128	0	128
	6B	4.50	56		3968	71	16	0
5	7	4.40	84	4.40	4608	55	20	0
6	8	4.60	84	4.60	8896	106	0	32
7	9A	4.40	21	4.70	2400	114	0	104
	9B	4.80	60		5184	86	0	48
8	10A	4.40	36	4.61	4608	128	0	128
	10B	4.80	40		2624	66	0	48
9	11A	4.20	41	4.45	5248	128	0	128
	11B	4.70	40		2688	67	0	64
10	12A	4.70	16	4.89	2048	128	0	128
	12B	4.95	48		4512	94	0	32
	12C	4.95	8		832	104	0	104
11	13A	4.00	37	4.47	4544	123	0	104
	13B	4.95	36		2976	83	0	32
12	14A	4.70	36	4.83	4224	117	0	80
	14B	4.95	40		3360	84	0	32
13	15A	4.10	56	4.34	6144	110	0	64
	15B	4.90	24		2048	85	0	48
14	16A	4.10	37	4.58	4352	118	0	104
	16B	4.95	48		4544	95	0	48
15	17A	4.10	56	4.38	6464	115	0	48
	17B	4.95	28		2112	75	0	48
16	18A	4.10	53	4.42	5824	110	0	80
	18B	4.95	32		2592	81	0	48
17	19A	4.10	52	4.42	5312	102	0	80
	19B	4.95	32		2784	87	0	48

Figure 12.4: Region 2 KENO Multiple Mis-load Model (partial X-Y view)

Multiple fresh fuel batch mis-load results are shown in Table 12.4. Loading Region 1 with un-poisoned fresh 5.0 w/o fuel produces a very low k-eff (~ 0.76) with 2550 soluble boron in the SFP water. Region 1 is non-limiting for the multiple mis-load accident. Loading Region 3 with un-poisoned fresh 5.0 w/o fuel produces a low k-eff (~ 0.86) with 2550 soluble boron in the SFP water. Region 3 is non-limiting for the multiple mis-load accident.

Soluble boron is less effective in reducing k-eff in Region 2 because the Region 2 racks have the smallest cell pitch of the three rack designs in the MPS3 SFP. Region 2 cases are modeled using the most reactive anticipated fuel batch. Region 2 multiple misload results in Table 12.4

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 229 of 300

with 2550 ppm soluble boron show essentially the same k-eff at 32°F and 150°F when the benchmarking temperature bias of 0.008 is added to the 150°F case. Two 2600 ppm cases are provided: a base case, and a 12 inch IFBA cutback case. The base case (and all tolerance cases in Table 12.5 modeled 9 inch IFBA cutback at the top and bottom of each IFBA pin rather than 12 inches. The 12 inch cutback increases k-eff by only 0.0001 dk. This case k-eff is used as the "Base Case k-eff" in the Table 12.6 calculation of bias, uncertainty and margin.

Table 12.5 contains the results of Region 2 multiple mis-load tolerance cases with 2550 ppm soluble boron. Total bias and uncertainty, including 0.01 dk NRC administrative margin, is nearly the same as for 600 ppm boron normal storage. This illustrates that it is acceptable to apply tolerance cases run at 2550 ppm to the 2600 ppm 95/95 k-eff. Table 12.6 shows the calculation of total bias, uncertainty, and margin. Code uncertainty and bias values are for UO2 fuel using a maximum EALF of 0.65 eV. With 2600 ppm soluble boron, the k-eff limit is met with Dominion administrative margin over 0.006 dk.

Table 12.4: Multiple Mis-load Base Case Results

Region	Enrich. (w/o)	Burnup (MWd/MTU)	Boron (ppm)	Temp. (F)	K-eff	Uncert.	EALF	Notes
1	5.0	0	2550	32	0.76359	0.00006	0.62	All 5.0 fresh, no IFBA
1	5.0	0	2550	150	0.76221	0.00006	0.675	All 5.0 fresh, no IFBA
2	5.0/5.0*	0/0*	2550	32	0.92495	0.00007	0.598	24 fresh 5.0 no IFBA, 376 fresh 5.0 32 IFBA in 20x20**
2	5.0/5.0*	0/0*	2550	150	0.92413	0.00006	0.651	24 fresh 5.0 no IFBA, 376 fresh 5.0 32 IFBA in 20x20**
2	5.0/5.0*	0/0*	2600	32	0.92133	0.00006	0.605	24 fresh 5.0 no IFBA, 376 fresh 5.0 32 IFBA in 20x20**
2	5.0/5.0*	0/0*	2600	32	0.92142	0.00006	0.604	24 fresh 5.0 no IFBA, 376 fresh 5.0 32 IFBA in 20x20, 12 inch IFBA cutback
3	5.0	0	2550	32	0.85038	0.00006	0.426	All 5.0 fresh
3	5.0	0	2550	150	0.85760	0.00007	0.451	All 5.0 fresh

*Case has 2 Regions of fuel

**Case has 9 inch IFBA cutback

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 230 of 300

Table 12.5: Multiple Mis-load Tolerance Results (2550 ppm boron)

Case	k-eff	Uncert. (dk)	EALF	Tolerance (dk)*	Notes
Base	0.92495	0.00007	0.598	N/A	Base case 20x20, 32F, 5.0 w/o, 24 unpoisoned, 376 with 32 IFBA. 2550 ppm boron
Rack wall thickness	0.92557	0.00006	0.599	0.00079	Increase cell wall thickness
Rack cell pitch	0.92748	0.00006	0.608	0.00271	Decrease rack cell pitch
Rack cell ID**	0.92482	0.00007	0.598	0.00033	Decrease rack cell ID
Fuel density	0.92614	0.00006	0.603	0.00137	Increase density 0.51%
Wrapper thickness	0.92524	0.00006	0.598	0.00047	Increase wrapper thickness
Wrapper width**	0.92481	0.00006	0.598	0.00032	Decrease wrapper width
Fuel OD	0.92547	0.00007	0.601	0.00070	Increase fuel OD
Clad OD	0.92506	0.00006	0.594	0.00029	Decrease clad OD
Clad ID	0.92504	0.00006	0.599	0.00026	Increase clad ID
Fuel pin pitch	0.92514	0.00007	0.597	0.00038	Increase fuel pin pitch
GT OD	0.92500	0.00006	0.597	0.00023	Decrease GT OD
GT ID	0.92503	0.00007	0.597	0.00026	Increase GT ID
Poison width	0.92665	0.00007	0.595	0.00189	Decrease BORAL width
Poison thickness	0.92545	0.00007	0.600	0.00069	Increase BORAL thickness
Fuel stack length	0.92497	0.00006	0.598	0.00019	Increase stack height
Blisters	0.92546	0.00006	0.600	0.00068	Void in wrapper water space
Maximum grids	0.92492	0.00006	0.602	0.00014	Maximum Zr grid
Large GT	0.92512	0.00006	0.601	0.00034	Larger GT

*Includes 2xRSS uncertainty

**Tolerance direction reversed vs. 0 ppm conditions, tolerance calculation reversed

Table 12.6: Multiple Mis-load Bias, Uncertainty and Margin

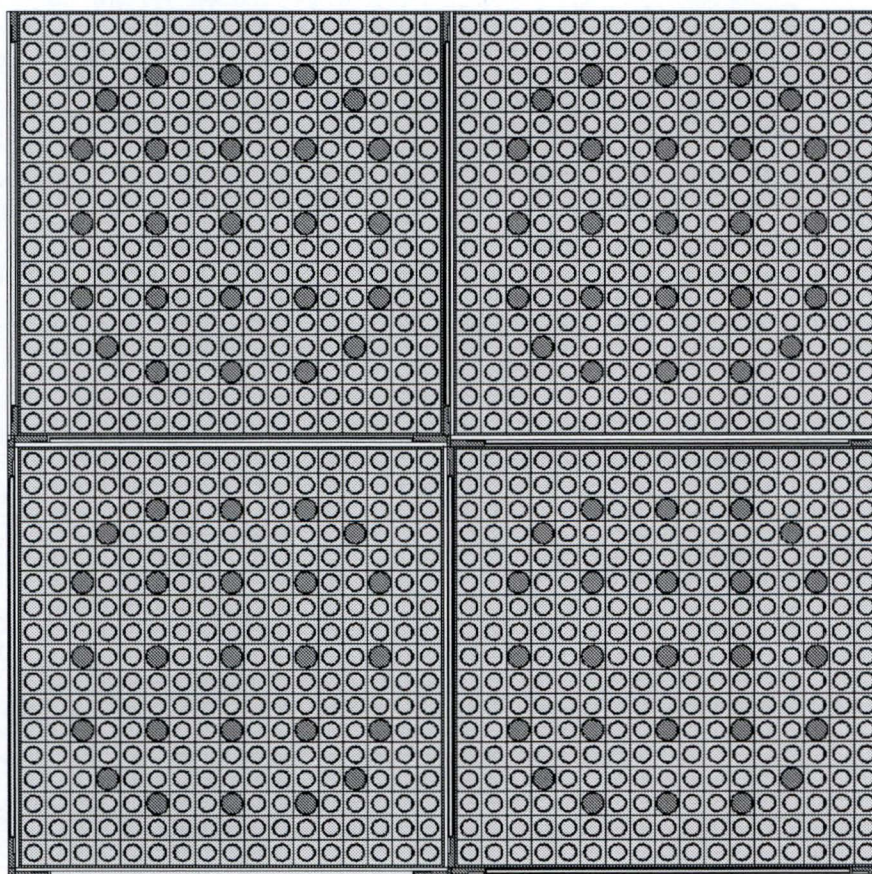
Soluble Boron (ppm)	2600
Burnup (GWd/MTU)	0
Enrichment (wt%)	5.0
Uncertainties (dk)	
Rack wall thickness	0.0008
Rack cell pitch	0.0027
Rack cell ID	0.0003
Fuel density	0.0014
Enrichment	0.0000
Wrapper thickness	0.0005
Wrapper width	0.0003
Fuel OD	0.0007
Clad OD	0.0003
Clad ID	0.0003
Fuel pin pitch	0.0004
GT OD	0.0002
GT ID	0.0003
Poison width	0.0019
Poison thickness	0.0007
Fuel stack length	0.0002
2 x KENO std. dev.	0.0001
Code uncertainty	0.0051
RSS OF UNCERTAINTIES	0.0064
Biases (dk)	
Code bias	0.0047
Blisters	0.0007
Guide tube design	0.0003
SUM OF BIASES	0.0057
Summary	
Base Case k-eff	0.9214
Total Bias and Uncertainty	0.0122
NRC Administrative Margin	0.0100
Maximum k-eff	0.9436
10CFR50.68 Limit	0.9500
Dominion Margin (dk)	0.0064

*Uncertainty and bias values were calculated with 2550 ppm

12.1.4 Dropped Assembly

To simulate a fuel assembly damaged as a result of being dropped into a rack cell, the fuel rod pitch is increased to represent grid damage in order to maximize damaged fuel reactivity. A fuel assembly with increased pin pitch is placed into every rack cell. This approach is conservative because all fuel in the rack is modeled as damaged and because depleted fuel is represented using only major actinides. Figure 12.5 shows an X-Y plane view of the Region 2 KENO dropped assembly model.

Figure 12.5: Region 2 Dropped Assembly Model



Dropped assembly accident results in Table 12.7 confirm that the dropped assembly accident is not limiting. These cases conservatively assume all fuel in the SFP is damaged such that the fuel pin pitch increases to an optimal value within the storage cell. Maximum k_{eff} for any region is 0.84. The dropped assembly event is not limiting.

Table 12.7: Dropped Assembly Results

Region	Enrich. (w/o)	Burnup (GWd/MTU)	Boron	Temp. (F)	K-eff	Uncert.	EALF	Notes
1	5.0	0	2550	32	0.76359	0.00006	0.620	All 5.0 fresh, no IFBA
1	5.0	0	2550	32	0.76603	0.00006	0.592	Pin pitch +0.02 cm
1	5.0	0	2550	32	0.76741	0.00006	0.567	Pin pitch +0.04 cm
2	5.0	40	2550	32	0.83245	0.00006	0.616	Nominal pitch
2	5.0	40	2550	32	0.83731	0.00005	0.596	Pin pitch +0.02 cm
2	5.0	40	2550	32	0.84005	0.00006	0.581	Pin pitch +0.04 cm
3	5.0	53	2550	150	0.72689	0.00006	0.498	Nominal pitch
3	5.0	53	2550	150	0.73586	0.00005	0.467	Pin pitch +0.02 cm
3	5.0	53	2550	150	0.74532	0.00006	0.439	Pin pitch +0.04 cm
3	5.0	53	2550	150	0.74898	0.00006	0.430	Pin pitch +0.047 cm

12.1.5 Handling Error

A fuel handling error is simulated by modeling two adjacent unpoisoned fresh 5.0 w/o fuel assemblies in water with 2550 ppm soluble boron. Fuel assembly separation of 0, 2 and 4 cm are considered. Fuel handling error results in Table 12.8 confirm that the handling error accident is not limiting. Maximum k-eff is 0.77 with no separation between fuel assemblies.

Table 12.8: Dropped Assembly Results

Enrich. (wt%)	Burnup (GWd/MTU)	Boron (ppm)	Temp. (F)	k-eff	Uncert.	EALF	Notes
5.0	0	2550	150	0.77425	0.00007	0.543	2 FA in water with 0 cm separation
5.0	0	2550	150	0.75303	0.00006	0.501	2 FA in water with 2 cm separation
5.0	0	2550	150	0.71605	0.00007	0.513	2 FA in water with 4 cm separation

12.1.6 Seismic Event

The Region 1 and 2 rack design provides more flux trap space between storage cells of adjacent racks than the nominal interior cell spacing (details in Sections 4.2.1 and 4.2.2 for Regions 1 and 2, respectively). This occurs because minimum rack to rack spacing is limited by the protrusion of the rack baseplate beyond the storage cell envelope. Therefore, seismic activity cannot result in Region 1 or 2 storage cells being closer together than in the infinite lattice model.

The Region 3 racks have no physical barriers preventing them from moving closer together during a seismic event. The Region 3 KENO model includes the as-installed minimum rack-to-rack spacing of 1 inch. The rack-to-rack separation uncertainty case of -0.125 inches was worth 0.0006 Δk (shown in Table 9.28). Multiplying this tolerance by 8 to simulate no space in-between rack walls results in a worth of $\sim 0.005 \Delta k$. Table 12.1 shows that Region 3 with 2550 ppm of boron has a k-eff of 0.72689. There is more than enough margin to accommodate an increase of 0.005 Δk . Therefore, the seismic event is bounded by the multiple mis-load accident.

12.2 Accident Analysis Summary

Evaluation of accident scenarios including single fuel assembly mis-load in the fuel racks, multiple fuel assembly mis-load in the fuel racks, loss of cooling including partial voiding, dropped assembly into the racks with grid damage (optimum fuel pin pitch), mis-placement of an assembly between fuel racks, and a fuel handling error confirms that 2600 ppm is more than sufficient to ensure $k\text{-eff} < 0.95$. The multiple mis-load accident results in the highest k-eff.

13 Summary and Conclusions

A SFP criticality analysis is presented for MPS3 to allow for a measurement uncertainty recapture power uprate, to incorporate spent fuel pool (SFP) soluble boron credit, to update computer codes and analysis methods, to eliminate the need for empty storage cells and cell blockers, and to increase identified margin to k-eff limits. A new fuel storage area (NFSA) criticality analysis is performed for MPS3 to update computer codes and analysis methods and to increase identified margin to k-eff limits. The current enrichment limit for the MPS3 NFSA and SFP is 5.0 wt% U-235.

As part of the criticality analyses, computer code validation is described in Appendix A. In support of the use of boron credit in the SFP criticality analysis, a boron dilution analysis is provided in Attachment 7 that identifies potential SFP dilution events, dilution water source flow rates and volumes, means of detection, time required for detection and mitigation, and minimum dilution event soluble boron concentration.

13.1 New Fuel Storage Area

The criticality analysis demonstrates adequate margin to the k-eff limits with the NFSA fully loaded with a fresh 5.0 wt% fuel assembly in each storage cell. A composite bounding fuel assembly design is used (Table 13.1).

Table 13.1: Bounding Fuel Design Values

Enrichment	≤ 5.0 wt% U-235
Pellet Diameter	0.3225 inch
Clad Inner Diameter	0.329 inch
Clad Outer Diameter	0.374 inch
Clad Material	Zirconium alloy
Rod Pitch	0.496 inch
Grid Volume	[] ^{a,c} , excluding Inconel grids
Grid Material	Zirconium alloy
Pellet Stack Net Density	≤ 95.5% of theoretical

13.2 Spent Fuel Pool

The MPS3 SFP contains three unique storage rack designs (Regions 1, 2, and 3). The criticality analysis demonstrates adequate margin to the k-eff limits in each of the three SFP Regions assuming four-out-of-four storage in each Region. Regions 1 and 2 credit installed BORAL neutron absorber. Region 3 contains but does not credit BORAFLEX neutron absorber. A composite bounding fuel assembly design is used (Table 13.2).

13.3 Bounding Fuel Design Values

Table 13.2 summarizes the bounding fuel design used for this analysis.

Table 13.2: Bounding Fuel Design Values

Enrichment	≤ 5.0 wt% U-235
Pellet Diameter	0.3225 inch
Clad Inner Diameter	0.329 inch
Clad Outer Diameter	0.374 inch
Clad Material	Zirconium alloy
Rod Pitch	0.496 inch
Grid Volume	[] ^{a,c}
Grid Material	Zirconium alloy
Pellet Stack Net Density	$\leq 95.5\%$ of theoretical
Grid Material	A low growth zirconium alloy such as ZIRLO. Zircaloy 4 is covered for existing inventory.

Axial blankets are not credited in the analysis and are permitted as well as annular pellets.

Burnable absorbers are credited in portions of the analysis and affect the fuel depletion.

Depletion related limits for burnable absorbers follow in the next section.

13.4 Bounding Depletion Condition Input

Depletion parameters were selected to cover past and anticipated future operation. Table 13.3 lists key depletion condition input selected to bound actual fuel depletion conditions. The temperature and soluble boron assumptions are averages over the total burnup (multi-cycle) for a given assembly. Exceptions to these parameters for existing inventory are justified in the analysis.

Table 13.3: Bounding Depletion Conditions for Burnup Credit

Parameter	Value	Notes
Maximum Burnup Averaged Soluble Boron	≤ 1050 ppm	This is an average for all cycles in which the assembly was depleted.
Maximum Core Average Moderator Exit Temperature	≤ 623.19 °F	This is not a value used directly in the analysis but is a proxy for the nodal moderator and fuel temperatures.
Maximum Burnup Averaged Relative Assembly Power	\leq Figure 8.1	0 to 30 GWd/MTU BARAP ≤ 1.44 Linearly decreases to 1.30 at 52 GWd/MTU Then linearly decreases to 1.0 at 60 GWd/MTU
Maximum Removable Burnable Absorbers with IFBA	≤ 8 fingered WABA with up to 200 IFBA, ≤ 24 fingered WABA with no IFBA	Analysis assumed 24 WABA as well as some combinations of IFBA and WABA.

13.5 Summary of Loading Constraints

Storage geometries for each Region of the SFP are listed in Table 13.4. Region 1 is composed of subregions 1A and 1B. Region 1B occupies the 2 rows of storage cells in each Region 1 rack module closest to the West SFP wall. Figure 13.1 shows the Region 1A/1B configuration which credits neutron leakage at the SFP wall. Maximum fuel enrichment for storage in the NFSA and the SFP is 5.0 wt% U-235. The Region 2 and Region 3 burnup credit curves are shown in Figures 13.2 and 13.3, respectively. Polynomial coefficients for each curve are shown in Table 13.5.

Satisfaction of the burnup curve requirement shall use fuel assembly characteristics as follows:

- 1) The fuel enrichment is the maximum planar volume averaged as-built initial enrichment in the assembly (blanket enrichment is not credited by averaging with the higher enrichment central zone).
- 2) The fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates with no reduction for measurement uncertainty.
- 3) The fuel assembly decay time is the time elapsed since last use at power in the reactor core.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 238 of 300

Table 13.4: MPS3 SFP Storage Constraint Summary

Region	Geometry	Credits	Description
1A	4 out of 4	None	≤ 4.75 wt% U-235
1A	4 out of 4	≥ 12 IFBA	≤ 5.00 wt% U-235
1A	4 out of 4	Burnup	≥ 2 GWd/MTU, ≤ 5.00 wt% U-235
1B	4 out of 4	SFP wall	≤ 5.0 wt% U-235, two rows of rack cells nearest the West SFP wall (Figure 13.1).
2	4 out of 4	Burnup	Table 13.5
2	4 out of 4	Control rod	≤ 5.00 wt% U-235
3	4 out of 4	Burnup, decay time	Table 13.5

Assemblies reconstituted with stainless steel rods can be loaded using their burnup and enrichment just as any non-reconstituted assembly. During reconstitution, the assembly must be isolated by vacating the four face-adjacent cells or by ensuring there is at least 12 inches of separation between the assembly being handled and any other assembly. The Fuel Rod Storage Canister and any non-fuel items are allowed to be stored in any storage location that an assembly can be stored. Depleted discrete burnable poisons can be stored in assembly guide tubes.

Figure 13.1: Region 1A/1B Orientation

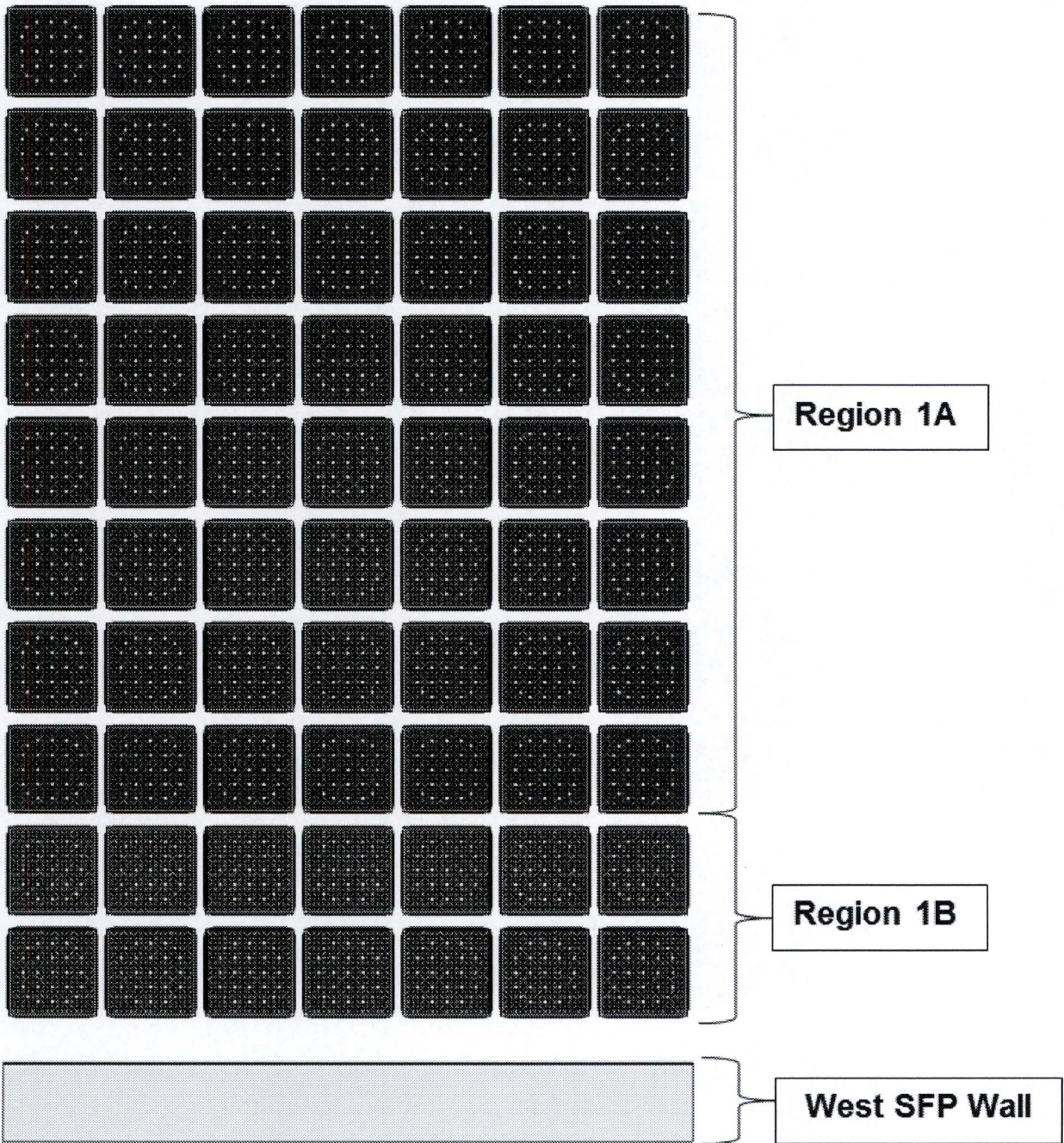


Figure 13.2: Region 2 Burnup Credit Curve

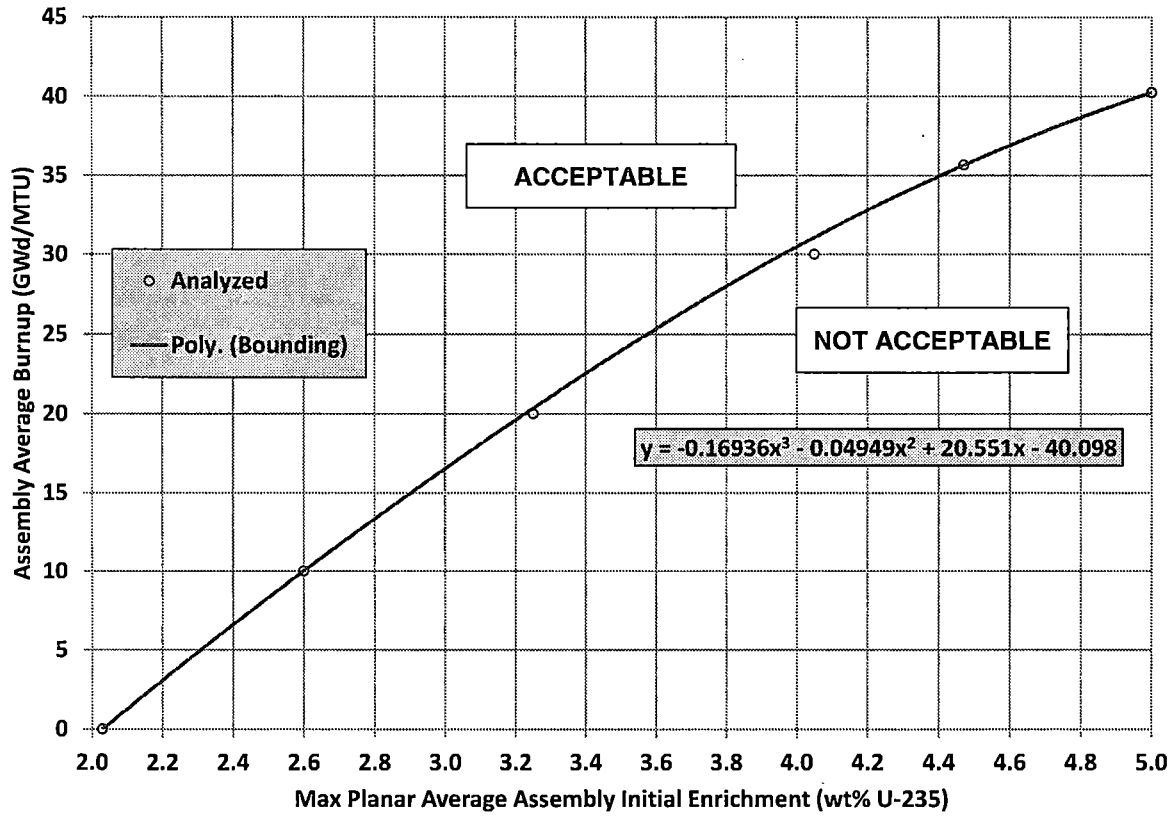


Figure 13.3: Region 3 Burnup Credit Curves

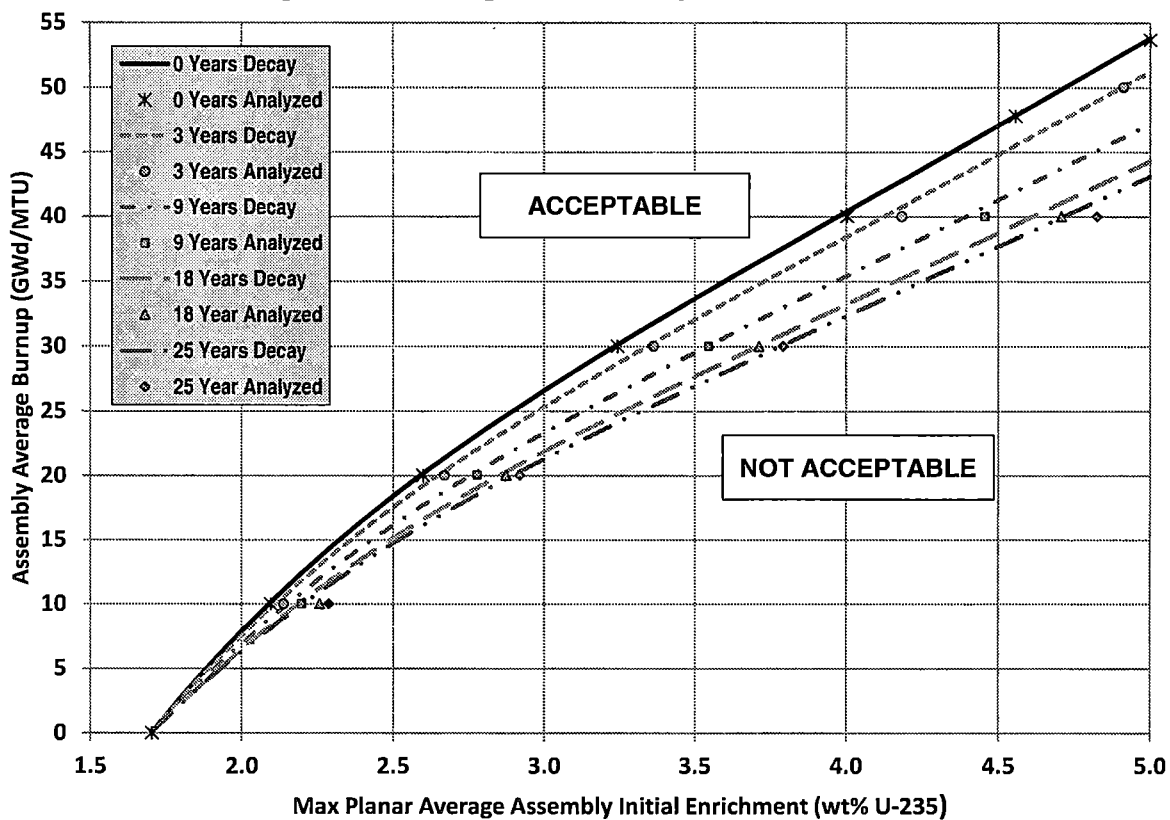


Table 13.5: Burnup Credit Curve Polynomial Coefficients

Region	Decay Time (Years)	a_4	a_3	a_2	a_1	a_0
2	No Credit	N/A	-0.16936	-0.04949	20.551	-40.098
3	No Credit	-0.2459	4.208	-26.80	88.70	-92.00
3	3 Years	-0.2338	4.001	-25.48	84.34	-87.47
3	9 Years	-0.2153	3.684	-23.46	77.66	-80.54
3	18 Years	-0.2020	3.458	-22.02	72.88	-75.59
3	25 Years	-0.1964	3.361	-21.40	70.84	-73.47

The burnup curve equations have the following polynomial format:

$$\text{BU [GWD/MTU]} = a_4 * \text{wt}\%^4 + a_3 * \text{wt}\%^3 + a_2 * \text{wt}\%^2 + a_1 * \text{wt}\%^1 + a_0$$

14 References

- [1] Code of Federal Regulations, Title 10, Part 50, Section 68, "*Criticality Accident Requirements.*"
- [2] Code of Federal Regulations, Title 10, Part 50, Appendix A, "*General Design Criteria for Nuclear Power Plants,*" Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."
- [3] NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Section 9.1.1 Revision 3 (Criticality Safety of Fresh and Spent Fuel Storage and Handling), U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.
- [4] *Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01 Revision 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,* Nuclear Regulatory Commission, October 2011. (ADAMS Accession Number ML110620086.)
- [5] Dominion Resources Services, Inc. Letter to NRC, "Response to Generic Letter (GL) 2016-01, Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools," 11/1/2016 (Serial No. 16-161).
- [6] Dominion Letter, "DOMINION NUCLEAR CONNECTICUT, INC. MILLSTONE POWER STATION UNIT 3 RESPONSE TO REQUEST FOR SUPPLEMENTAL INFORMATION REGARDING GENERIC LETTER 2016-01, MONITORING OF NEUTRON-ABSORBING MATERIALS IN SPENT FUEL POOLS", 11/22/2017 (Serial No. 17-447, ADAMS Accession No. ML17338A05).
- [7] "*SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,*" ORNL/TM-2005/39, Version 6, Volumes 1-3, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA, January 2009.
- [8] S. Goluoglu, N. F. Landers, L. M. Petrie, and D. F. Hollenbach, "*CSAS5: Control Module For Enhanced Criticality Safety Analysis Sequences with KENO V.A,*" ORNL/TM-2005/39, Version 6, Vol. 1, Section C5, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA, January 2009.
- [9] J. C. Wagner, M. D. DeHart, and C. V. Parks, "*Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,*" US Nuclear Regulatory Commission, NUREG/CR-6801 Oak Ridge National Laboratory, Oak Ridge, Tenn. (2003).

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 244 of 300

- [10] Letter from U.S.N.R.C (Richard Guzman) to Dominion (D. Heacock), "Millstone Power Station, Unit No. 3 – Issuance of Amendment Adopting Dominion Core Design And Safety Analysis Methods and Addressing The Issues Identified in Three Westinghouse Communication Documents (CAC NO. MF6251).", July 28, 2016. Dominion Energy Nuclear Licensing Serial Number 16-317 (Received August 2, 2016). NRC Accession Number: ML16131A728
- [11] M. A. Jessee, and M. D. DeHart, "*TRITON: A Two-Dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel*," ORNL/TM-2005/39, Version 6 Vol. 1, Section T1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA, January 2009.
- [12] J.C. Dean and R.W. Tayloe, Jr., *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, NUREG/CR-6698, Nuclear Regulatory Commission, Washington, DC January 2001
- [13] J. M. Scaglione, D. E. Mueller, J.C. Wagner and W. J. Marshall, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (keff) Predictions*, US Nuclear Regulatory Commission, NUREG/CR-7109, Oak Ridge National Laboratory, Oak Ridge, Tenn. (2012).
- [14] Letter from U.S.N.R.C (Richard Guzman) to Dominion (D. Heacock), "Millstone Power Station, Unit No. 2 – Issuance of Amendment RE: Technical Specification Changes for Spent Fuel Storage (TAC NO. MF0435)", June 23, 2016. NRC ADAMS Accession Number: ML16003A008
- [15] USNRC Reg. Guide 1.183, "*Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*", US Nuclear Regulatory Commission, Washington, DC, July 2000.
- [16] C. E. Beyer, and P. M. Clifford, "*Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard*", PNNL-18212 Rev. 1, Pacific Northwest National Laboratory, Richland, Washington, June 2011. ADAMS Accession Number ML112070118.
- [17] J. O. Barner, "*Characterization Of LWR Spent Fuel MCC - Approved Testing Material--ATM-101*," PNL-5109 Rev. 1, Pacific Northwest National Laboratory, Richland, Washington (1985).
- [18] R. J. Guenther, et al, "*Characterization Of Spent Fuel Approved Testing Material--ATM-103*," PNL-5109-103, Pacific Northwest National Laboratory, Richland, Washington (1988).
- [19] R. J. Guenther, et al, "*Characterization Of Spent Fuel Approved Testing Material--ATM-104*," PNL-5109-104, Pacific Northwest National Laboratory, Richland, Washington (1991).
- [20] R. J. Guenther, et al, "*Characterization Of Spent Fuel Approved Testing Material--ATM-105*," PNL-5109-105, Pacific Northwest National Laboratory, Richland, Washington (1991).

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 245 of 300

- [21] R. J. Guenther, et al, "*Characterization Of Spent Fuel Approved Testing Material--ATM-106*," PNL-5109-106, Pacific Northwest National Laboratory, Richland, Washington (1988).
- [22] A. Simpson, T. Marlow, J. Franco, M. Clapham and A. Chesterman, "Operational Experience in Radiometric Instrumentation for Spent Fuel Monitoring," Proceedings of the Institute of Nuclear Materials Management: INMM 44th Annual Meeting, Phoenix, Arizona, July 13-17, 2003, Institute of Nuclear Materials Management, Oakbrook Terrace, IL (2003).
- [23] Letter from Dominion (M. Sartain) to U.S.N.R.C, "Virginia Electric and Power Company (DOMINION) North Anna Power Station Units 1 and 2 Proposed Licensing Amendment Request for Spent Fuel Storage and New Fuel Storage (Serial No. 16-383)", May 2, 2017. NRC ADAMS Accession Number: ML17128A451
- [24] "Studsvik Scandpower Inc. Approved Version Submittal for TOPICAL REPORT SSP-14-P01/028-TR, "GENERIC APPLICATION OF THE STUDSVIK SCANDPOWER CORE MANAGEMENT SYSTEM TO PRESSURIZED WATER REACTORS" (CAC NO. MF7273)", October 6, 2017. NRC ADAMS Accession Number: ML17279A983
- [25] ZIRAT14 Special Topical Report, "In-Reactor Creep of Zirconium Alloys," September 2009, Advanced Nuclear Technology International.
- [26] *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)3, Volumes IV and VI, Nuclear Energy Agency, OECD, Paris, September, 2014.
- [27] D. E. Mueller, K. R. Elam, and P. B. Fox, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979 (ORNL/TM-2007/083), prepared for the US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn., September 2008
- [28] U.S. Nuclear Regulatory Commission, *Spent Fuel Project Office Interim Staff Guidance – 8, Rev. 3 – Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*, U.S. Nuclear Regulatory Commission, April, 2012.
- [29] A. Tsiboulia, et al., "Water-Moderated U(4.92)O₂ Fuel Rods in 1.29, 1.09, and 1.01 cm Pitch Hexagonal Lattices at Different Temperatures," LEU-COMP-THERM-026, *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)3, Volume IV, Nuclear Energy Agency, OECD, Paris, September, 2014.
- [30] A. D. Santos, et al., "Critical Loading Configurations of the IPEN/MB-01 Reactor Considering Temperature Variation From 14° C to 85° C," LEU-COMP-THERM-046, *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)3, Volume IV, Nuclear Energy Agency, OECD, Paris, September, 2014.

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 246 of 300

- [31] D. Lancaster, *Sensitivity Analyses for Spent Fuel Pool Criticality*, Technical Report EPRI Doc No. 3002003073, December 2014.
- [32] ANS-8.1/N16.1-1975, "Criticality safety in operations with fissionable material".
- [33] YAE-1937, "Axial Burnup Profile Database for Pressurized Water Reactors", 05/1997.
- [34] EPRI Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications, 11/25/2009.
- [35] NRC Letter, "SUMMARY OF APRIL 21, 2004, MEETING WITH PACIFIC GAS & ELECTRIC AND AAR TO DISCUSS BLISTERING ON BORAL MATERIAL USED IN THE SPENT FUEL POOL AT THE DECOMMISSIONING HUMBOLDT BAY PLANT", April 21 2004.
- [36] Regulatory Information Conference, "Degradation of Boral in Spent Fuel Pool at Taiwan's BWR NPPs", 3/13/2014.
- [37] WCAP-15063-P-A, Westinghouse Electric Company LLC., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)", July 2000
- [38] Dennis Gottuso, Jean-Noel Canat, and Pierre Mollard, "A Family of Upgraded Fuel Assemblies for PWR," *Top Fuel 2006, 2006 International Meeting on LWR Fuel Performance*, October 22-26, 2006, Salamanca, Spain, European Nuclear Society.
- [39] Not Used.
- [40] Not Used.
- [41] Not Used.
- [42] David Mitchell, Anand Garde, and Dennis Davis, "Optimized ZIRLO™ Fuel Performance in Westinghouse PWRs," *Proceedings of the 2010 LWR Fuel Performance Meeting/Top Fuel/WRFPM*, September 26-29, 2010, Orlando, Florida, USA, American Nuclear Society, La Grange Park, Illinois.

Appendix A: Validation for Criticality Analysis Using Laboratory Critical Experiments

A.1. Overview

This appendix determines the computer code and cross-section library bias and uncertainty in the k's calculated for the MPS3 spent fuel pool and new fuel storage area when using the CSAS5 module of SCALE 6.0 and the 238 Group ENDF/B-VII.0 cross section library. [A.1] The CSAS5 module executes the CENTRM and BONAMI programs for the resonance self-shielding calculations and KENO V.a for the Monte Carlo calculation of k. All the computer runs use a large Monte Carlo sampling of at least 1500 generations and 6000 neutrons per generation. The bias and uncertainties determined in this Appendix covers the major actinides plus structural and absorber materials.

This validation follows the direction of NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" [A.2]. The guide establishes the following steps for performing the validation:

1. Define operation/process to identify the range of parameters to be validated
2. Select critical experiment data
3. Model the experiments
4. Analyze the data
5. Define the area of applicability of the validation and limitations

It further defines the steps of "Analyze the data" as:

1. Determine the Bias and Bias Uncertainty
2. Identify Trends in Data, Including Discussion of Methods for Establishing Bias Trends
3. Test for Normal or Other Distributions
4. Select the Statistical Method for Treatment of Data
5. Identify and Support Subcritical Margin
6. Calculate the Upper Safety Limit

The validation consists of modeling 232 UO₂ and 136 MOX critical experiments for the determination of the bias and the uncertainty in the calculation of k for fresh fuel and spent fuel. In addition 17 critical experiments are modeled to determine a temperature bias.

A.2. Definition of the Range of Parameters to Be Validated

The validation guidance document [A.2] states:

“Prior to the initiation of the validation activity, the operating conditions and parameters for which the validation is to apply must be identified. The fissile isotope, enrichment of fissile isotope, fuel density, fuel chemical form, types of neutron moderators and reflectors, range of moderator to fissile isotope, neutron absorbers, and physical configurations are among the parameters to specify. These parameters will come to define the area of applicability for the validation effort.”

The fuel is low enriched uranium dioxide (less than or equal to 5.0 wt% U-235). The fuel is in pellets with a density of greater than 94% of the theoretical density. The only significant neutron moderators are water and the oxygen in the fuel pellet. The neutron absorbers credited are boron (in Boral panels, IFBA coatings, or in solution) and Ag-In-Cd or Hf control rods which may be credited. The reflectors are water, steel, or concrete. The fuel is in assemblies with a rectangular pitch. The assemblies are arranged in cells with space between the cells. The assemblies and cells are in water with varying density and temperature.

A.3. Selection of the Critical Benchmark Experiments

A.3.1 Selection of the Fresh UO₂ Critical Benchmark Experiments

The UO₂ benchmarks that were selected met the following criteria:

- Low enriched (5 wt% U-235 or less) UO₂ to cover the principle isotopes of concern.
- Fuel in rods to assure that the heterogeneous analysis used in SCALE also is applied in the benchmark analysis.
- Square lattices to assure the lattice features of SCALE used in the rack analysis are also modeled in the critical benchmarks selected.

- Presence of soluble boron, boron bearing rods or plates and Cd to cover most of the control rod absorption.
- No emphasis on a feature or material not of importance to the rack analysis such as lead or copper.

The OECD/NEA *International Handbook of Evaluated Criticality Safety Benchmarks Experiments* [A.3] is the appropriate reference for criticality safety benchmarks. This handbook has reviewed the available benchmarks and evaluated the uncertainties in the experiments. The appropriate modeling is presented. All of the experiments used in this validation were taken from this handbook. Volume IV of the handbook is for low enriched uranium systems. The section of Volume IV of interest to this validation is the "Thermal Compound Systems." All of the experiments selected are numbered LEU-COMP-THERM-0XX. This validation will refer to the experiments LEU-COMP-THERM-0XX as just XX where any leading zero is not included. (Experiments are also referred to as LCT-XX.)

There are more critical experiments in the handbook that meet the requirements for this validation than would be necessary to use. However, most of the applicable available benchmarks were used. There are 92 sets of benchmarks in the September 2014 version of the handbook. Of these, 24 sets were excluded since they used hexagonal arrays; 4 sets were excluded due to enrichments of 6.9 wt% U-235 or higher; 8 sets were excluded due to not being water moderated fuel rods; 4 sets were excluded due to large experimental uncertainties; 8 experiment sets were excluded due to using materials that are not in spent fuel pool racks, such as copper tubes, Gadolinium rods or solution, Titanium screens, or lead reflectors. This leaves 44 benchmark sets of which 27 sets were used for this validation. The 17 unused benchmark sets were reviewed to be sure that there was no feature of the experimental set that was missing in the selected 25 sets. LCT-COMP-THERM-46 is used for the temperature bias is covered in Section A.4 but not included in this set.

The selected 27 benchmark sets include critical experiments from six different critical experiment facilities. The fuel was mainly clad in aluminum, but experiments with stainless steel and zirconium cladding were also in the set.

The critical benchmark sets generally contained multiple experiments, but not all cases from each critical benchmark set is used. In some sets there are experiments that utilize materials

that are outside the area of applicability for this validation, such as lead or copper reflectors.

The 27 selected benchmark sets resulted in **232 experiments** that are used for the statistical analysis.

A later section will evaluate the area of applicability provided by this selection of critical benchmarks.

Table A.3.1 provides a summary of all the low enriched thermal experiments (non-U metal) from the OECD/NEA handbook [A.3] and why some experiments were not used.

Table A.3.1: Selection Review of OECD/NEA Criticality Benchmarks
(All Experiments Start With LEU-COMP-THERM-)

Benchmark Number	Description	Lab	Selected
1	WATER-MODERATED U(2.35)O ₂ FUEL RODS IN 2.032-CM SQUARE-PITCHED ARRAYS	PNL	All 8
2	WATER-MODERATED U(4.31)O ₂ FUEL RODS IN 2.54-CM SQUARE-PITCHED ARRAYS	PNL	All 5
3	WATER-MODERATED U(2.35)O ₂ FUEL RODS IN 1.684-CM SQUARE-PITCHED ARRAYS (GADOLINIUM WATER IMPURITY)	PNL	None. Gd impurity not well known. Not benchmark quality.
4	WATER-MODERATED U(4.31)O ₂ FUEL RODS IN 1.892-CM SQUARE-PITCHED ARRAYS (GADOLINIUM WATER IMPURITY)	PNL	None. Gd impurity not well known. Not benchmark quality.
5	CRITICAL EXPERIMENTS WITH LOW-ENRICHED URANIUM DIOXIDE FUEL RODS IN WATER CONTAINING DISSOLVED GADOLINIUM	PNL	None. Soluble Gd not used in pools.
6	CRITICAL ARRAYS OF LOW-ENRICHED UO ₂ FUEL RODS WITH WATER-TO-FUEL VOLUME RATIOS RANGING FROM 1.5 TO 3.0	JAEA	All 18
7	WATER-REFLECTED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL-ROD ARRAYS	Valduc	Only 4 cases used rest are in hexagonal arrays.
8	CRITICAL LATTICES OF UO ₂ FUEL RODS AND PERTURBING RODS IN BORATED WATER	B&W	All 17
9	WATER-MODERATED RECTANGULAR CLUSTERS OF U(4.31)O ₂ FUEL RODS (2.54-CM PITCH) SEPARATED BY STEEL, BORAL, COPPER, CADMIUM, ALUMINUM, OR ZIRCALOY-4 PLATES	PNL	21 cases used. Did not include Copper cases

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 251 of 300

Benchmark Number	Description	Lab	Selected
10	WATER-MODERATED U(4.31)O ₂ FUEL RODS REFLECTED BY TWO LEAD, URANIUM, OR STEEL WALLS	PNL	22 cases used. Did not use lead cases since no lead in pools.
11	CRITICAL EXPERIMENTS SUPPORTING CLOSE PROXIMITY WATER STORAGE OF POWER REACTOR FUEL (PART I - ABSORBER RODS)	B&W	All 15
12	WATER-MODERATED RECTANGULAR CLUSTERS OF U(2.35)O ₂ FUEL RODS(1.684-CM PITCH) SEPARATED BY STEEL, BORAL, BOROFLEX, CADMIUM,OR COPPER PLATES (GADOLINIUM WATER IMPURITY)	PNL	None. Gd impurity not well known. Not benchmark quality.
13	WATER-MODERATED RECTANGULAR CLUSTERS OF U(4.31)O ₂ FUEL RODS (1.892-CM PITCH) SEPARATED BY STEEL, BORAL, BOROFLEX, CADMIUM, OR COPPER PLATES, WITH STEEL REFLECTING WALLS	PNL	5 cases used. Did not use the cases with copper.
14	WATER-REFLECTED ARRAYS OF U(4.31)O ₂ FUEL RODS (1.890-CM AND 1.715-CM SQUARE PITCH) IN BORATED WATER	PNL	None used. High boron content uncertainty. Not benchmark quality.
15	THE VVER EXPERIMENTS: REGULAR AND PERTURBED HEXAGONAL LATTICES OF LOW-ENRICHED UO ₂ FUEL RODS IN LIGHT WATER	KFKI	None used due to hex arrays.
16	WATER-MODERATED RECTANGULAR CLUSTERS OF U(2.35)O ₂ FUEL RODS (2.032-CM PITCH) SEPARATED BY STEEL, BORAL, COPPER, CADMIUM, ALUMINUM, OR ZIRCALOY-4 PLATES	PNL	26 cases used. Did not use the copper or borated panel cases
17	WATER-MODERATED U(2.35)O ₂ FUEL RODS REFLECTED BY TWO LEAD, URANIUM, OR STEEL WALLS	PNL	23 cases used. Did not use the 6 cases with a lead reflector.
18	LIGHT WATER MODERATED AND REFLECTED LOW ENRICHED URANIUM DIOXIDE (7 WT.%) ROD LATTICE	Winfrith	None used. Only 1 case Complex system.
19	WATER-MODERATED HEXAGONALLY PITCHED LATTICES OF U(5%)O ₂ STAINLESS STEEL CLAD FUEL RODS	Kurchatov Institute	None used due to hex arrays.
20	WATER-MODERATED HEXAGONALLY PITCHED PARTIALLY FLOODED LATTICES OF U(5%)O ₂ ZIRCONIUM CLAD FUEL RODS, 1.3-CM PITCH	Kurchatov Institute	None used due to hex arrays.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 252 of 300

Benchmark Number	Description	Lab	Selected
21	HEXAGONALLY PITCHED PARTIALLY FLOODED LATTICES OF U(5%)O ₂ ZIRCONIUM CLAD FUEL RODS MODERATED BY WATER WITH BORIC ACID	Kurchatov Institute	None used due to hex arrays.
22	UNIFORM WATER-MODERATED HEXAGONALLY PITCHED LATTICES OF RODS WITH U(10%)O ₂ FUEL	Kurchatov Institute	None used due to hex arrays.
23	PARTIALLY FLOODED UNIFORM LATTICES OF RODS WITH U(10%)O ₂ FUEL	Kurchatov Institute	None used due to hex arrays.
24	WATER-MODERATED SQUARE-PITCHED UNIFORM LATTICES OF RODS WITH U(10%)O ₂ FUEL	Kurchatov Institute	Did not use either case due to 10 wt% U-235 enrichment
25	WATER-MODERATED HEXAGONALLY PITCHED LATTICES OF U(7.5%)O ₂ STAINLESS-STEEL-CLAD FUEL RODS	Kurchatov Institute	None used due to hex arrays.
26	WATER-MODERATED U(4.92)O ₂ FUEL RODS IN 1.29, 1.09, AND 1.01 CM PITCH HEXAGONAL LATTICES AT DIFFERENT TEMPERATURES	IPPE	None used due to hex arrays.
27	WATER-MODERATED AND LEAD-REFLECTED 4.738% ENRICHED URANIUM DIOXIDE ROD ARRAYS	Valduc	None used due to lead reflector.
28	WATER-MODERATED U(4.31)O ₂ FUEL RODS IN TRIANGULAR LATTICES WITH BORON, CADMIUM AND GADOLINIUM AS SOLUBLE POISONS	PNL	None used due to hex arrays.
29	WATER MODERATED AND WATER REFLECTED 4.74% ENRICHED URANIUM DIOXIDE ROD ARRAYS SURROUNDED BY HAFNIUM PLATES	Valduc	None used. No SCALE sample decks. hf plates cases without hf have the same pitch and pin as benchmark 7 above. No significant additional value
30	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.27-CM PITCH) LATTICES OF LOW-ENRICHED U(3.5 WT.% 235U)O ₂ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used due to hex arrays.
31	WATER-MODERATED HEXAGONALLY PITCHED PARTIALLY FLOODED LATTICES OF U(5%)O ₂ ZIRCONIUM-CLAD FUEL RODS, 0.8-CM PITCH	Kurchatov Institute	None used due to hex arrays.
32	UNIFORM WATER-MODERATED LATTICES OF RODS WITH U(10%)O ₂ FUEL IN RANGE FROM 20°C TO 274°C	Kurchatov Institute	None used due to hex arrays.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 253 of 300

Benchmark Number	Description	Lab	Selected
33	REFLECTED AND UNREFLECTED ASSEMBLIES OF 2 AND 3%-ENRICHED URANIUM FLUORIDE IN PARAFFIN	ORNL	None used. Not UO ₂
34	FOUR 4.738-WT.-%-ENRICHED URANIUM DIOXIDE ROD ASSEMBLIES CONTAINED IN CADMIUM, BORATED STAINLESS STEEL, OR BORAL SQUARE CANISTERS, WATER-MODERATED AND -REFLECTED	Valduc	6 borated plate cases used. Rejected cases with less than 2.5 cm separation due to high uncertainty.
35	CRITICAL ARRAYS OF LOW-ENRICHED UO ₂ FUEL RODS IN WATER WITH SOLUBLE GADOLINIUM OR BORON POISON	JAEA	Used 2 cases. Did not use the case with dissolved Gd. (not like pool).
36	THE VVER EXPERIMENTS: REGULAR AND PERTURBED HEXAGONAL LATTICES OF LOW-ENRICHED UO ₂ FUEL RODS IN LIGHT WATER - Part 2	KFKI	None used due to hex arrays.
37	WATER-MODERATED AND PARTIALLY CONCRETE-REFLECTED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE ROD ARRAYS	Valduc	None used. No SCALE sample decks.
38	WATER-MODERATED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE ROD ARRAYS NEXT TO A BORATED CONCRETE SCREEN	Valduc	None used. No SCALE sample decks. Used a borated concrete reflector (not like pool).
39	INCOMPLETE ARRAYS OF WATER-REFLECTED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL-ROD ARRAYS	Valduc	Used all 17 cases.
40	FOUR 4.738-WT.-%-ENRICHED URANIUM DIOXIDE ROD ASSEMBLIES CONTAINED IN BORATED STAINLESS STEEL OR BORAL SQUARE CANISTERS, WATER MODERATED AND REFLECTED BY LEAD OR STEEL	Valduc	4 cases used. Did not use lead reflector cases.
41	STORAGE ARRAYS OF 3%-ENRICHED LWR ASSEMBLIES: THE CRISTO II EXPERIMENT IN THE EOLE REACTOR	Cadarache	Did not use the 5 cases due to complex geometry and no SCALE sample deck.
42	WATER-MODERATED RECTANGULAR CLUSTERS OF U(2.35)O ₂ FUEL RODS (1.684-CM PITCH) SEPARATED BY STEEL, BORAL, BOROFLEX, CADMIUM, OR COPPER PLATES, WITH STEEL REFLECTING WALLS	PNL	Used 5 cases. Did not use copper cases.
43	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH A HEAVY SS-304 REFLECTOR	IPEN	None used due to Gd rods.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 254 of 300

Benchmark Number	Description	Lab	Selected
44	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH UO_2 , STAINLESS STEEL AND COPPER RODS	IPEN	None used due to copper rods.
45	PLEXIGLAS OR CONCRETE-REFLECTED $U(4.46)_3O_8$ WITH $H/U=0.77$ AND INTERSTITIAL MODERATION	Rocky Flats	None used since not pin geometry.
46	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR CONSIDERING TEMPERATURE VARIATION FROM 14°C TO 85°C	IPEN	Used only for temperature bias. See Reference 9.
47	FUEL TRANSPORT FLASK CRITICAL BENCHMARK EXPERIMENTS WITH LOW-ENRICHED URANIUM DIOXIDE FUEL	Winfrith	None used. 3 complex cases. No SCALE sample decks.
48	LIGHT WATER MODERATED AND REFLECTED LOW-ENRICHED (3 WT.% ^{235}U) URANIUM DIOXIDE ROD LATTICES	Winfrith	All 5 cases used
49	MARACAS PROGRAMME: POLYTHENE-REFLECTED CRITICAL CONFIGURATIONS WITH LOW-ENRICHED AND LOW-MODERATED URANIUM DIOXIDE POWDER, $U(5)O_2$	Valduc	None used. Powder rather than pellets. Not similar to pools.
50	149SM SOLUTION TANK IN THE MIDDLE OF WATER-MODERATED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE ROD ARRAYS	Valduc	7 cases used. Did not use cases with dissolved Sm. This is not typical of pools.
51	CRITICAL EXPERIMENTS SUPPORTING CLOSE PROXIMITY WATER STORAGE OF POWER REACTOR FUEL (PART II - ISOLATING PLATES)	B&W	9 cases used. Did not use cases with the borated Al plates since primary source listed a high uncertainty in the boron content.
52	URANIUM DIOXIDE (4.738-WT.-%-ENRICHED) FUEL ROD ARRAYS MODERATED AND REFLECTED BY GADOLINIUM NITRATE SOLUTION	Valduc	None used due to hex arrays.
53	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.27 CM PITCH) LATTICES OF LOW-ENRICHED $U(4.4 \text{ WT.}\% \text{ } ^{235}U)O_2$ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used due to hex arrays.
54	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH UO_2 , AND UO_2 - Gd_2O_3 RODS	IPEN	None used due to Gd rods.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 255 of 300

Benchmark Number	Description	Lab	Selected
55	LIGHT-WATER MODERATED AND REFLECTED LOW-ENRICHED URANIUM (3 wt.% 235U) DIOXIDE ROD LATTICES	Winfrith	Neither case used. Complex geometry no KENO-V.a sample deck
56	CRITICAL EXPERIMENT WITH BORAX-V BOILING WATER REACTOR TYPE FUEL ASSEMBLIES	INL	None used. No sample SCALE decks. Complex BWR geometry.
57	4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL ROD ARRAYS REFLECTED BY WATER IN A DRY STORAGE CONFIGURATION	Valduc	None used. No sample SCALE decks.
58	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH LARGE VOID IN THE REFLECTOR	IPEN	None used. No sample SCALE decks.
59	Not included in 2014 Handbook		
60	RBMK GRAPHITE REACTOR: UNIFORM CONFIGURATIONS OF U(1.8, 2.0, or 2.4% 235U)O ₂ FUEL ASSEMBLIES, AND CONFIGURATIONS OF U(2.0% 235U)O ₂ ASSEMBLIES WITH EMPTY CHANNELS, WATER COLUMNS, AND BORON OR THORIUM ABSORBERS, WITH OR WITHOUT WATER IN CHANNELS	Kurchatov Institute	None used. RBMK – not typical of LWRs
61	VVER PHYSICS EXPERIMENTS: HEXAGONAL (1.27-CM PITCH) LATTICES OF U(4.4 WT.-% 235U)O ₂ FUEL RODS IN LIGHT WATER, PERTURBED BY BORON, HAFNIUM, OR DYSPROSIUM ABSORBER RODS, OR BY WATER GAP WITH/WITHOUT EMPTY ALUMINIUM TUBES	Kurchatov Institute	None used due to hex arrays.
62	2.6%-ENRICHED UO ₂ RODS IN LIGHT-WATER MODERATOR WITH BORATED STAINLESS STEEL PLATE: SINGLE ARRAYS	JAEA	None used. No SCALE sample decks.
63	LIGHT-WATER MODERATED AND REFLECTED LOW-ENRICHED URANIUM (3 wt.% 235U) DIOXIDE ROD LATTICES WITH DISCRETE POISON-ROD ARRAYS	Winfrith	None used. No SCALE sample decks.
64	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.27 CM PITCH) LATTICES OF LOW-ENRICHED U(2.4 WT.-% 235U)O ₂ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used since hex geometry.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 256 of 300

Benchmark Number	Description	Lab	Selected
65	CRITICAL CONFIGURATIONS OF 2.6%-ENRICHED UO_2 ROD ARRAYS IN LIGHT-WATER MODERATOR WITH BORATED STAINLESS STEEL PLATE: COUPLED ARRAYS	JAEA	None used. No SCALE sample decks.
66	PLEXIGLAS-REFLECTED, CONCRETE-REFLECTED, OR THIN STEEL-REFLECTED $U(4.46)_3O_8$ WITH $H/U=0.77$ AND HEU DRIVERS	Rocky Flats	None used. Not an array of rods.
67	Not included in 2014 Handbook		
68	PLEXIGLAS-REFLECTED, CONCRETE-REFLECTED, OR THIN STEEL-REFLECTED $U(4.48)_3O_8$ WITH $H/U=1.25$ OR $H/U=2.03$ AND HEU DRIVERS	Rocky Flats	None used. Not an array of rods.
69	PLEXIGLAS-REFLECTED $U(4.48)_3O_8$ WITH $H/U=1.25$ OR $H/U=2.03$ AND INTERSTITIAL MODERATION	Rocky Flats	None used. Not an array of rods.
70	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.10-CM PITCH) LATTICES OF LOW-ENRICHED $U(6.5 \text{ WT.}\% \text{ } ^{235}\text{U})O_2$ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used due to hex arrays.
71	LOW MODERATED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL ROD ARRAYS	Valduc	All 4 cases used.
72	UNDER-MODERATED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL ROD ARRAYS REFLECTED BY WATER OR POLYETHYLENE	Valduc	Used 3 cases. Did not use Polyethylene reflector cases.
73	UNDER-MODERATED 4.738-WT.-%-ENRICHED URANIUM DIOXIDE FUEL ROD ARRAYS REFLECTED BY WATER WITH HETEROGENEITIES	Valduc	None used. No SCALE sample decks.
74	MIRTE PROGRAM FOUR 4.738-WT.-%-ENRICHED URANIUM-DIOXIDE FUEL-ROD ARRAYS IN WATER SEPARATED BY A CROSS-SHAPED SCREEN OF TITANIUM (5 MM AND 10 MM THICK)	Valduc	Not used due to Titanium screens
75	VVER PHYSICS EXPERIMENTS: HEXAGONAL (1.10 CM PITCH) LATTICES OF LOW-ENRICHED $U(6.5 \text{ WT.}\% \text{ } ^{235}\text{U})O_2$ FUEL RODS IN LIGHT WATER, PERTURBED BY BORON ABSORBER RODS AND WATER HOLES	Kurchatov Institute	None used due to hex arrays.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 257 of 300

Benchmark Number	Description	Lab	Selected
76	LIGHT WATER MODERATED AND REFLECTED LOW ENRICHED URANIUM (3 WT.% 235U) DIOXIDE ROD LATTICES WITH EX-CORE DETECTOR FEATURE	Winfrith	None used. No KENO Va sample decks.
77	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR	IPEN	Only one case used. Rest of cases same materials with small modification of arrays. Not sufficiently independent.
78	WATER-MODERATED SQUARE-PITCHED U(6.90)O ₂ FUEL ROD LATTICES WITH 0.52 FUEL-TO-WATER VOLUME RATIO (0.855 CM PITCH)	SNL	Not used due to high enrichment.
79	WATER-MODERATED U(4.31)O ₂ FUEL ROD LATTICES CONTAINING RHODIUM FOILS	SNL	None used due to hex arrays..
80	WATER-MODERATED SQUARE-PITCHED U(6.90)O ₂ FUEL ROD LATTICES WITH 0.67 FUEL TO WATER VOLUME RATIO	SNL	Not used due to high enrichment.
81	PWR TYPE UO ₂ FUEL RODS WITH ENRICHMENTS OF 3.5 AND 6.6 WT.% WITH BURNABLE ABSORBER ("OTTO HAHN" NUCLEAR SHIP PROGRAM, SECOND CORE)	ANEX	Single case not use. No sample SCALE deck. Unusual case.
82	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH LOW ENRICHED FUEL AND BURNABLE POISON RODS	IPEN	Used only one case. Rest of cases were not significantly different.
83	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH A BIG CENTRAL VOID	IPEN	Used only one case. Rest of cases were not significantly different.
84	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH A CENTRAL CRUCIFORM ROD	IPEN	Used the single case..
85	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.27 CM PITCH) LATTICES OF LOW-ENRICHED U(6.5 WT.% 235U)O ₂ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used due to hex arrays.

Benchmark Number	Description	Lab	Selected
86	VVER PHYSICS EXPERIMENTS: HEXAGONAL LATTICES (1.275 CM PITCH) OF LOW ENRICHED U(3.6, 4.4 WT.% ^{235}U)O ₂ FUEL ASSEMBLIES IN LIGHT WATER WITH H ₃ BO ₃	NRI	None used due to hex arrays.
87	VVER PHYSICS EXPERIMENTS: HEXAGONAL LATTICES (1.22-CM PITCH) OF LOW-ENRICHED U(3.6, 4.4 WT.% ^{235}U)O ₂ FUEL ASSEMBLIES IN LIGHT WATER WITH VARIABLE FUEL-ASSEMBLY PITCH	NRI	None used due to hex arrays.
88	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH HEAVY REFLECTORS COMPOSED OF CARBON STEEL AND NICKEL	IPEN	Not used due to no same SCALE decks and no significant contribution.
89	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH UO ₂ AND BORATED STAINLESS STEEL PLATES	IPEN	Used only one case. Rest of cases were not significantly different.
90	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH UO ₂ AND STAINLESS STEEL RODS	IPEN	Used only one case. Rest of cases were not significantly different.
91	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH UO ₂ , STAINLESS STEEL AND GD ₂ O ₃ RODS	IPEN	Not used due to Gd rods.
92	CRITICAL LOADING CONFIGURATIONS OF THE IPEN/MB-01 REACTOR WITH SOLUBLE BORON	IPEN	Not used due to sufficient boron cases already and no SCALE sample input.
93	DEUTERIUM CRITICAL ASSEMBLY WITH 1.2% ENRICHED URANIUM VARYING COOLANT VOID FRACTION AND LATTICE PITCH	PNC	Not used since cases use D ₂ O rather than H ₂ O
94	VVER PHYSICS EXPERIMENTS: REGULAR HEXAGONAL (1.10 CM PITCH) TWO-REGION LATTICES OF LOW-ENRICHED U(6.5 AND 4.4 WT.% ^{235}U)O ₂ FUEL RODS IN LIGHT WATER AT DIFFERENT CORE CRITICAL DIMENSIONS	Kurchatov Institute	None used due to hex arrays.

A.3.2 Selection of MOX Critical Experiments

Burned fuel contains a low concentration of plutonium (less than 2 wt%) as well as uranium, and thus is actually Mixed Oxide (MOX) fuel. Most classical MOX experiments have plutonium concentrations at least twice as high as that contained in burned fuel. A series of experiments

were performed in France, Haut Taux de Combustion (HTC) critical experiments, and purchased by the US for domestic use. These experiments model the uranium and plutonium concentration of 4.5 wt % U-235 fuel burned to 37.5 GWd/T [A.4]. This fuel has 1.1 wt% plutonium and 1.57 wt% U-235. All of the HTC critical experiments used the same fuel pins. The criticality of these experiments was controlled by adjusting the critical water height. The fuel pins were used in 156 critical arrangements. The experiments were performed in four phases.

HTC Phase 1 [A.5] consists of 17 cases where the pin pitch was varied from 1.3 cm to 2.3 cm and different quantities of pins were used to change the critical height. An 18th case was done where the array was moved to the edge of the tank, so the boundary was the steel tank followed by void. This condition is not typical of a spent fuel pool, so this case was not analyzed. HTC Phase 2 [A.6] consisted of 20 cases where gadolinium of various concentrations was dissolved in the water (Phase 2a) and 21 cases where boron was dissolved in the water (Phase 2b). Since Gadolinium is not credited except as a fission product, Phase 2a cases are not selected for analysis. Phase 3 [A.7] consists of 26 experiments where the pins were arranged as 4 assemblies. Each assembly used a 1.6 cm pin pitch. The assembly separation was varied, as well as the number of pins in each assembly. Eleven cases boxed the assemblies with an absorber (borated steel, Boral, or cadmium). All boxed cases are selected including the cadmium boxed assembly cases, as Cadmium is credited as part of the control rod credit. Finally, Phase 4 [A.8] consisted of redoing the same type of experiments as Phase 3, except with lead and steel reflector screens. All the cases with steel reflectors were selected. The lead reflector cases were not used. In review, a total of 97 HTC critical experiments are included.

Since the burnup requirements may exceed 37.5 GWd/MTU, and so that the MOX case set may include benchmarks from more than the HTC experiments alone, MOX benchmarks from the OECD/NEA handbook [A.3] were reviewed for inclusion. There are only 63 low enriched MOX pin critical experiments documented in the OECD/NEA handbook. Since spent nuclear fuel never reaches greater than 2 wt% Pu, 24 cases with above 2 wt% Pu are not used in the analysis. Thus only 39 non-HTC MOX critical experiments are used. The total MOX set is 39 (OECD/NEA)+97 (HTC) or 136 critical experiments. The 24 MOX cases with greater than 2 wt% Pu were analyzed but are only used to show the dependence of the bias on plutonium content.

A.4. Modeling and Calculating k of the Critical Experiments

For most cases, input decks exist on the OECD/NEA handbook [A.3] disc. In general, these input decks were used with minor modifications. For example, none of the decks were for SCALE 6.0 or the ENDF/B-VII library, and the number of neutrons per generation and the number of generations were, in general, too low. All of the decks were modified to 6000 neutrons per generation and 1500 generations. This was sufficient to make the Monte Carlo uncertainty to be 0.0002 or about one tenth the experimental uncertainty. It was confirmed that the input decks matched the isotopic content given in the handbook. The geometric modeling in the decks also matched the descriptions in the handbook. In short, although there was considerable help by starting with the input files given in the handbook, the ownership of the files was taken, as required by NUREG/CR-6698 [A.2] and as stated in section 2.3:

For specific critical experiments, the facility or site may choose to use input files generated elsewhere to expedite the validation process. The site has the responsibility for ensuring that input files and the options selected are appropriate for use. Regardless of the source of the input file, the site must have reviewed the description of each critical experiment and determined that the representation of the experiment, including simplifying assumptions and options, are consistent with the intended use. In other words, the site must assume ownership of the input file.

KENO case k convergence was verified using two techniques; checking for satisfaction of the chi-squared test in the output, and performing a statistical test that compares the average k of the first half of generations with the average k of the second half of generations within their respective uncertainties. This second technique is considered equivalent of viewing the plotted output and looking for a variation or trend which would indicate a lack of convergence. If either of these tests failed and the output looked suspect, then the cases were rerun with more generations so that the tests succeeded.

Table A.4.1 shows the results of the analysis of the 232 UO₂ critical experiments, along with parameters that are used to check for trends in the results. The spectral index, the Energy of the Average Lethargy of the neutrons causing Fission (EALF) is a calculated value from the SCALE output. Note that some of the critical experiments were actually slightly supercritical. For the supercritical experiments the calculated k's were divided by the measured k before being placed on Table A.4.1.

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 261 of 300

Table A.4.1: UO2 Critical Experiment Results with SCALE 6.0 and ENDF/B-VII

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (delta k)	Calculated k_{eff}
LCT-1	1	2.350	1.270	2.032	0.0960	0.003	0.9981
	2	2.350	1.270	2.032	0.0955	0.003	0.9980
	3	2.350	1.270	2.032	0.0945	0.003	0.9972
	4	2.350	1.270	2.032	0.0952	0.003	0.9976
	5	2.350	1.270	2.032	0.0939	0.003	0.9958
	6	2.350	1.270	2.032	0.0947	0.003	0.9980
	7	2.350	1.270	2.032	0.0931	0.0031	0.9977
	8	2.350	1.270	2.032	0.0941	0.003	0.9966
LCT-2	1	4.310	1.415	2.540	0.1129	0.002	0.9974
	2	4.310	1.415	2.540	0.1128	0.002	0.9994
	3	4.310	1.415	2.540	0.1128	0.002	0.9982
	4	4.310	1.415	2.540	0.1117	0.0018	0.9978
	5	4.310	1.415	2.540	0.1101	0.0019	0.9963
LCT-6	1	2.596	1.417	1.849	0.2351	0.002	0.9977
	2	2.596	1.417	1.849	0.2420	0.002	0.9981
	3	2.596	1.417	1.849	0.2484	0.002	0.9985
	4	2.596	1.417	1.956	0.1812	0.002	0.9983
	5	2.596	1.417	1.956	0.1866	0.002	0.9979
	6	2.596	1.417	1.956	0.1913	0.002	0.9992
	7	2.596	1.417	1.956	0.1963	0.002	0.9988
	8	2.596	1.417	1.956	0.2018	0.002	0.9990
	9	2.596	1.417	2.150	0.1352	0.002	0.9987
	10	2.596	1.417	2.150	0.1388	0.002	0.9983
	11	2.596	1.417	2.150	0.1421	0.002	0.9985
	12	2.596	1.417	2.150	0.1456	0.002	0.9979
	13	2.596	1.417	2.150	0.1486	0.002	0.9986
	14	2.596	1.417	2.293	0.1142	0.002	0.9991
	15	2.596	1.417	2.293	0.1171	0.002	0.9991
	16	2.596	1.417	2.293	0.1196	0.002	0.9991
	17	2.596	1.417	2.293	0.1223	0.002	0.9990
	18	2.596	1.417	2.293	0.1249	0.002	0.9987
LCT-7	1	4.738	0.940	1.260	0.2406	0.0014	0.9958
	2	4.738	0.940	1.600	0.1089	0.0008	0.9986
	3	4.738	0.940	2.100	0.0707	0.0007	0.9976
	4	4.738	0.940	2.520	0.0605	0.0008	0.9980
LCT-8	1	2.459	1.206	1.636	0.2780	0.0012	0.9965
	2	2.459	1.206	1.636	0.2452	0.0012	0.9969
	3	2.459	1.206	1.636	0.2450	0.0012	0.9974
	4	2.459	1.206	1.636	0.2458	0.0012	0.9969
	5	2.459	1.206	1.636	0.2454	0.0012	0.9962
	6	2.459	1.206	1.636	0.2445	0.0012	0.9966
	7	2.459	1.206	1.636	0.2445	0.0012	0.9965
	8	2.459	1.206	1.636	0.2426	0.0012	0.9960

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 262 of 300

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (delta k)	Calculated k_{eff}
	9	2.459	1.206	1.636	0.2419	0.0012	0.9964
	10	2.459	1.206	1.636	0.2481	0.0012	0.9966
	11	2.459	1.206	1.636	0.2534	0.0012	0.9970
	12	2.459	1.206	1.636	0.2470	0.0012	0.9970
	13	2.459	1.206	1.636	0.2474	0.0012	0.9969
	14	2.459	1.206	1.636	0.2492	0.0012	0.9967
	15	2.459	1.206	1.636	0.2496	0.0012	0.9962
	16	2.459	1.206	1.636	0.2272	0.0012	0.9973
	17	2.459	1.206	1.636	0.1982	0.0012	0.9963
LCT-9	1	4.310	1.415	2.540	0.1126	0.0021	0.9983
	2	4.310	1.415	2.540	0.1119	0.0021	0.9978
	3	4.310	1.415	2.540	0.1123	0.0021	0.9980
	4	4.310	1.415	2.540	0.1121	0.0021	0.9982
	5	4.310	1.415	2.540	0.1135	0.0021	0.9987
	6	4.310	1.415	2.540	0.1125	0.0021	0.9981
	7	4.310	1.415	2.540	0.1136	0.0021	0.9993
	8	4.310	1.415	2.540	0.1126	0.0021	0.9985
	9	4.310	1.415	2.540	0.1134	0.0021	0.9986
	16	4.310	1.415	2.540	0.1135	0.0021	0.9981
	17	4.310	1.415	2.540	0.1128	0.0021	0.9986
	18	4.310	1.415	2.540	0.1136	0.0021	0.9981
	19	4.310	1.415	2.540	0.1129	0.0021	0.9991
	20	4.310	1.415	2.540	0.1136	0.0021	0.9979
	21	4.310	1.415	2.540	0.1128	0.0021	0.9988
	22	4.310	1.415	2.540	0.1136	0.0021	0.9989
	23	4.310	1.415	2.540	0.1128	0.0021	0.9989
	24	4.310	1.415	2.540	0.1120	0.0021	0.9982
	25	4.310	1.415	2.540	0.1118	0.0021	0.9984
	26	4.310	1.415	2.540	0.1119	0.0021	0.9986
LCT-10	27	4.310	1.415	2.540	0.1117	0.0021	0.9982
	5	4.310	1.415	2.540	0.3478	0.0021	0.9995
	6	4.310	1.415	2.540	0.2567	0.0021	1.0001
	7	4.310	1.415	2.540	0.2058	0.0021	1.0003
	8	4.310	1.415	2.540	0.1819	0.0021	0.9972
	9	4.310	1.415	2.540	0.1219	0.0021	1.0008
	10	4.310	1.415	2.540	0.1179	0.0021	1.0004
	11	4.310	1.415	2.540	0.1152	0.0021	1.0009
	12	4.310	1.415	2.540	0.1121	0.0021	0.9992
	13	4.310	1.415	2.540	0.1104	0.0021	0.9971
	14	4.310	1.415	1.892	0.3064	0.0028	1.0008
	15	4.310	1.415	1.892	0.2941	0.0028	1.0014
	16	4.310	1.415	1.892	0.2845	0.0028	1.0022
	17	4.310	1.415	1.892	0.2786	0.0028	1.0013
	18	4.310	1.415	1.892	0.2736	0.0028	1.0016

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 263 of 300

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (delta k)	Calculated k _{eff}
	19	4.310	1.415	1.892	0.2668	0.0028	1.0007
	24	4.310	1.415	1.892	0.5905	0.0028	0.9988
	25	4.310	1.415	1.892	0.5448	0.0028	1.0001
	26	4.310	1.415	1.892	0.5056	0.0028	1.0007
	27	4.310	1.415	1.892	0.4722	0.0028	1.0010
	28	4.310	1.415	1.892	0.4419	0.0028	1.0016
	29	4.310	1.415	1.892	0.4177	0.0028	1.0015
	30	4.310	1.415	1.892	0.3642	0.0028	0.9987
LCT-11	1	2.459	1.206	1.636	0.1677	0.0018	0.9970
	2	2.459	1.206	1.636	0.2436	0.0032	0.9963
	3	2.459	1.206	1.636	0.1915	0.0032	0.9967
	4	2.459	1.206	1.636	0.1917	0.0032	0.9974
	5	2.459	1.206	1.636	0.1926	0.0032	0.9967
	6	2.459	1.206	1.636	0.1937	0.0032	0.9968
	7	2.459	1.206	1.636	0.1949	0.0032	0.9975
	8	2.459	1.206	1.636	0.1962	0.0032	0.9968
	9	2.459	1.206	1.636	0.1971	0.0032	0.9969
	10	2.459	1.206	1.636	0.1854	0.0017	0.9942
	11	2.459	1.206	1.636	0.1622	0.0017	0.9942
	12	2.459	1.206	1.636	0.1664	0.0017	0.9939
	13	2.459	1.206	1.636	0.1466	0.0017	0.9945
	14	2.459	1.206	1.636	0.1498	0.0017	0.9949
	15	2.459	1.206	1.636	0.1382	0.0018	0.9953
LCT-13	1	4.310	1.415	1.892	0.2836	0.0018	1.0000
	2	4.310	1.415	1.892	0.2935	0.0018	1.0001
	3	4.310	1.415	1.892	0.2969	0.0018	0.9998
	4	4.310	1.415	1.892	0.2958	0.0018	1.0014
	5	4.310	1.415	1.892	0.2953	0.0032	1.0001
LCT-16	1	2.350	1.270	2.032	0.0951	0.0031	0.9973
	2	2.350	1.270	2.032	0.0948	0.0031	0.9958
	3	2.350	1.270	2.032	0.0947	0.0031	0.9974
	4	2.350	1.270	2.032	0.0949	0.0031	0.9964
	5	2.350	1.270	2.032	0.0945	0.0031	0.9966
	6	2.350	1.270	2.032	0.0955	0.0031	0.9970
	7	2.350	1.270	2.032	0.0953	0.0031	0.9970
	8	2.350	1.270	2.032	0.0966	0.0031	0.9974
	9	2.350	1.270	2.032	0.0958	0.0031	0.9978
	10	2.350	1.270	2.032	0.0966	0.0031	0.9968
	11	2.350	1.270	2.032	0.0959	0.0031	0.9979
	12	2.350	1.270	2.032	0.0971	0.0031	0.9975
	13	2.350	1.270	2.032	0.0961	0.0031	0.9977
	14	2.350	1.270	2.032	0.0971	0.0031	0.9976
	21	2.350	1.270	2.032	0.0967	0.0031	0.9978
	22	2.350	1.270	2.032	0.0964	0.0031	0.9976

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 264 of 300

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (delta k)	Calculated k_{eff}
	23	2.350	1.270	2.032	0.0960	0.0031	0.9974
	24	2.350	1.270	2.032	0.0964	0.0031	0.9974
	25	2.350	1.270	2.032	0.0960	0.0031	0.9977
	26	2.350	1.270	2.032	0.0965	0.0031	0.9976
	27	2.350	1.270	2.032	0.0959	0.0031	0.9975
	28	2.350	1.270	2.032	0.0943	0.0031	0.9977
	29	2.350	1.270	2.032	0.0942	0.0031	0.9966
	30	2.350	1.270	2.032	0.0943	0.0031	0.9967
	31	2.350	1.270	2.032	0.0943	0.0031	0.9978
	32	2.350	1.270	2.032	0.0942	0.0031	0.9970
LCT-17	4	2.350	1.270	2.032	0.1979	0.0031	0.9979
	5	2.350	1.270	2.032	0.1749	0.0031	0.9991
	6	2.350	1.270	2.032	0.1652	0.0031	0.9996
	7	2.350	1.270	2.032	0.1575	0.0031	0.9990
	8	2.350	1.270	2.032	0.1316	0.0031	0.9973
	9	2.350	1.270	2.032	0.1084	0.0031	0.9970
	10	2.350	1.270	2.032	0.0993	0.0031	0.9980
	11	2.350	1.270	2.032	0.0975	0.0031	0.9979
	12	2.350	1.270	2.032	0.0963	0.0031	0.9977
	13	2.350	1.270	2.032	0.0950	0.0031	0.9975
	14	2.350	1.270	2.032	0.0942	0.0031	0.9983
	15	2.350	1.270	1.684	0.1763	0.0028	0.9974
	16	2.350	1.270	1.684	0.1705	0.0028	0.9974
	17	2.350	1.270	1.684	0.1656	0.0028	0.9991
	18	2.350	1.270	1.684	0.1640	0.0028	0.9972
	19	2.350	1.270	1.684	0.1615	0.0028	0.9972
	20	2.350	1.270	1.684	0.1600	0.0028	0.9964
	21	2.350	1.270	1.684	0.1587	0.0028	0.9969
	22	2.350	1.270	1.684	0.1575	0.0028	0.9956
	26	2.350	1.270	1.684	0.3652	0.0028	0.9950
	27	2.350	1.270	1.684	0.3144	0.0028	0.9971
	28	2.350	1.270	1.684	0.2748	0.0028	0.9979
	29	2.350	1.270	1.684	0.2463	0.0028	0.9981
LCT-34	4	4.740	0.940	1.600	0.1363	0.0039	1.0010
	5	4.740	0.940	1.600	0.1329	0.0039	0.9995
	6	4.740	0.940	1.600	0.1297	0.0039	1.0012
	7	4.740	0.940	1.600	0.1278	0.0039	0.9993
	8	4.740	0.940	1.600	0.1256	0.0039	0.9992
	15	4.740	0.940	1.600	0.1348	0.0043	0.9945
LCT-35	1	2.596	1.417	1.956	0.2073	0.0018	0.9981
	2	2.596	1.417	1.956	0.2111	0.0019	0.9971
LCT-39	1	4.738	0.940	1.260	0.2216	0.0014	0.9953
	2	4.738	0.940	1.260	0.2112	0.0014	0.9968
	3	4.738	0.940	1.260	0.1920	0.0014	0.9964

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 265 of 300

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (delta k)	Calculated k_{eff}
	4	4.738	0.940	1.260	0.1834	0.0014	0.9955
	5	4.738	0.940	1.260	0.1391	0.0009	0.9980
	6	4.738	0.940	1.260	0.1452	0.0009	0.9979
	7	4.738	0.940	1.260	0.2124	0.0012	0.9964
	8	4.738	0.940	1.260	0.2026	0.0012	0.9958
	9	4.738	0.940	1.260	0.1970	0.0012	0.9967
	10	4.738	0.940	1.260	0.1727	0.0012	0.9973
	11	4.738	0.940	1.260	0.2214	0.0013	0.9950
	12	4.738	0.940	1.260	0.2159	0.0013	0.9956
	13	4.738	0.940	1.260	0.2140	0.0013	0.9953
	14	4.738	0.940	1.260	0.2120	0.0013	0.9957
	15	4.738	0.940	1.260	0.2109	0.0013	0.9958
	16	4.738	0.940	1.260	0.2099	0.0013	0.9963
	17	4.738	0.940	1.260	0.2096	0.0013	0.9965
LCT-40	1	4.740	0.940	1.600	0.1426	0.0039	0.9968
	5	4.740	0.940	1.600	0.1375	0.0042	0.9948
	9	4.740	0.940	1.600	0.1469	0.0046	0.9983
	10	4.740	0.940	1.600	0.1416	0.0046	0.9928
LCT-42	1	2.350	1.270	1.684	0.1680	0.0016	0.9972
	2	2.350	1.270	1.684	0.1743	0.0016	0.9970
	3	2.350	1.270	1.684	0.1809	0.0016	0.9978
	4	2.350	1.270	1.684	0.1796	0.0017	0.9982
	5	2.350	1.270	1.684	0.1765	0.0033	0.9983
LCT-48	1	3.005	1.094	1.320	0.6740	0.0025	0.9978
	2	3.005	1.094	1.320	0.6467	0.0025	0.9984
	3	3.005	1.094	1.320	0.6771	0.0025	0.9977
	4	3.005	1.094	1.320	0.6788	0.0025	0.9983
	5	3.005	1.094	1.320	0.6691	0.0025	0.9977
LCT-50	1	4.738	0.940	1.300	0.1992	0.0010	0.9976
	2	4.738	0.940	1.300	0.1906	0.0010	0.9972
	3	4.738	0.940	1.300	0.2072	0.0010	0.9970
	4	4.738	0.940	1.300	0.1976	0.0010	0.9967
	5	4.738	0.940	1.300	0.2218	0.0010	0.9983
	6	4.738	0.940	1.300	0.2134	0.0010	0.9986
	7	4.738	0.940	1.300	0.2094	0.0010	0.9988
LCT-51	1 C10	2.459	1.206	1.636	0.1468	0.0020	0.9960
	2 c11a	2.459	1.206	1.636	0.1953	0.0024	0.9979
	3 c11b	2.459	1.206	1.636	0.1951	0.0024	0.9973
	4 c11c	2.459	1.206	1.636	0.1968	0.0024	0.9970
	5 c11d	2.459	1.206	1.636	0.1974	0.0024	0.9974
	6 c11e	2.459	1.206	1.636	0.1989	0.0024	0.9967
	7 c11f	2.459	1.206	1.636	0.1991	0.0024	0.9972
	8 c11g	2.459	1.206	1.636	0.2000	0.0024	0.9970
	9 c12	2.459	1.206	1.636	0.1660	0.0019	0.9968

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 266 of 300

Benchmark ID	Case No.	Enrichment (wt% U-235)	Fuel Pin Diameter (cm)	Fuel Pin Pitch (cm)	EALF (eV)	Measurement Uncertainty (Δk)	Calculated k_{eff}
LCT-71	1	4.738	0.949	1.100	0.7553	0.00076	0.9943
	2	4.738	0.949	1.100	0.6915	0.00076	0.9945
	3	4.738	0.949	1.100	0.6563	0.00076	0.9943
	4	4.738	0.949	1.075	0.8432	0.0008	0.9938
LCT-72	1	4.738	0.949	1.600	0.1101	0.0012	0.9985
	2	4.738	0.949	1.600	0.1062	0.0012	0.9973
	3	4.738	0.949	1.600	0.1083	0.0012	0.9980
LCT-77	3	4.349	0.980	1.500	0.1618	0.0010	1.0005
LCT-82	3	4.349	0.980	1.500	0.1494	0.0010	1.0005
LCT-83	1	4.349	0.980	1.500	0.1512	0.0010	0.9999
LCT-84	1	4.349	0.980	1.500	0.1541	0.0010	0.9997
LCT-89	1	4.349	0.980	1.500	0.1529	0.0010	1.0000
LCT-90	1	4.349	0.980	1.500	0.1458	0.0010	0.9997

The HTC modeling utilized References A.5 through A.8 for all the details for the analysis. The references include detailed experiment setup materials and geometry, which was used to create detailed SCALE models. Table A.4.2 shows the results of the SCALE calculations of the HTC experiments. The fuel pins for all the HTC cases are the same. The plutonium weight % is always 1.1 wt% Pu.

Table A.4.3 is the results of the SCALE calculations of the MOX critical experiments from the OECD/NEA handbook. [A.3]

Table A.4.2: HTC Critical Experiment Results with SCALE 6.0 and ENDF/B-VII

Experiment	k_{eff}	σ_{calc}	σ_{exp}	EALF (eV)	Pitch (cm)	Absorber
HTC-P1-C01	0.99924	0.00020	0.00182	0.0691	2.3	
HTC-P1-C02	0.99915	0.00019	0.00182	0.0662	2.3	
HTC-P1-C03	0.99929	0.00019	0.00182	0.0661	2.3	
HTC-P1-C04	1.00021	0.00024	0.00182	0.0845	1.9	
HTC-P1-C05	1.00028	0.00023	0.00182	0.0823	1.9	
HTC-P1-C06	0.99974	0.00022	0.00182	0.0817	1.9	
HTC-P1-C07	0.99992	0.00024	0.00182	0.1019	1.7	
HTC-P1-C08	0.99958	0.00024	0.00182	0.1002	1.7	
HTC-P1-C09	0.99917	0.00024	0.00182	0.0993	1.7	
HTC-P1-C10	1.00017	0.00025	0.00182	0.1397	1.5	
HTC-P1-C11	0.99882	0.00023	0.00182	0.1350	1.5	
HTC-P1-C12	0.99864	0.00023	0.00182	0.1331	1.5	
HTC-P1-C13	0.99834	0.00027	0.00182	0.2542	1.3	
HTC-P1-C14	0.99813	0.00024	0.00182	0.2322	1.3	
HTC-P1-C15	0.99766	0.00025	0.00182	0.2286	1.3	
HTC-P1-C16	1.00008	0.00023	0.00182	0.1010	1.7	
HTC-P1-C17	0.99937	0.00021	0.00182	0.0989	1.7	
						Boron ppm
HTC-P2-BOR-C01	0.99878	0.00024	0.00247	0.2451	1.3	100
HTC-P2-BOR-C02	0.99783	0.00026	0.00247	0.2426	1.3	106
HTC-P2-BOR-C03	0.99790	0.00024	0.00247	0.2530	1.3	205
HTC-P2-BOR-C04	0.99880	0.00024	0.00247	0.2612	1.3	299
HTC-P2-BOR-C05	0.99855	0.00022	0.00247	0.2721	1.3	400
HTC-P2-BOR-C06	0.99823	0.00023	0.00247	0.2688	1.3	399
HTC-P2-BOR-C07	0.99934	0.00027	0.00247	0.2776	1.3	486
HTC-P2-BOR-C08	0.99847	0.00022	0.00247	0.2847	1.3	587
HTC-P2-BOR-C09	0.99930	0.00022	0.00247	0.1652	1.5	595
HTC-P2-BOR-C10	0.99789	0.00022	0.00247	0.1600	1.5	499
HTC-P2-BOR-C11	0.99959	0.00023	0.00247	0.1555	1.5	393
HTC-P2-BOR-C12	0.99963	0.00021	0.00247	0.1492	1.5	295
HTC-P2-BOR-C13	0.99893	0.00024	0.00247	0.1445	1.5	200
HTC-P2-BOR-C14	1.00255	0.00026	0.00247	0.1391	1.5	89
HTC-P2-BOR-C15	1.00337	0.00024	0.00247	0.1026	1.7	90
HTC-P2-BOR-C16	1.00162	0.00024	0.00247	0.1066	1.7	194
HTC-P2-BOR-C17	1.00309	0.00021	0.00247	0.1098	1.7	286
HTC-P2-BOR-C18	0.99343	0.00020	0.00247	0.1152	1.7	415
HTC-P2-BOR-C19	1.00041	0.00023	0.00247	0.1041	1.7	100

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 268 of 300

Experiment	k_{eff}	σ_{calc}	σ_{exp}	EALF (eV)	Pitch (cm)	Absorber
HTC-P2-BOR-C20	0.99279	0.00020	0.00247	0.0892	1.9	220
HTC-P2-BOR-C21	0.99689	0.00026	0.00247	0.0857	1.9	110
HTC-P3-C1	0.99816	0.00019	0.00322	0.1230	1.6	Borated SS
HTC-P3-C2	0.99992	0.00018	0.00355	0.1399	1.6	Borated SS
HTC-P3-C3	0.99713	0.00019	0.00322	0.1285	1.6	Borated SS
HTC-P3-C4	0.99721	0.00019	0.00322	0.1232	1.6	Borated SS
HTC-P3-C5	0.99697	0.00019	0.00322	0.1340	1.6	Borated SS
HTC-P3-C6	1.00014	0.00018	0.00489	0.1288	1.6	Boral
HTC-P3-C7	0.99594	0.00018	0.00322	0.1278	1.6	Cd
HTC-P3-C8	1.00361	0.00017	0.00322	0.1381	1.6	Cd
HTC-P3-C9	0.99672	0.00019	0.00322	0.1325	1.6	Cd
HTC-P3-C10	0.99653	0.00021	0.00322	0.1288	1.6	Cd
HTC-P3-C11	0.99619	0.00019	0.00322	0.1364	1.6	Cd
HTC-P3-C12	0.99965	0.00024	0.00254	0.1120	1.6	
HTC-P3-C13	0.99956	0.00028	0.00254	0.1110	1.6	
HTC-P3-C14	0.99990	0.00023	0.00254	0.1111	1.6	
HTC-P3-C15	0.99938	0.00017	0.00254	0.1103	1.6	
HTC-P3-C16	0.99949	0.00027	0.00254	0.1098	1.6	
HTC-P3-C17	0.99991	0.00023	0.00254	0.1079	1.6	
HTC-P3-C18	0.99955	0.00023	0.00254	0.1060	1.6	
HTC-P3-C19	1.00008	0.00022	0.00254	0.1036	1.6	
HTC-P3-C20	0.99967	0.00023	0.00254	0.1016	1.6	
HTC-P3-C21	1.00014	0.00022	0.00254	0.1041	1.6	
HTC-P3-C22	1.00056	0.00023	0.00254	0.1065	1.6	
HTC-P3-C23	0.99996	0.00023	0.00254	0.1141	1.6	
HTC-P3-C24	0.99981	0.00025	0.00254	0.1497	1.6	
HTC-P3-C25	0.99952	0.00023	0.00254	0.1261	1.6	
HTC-P3-C26	0.99922	0.00026	0.00254	0.1148	1.6	
HTC-P4-ST-C1	1.00334	0.00018	0.00499	0.1519	1.6	Borated SS
HTC-P4-ST-C2	0.99844	0.00018	0.00305	0.1496	1.6	Borated SS
HTC-P4-ST-C3	0.99817	0.00019	0.00305	0.1457	1.6	Borated SS
HTC-P4-ST-C4	0.99799	0.00020	0.00305	0.1416	1.6	Borated SS
HTC-P4-ST-C5	0.99806	0.00019	0.00318	0.1386	1.6	Borated SS
HTC-P4-ST-C6	0.99732	0.00018	0.00318	0.1373	1.6	Borated SS
HTC-P4-ST-C7	0.99677	0.00018	0.00318	0.1360	1.6	Borated SS
HTC-P4-ST-C8	0.99674	0.00019	0.00232	0.1351	1.6	Borated SS
HTC-P4-ST-C9	0.99634	0.00021	0.00232	0.1341	1.6	Borated SS
HTC-P4-ST-C10	0.99813	0.00020	0.00232	0.1336	1.6	Borated SS

Criticality Safety Evaluation Report

Serial No. 18-039
Docket No. 50-423
Attachment 6, Page 269 of 300

Experiment	k_{eff}	σ_{calc}	σ_{exp}	EALF (eV)	Pitch (cm)	Absorber
HTC-P4-ST-C11	0.99846	0.00018	0.00232	0.1315	1.6	Borated SS
HTC-P4-ST-C12	1.00095	0.00018	0.00511	0.1347	1.6	Boral
HTC-P4-ST-C13	0.99674	0.00020	0.00511	0.1330	1.6	Boral
HTC-P4-ST-C14	1.00411	0.00017	0.00616	0.1486	1.6	Cd
HTC-P4-ST-C15	0.99846	0.00020	0.00406	0.1423	1.6	Cd
HTC-P4-ST-C16	0.99788	0.00018	0.00190	0.1358	1.6	Cd
HTC-P4-ST-C17	0.99585	0.00018	0.00190	0.1349	1.6	Cd
HTC-P4-ST-C18	0.99561	0.00019	0.00190	0.1335	1.6	Cd
HTC-P4-ST-C19	0.99517	0.00018	0.00190	0.1324	1.6	Cd
HTC-P4-ST-C20	0.99438	0.00019	0.00190	0.1316	1.6	Cd
HTC-P4-ST-C21	0.99887	0.00018	0.00237	0.1330	1.6	Cd
HTC-P4-ST-C22	1.00070	0.00023	0.00432	0.1724	1.6	
HTC-P4-ST-C23	1.00096	0.00024	0.00432	0.1650	1.6	
HTC-P4-ST-C24	0.99964	0.00024	0.00470	0.1573	1.6	
HTC-P4-ST-C25	0.99960	0.00021	0.00470	0.1557	1.6	
HTC-P4-ST-C26	0.99967	0.00017	0.00470	0.1543	1.6	
HTC-P4-ST-C27	0.99917	0.00024	0.00470	0.1533	1.6	
HTC-P4-ST-C28	0.99909	0.00025	0.00470	0.1523	1.6	
HTC-P4-ST-C29	0.99914	0.00023	0.00470	0.1431	1.6	
HTC-P4-ST-C30	0.99993	0.00023	0.00090	0.1335	1.6	
HTC-P4-ST-C31	0.99925	0.00021	0.00090	0.1278	1.6	
HTC-P4-ST-C32	0.99990	0.00028	0.00090	0.1244	1.6	
HTC-P4-ST-C33	0.99961	0.00022	0.00090	0.1224	1.6	

Table A.4.3: Results of Low Enriched MOX Critical Experiments Calculated with SCALE

Case ID	Reference	k_{eff}	sigma	EALF (eV)	Pu wt%	Pu 240 wt%	Am241/U238
093array	OECD-7	1.00201	0.00024	0.189	2	16	6.816E-05
105a1	OECD-7	0.99513	0.00025	0.136	2	16	7.554E-05
105array	OECD-7	0.99680	0.00026	0.137	2	16	7.554E-05
105b1	OECD-7	0.99202	0.00027	0.137	2	16	7.554E-05
105b2	OECD-7	0.99304	0.00023	0.137	2	16	7.554E-05
105b3	OECD-7	0.99359	0.00024	0.137	2	16	7.554E-05
105b4	OECD-7	0.99450	0.00024	0.136	2	16	7.554E-05
1143arra	OECD-7	0.99785	0.00024	0.116	2	16	8.134E-05
132array	OECD-7	0.99689	0.00022	0.095	2	16	8.134E-05
1386arra	OECD-7	0.99511	0.00023	0.090	2	16	6.965E-05
epri70b	OECD-2	0.99893	0.00025	0.712	2	7.8	7.287E-05
epri70un	OECD-2	0.99719	0.00026	0.536	2	7.8	7.287E-05
epri87b	OECD-2	1.00160	0.00022	0.269	2	7.8	7.287E-05
epri87un	OECD-2	0.99862	0.00025	0.184	2	7.8	7.287E-05
epri99b	OECD-2	1.00083	0.00021	0.176	2	7.8	7.287E-05
epri99un	OECD-2	1.00140	0.00028	0.133	2	7.8	7.287E-05
k1mct009	OECD-9	0.99917	0.00026	0.508	1.5	8	1.058E-05
k2mct009	OECD-9	0.99440	0.00024	0.290	1.5	8	9.774E-06
k3mct009	OECD-9	0.99460	0.00024	0.151	1.5	8	8.960E-06
k4mct009	OECD-9	0.99240	0.00024	0.114	1.5	8	8.960E-06
k5mct009	OECD-9	0.99287	0.00022	0.094	1.5	8	8.960E-06
k6mct009	OECD-9	0.99375	0.00024	0.090	1.5	8	9.774E-06
omct61	OECD-6	0.99669	0.00024	0.373	2	8	2.609E-05
omct62	OECD-6	0.99726	0.00024	0.190	2	8	2.271E-05
omct63	OECD-6	0.99684	0.00024	0.137	2	8	2.512E-05
omct64	OECD-6	0.99727	0.00024	0.116	2	8	2.222E-05
omct65	OECD-6	0.99868	0.00022	0.095	2	8	2.271E-05
omct66	OECD-6	0.99708	0.00021	0.090	2	8	2.367E-05
mct8c1	OECD-8	1.00038	0.00024	0.137	2	24	7.931E-05
mct8c2	OECD-8	0.99496	0.00027	0.138	2	24	7.270E-05
mct8c3	OECD-8	0.99457	0.00022	0.137	2	24	8.586E-05
mct8c4	OECD-8	0.99527	0.00024	0.137	2	24	9.882E-05
mct8c5	OECD-8	0.99458	0.00023	0.137	2	24	9.560E-05
mct8c6	OECD-8	0.99833	0.00024	0.372	2	24	7.270E-05
mct8cal	OECD-8	0.99770	0.00024	0.247	2	24	8.586E-05
mct8cb1	OECD-8	1.00459	0.00026	0.171	2	24	8.586E-05
mct8cb3	OECD-8	1.00137	0.00023	0.142	2	24	8.586E-05
mctcb2	OECD-8	1.00384	0.00026	0.106	2	24	8.586E-05
mctcb4	OECD-8	1.00345	0.00023	0.092	2	24	8.586E-05
mixo251k	OECD-5	1.00686	0.00024	0.087	4	18	1.587E-04
mixo252k	OECD-5	0.99585	0.00025	0.354	4	18	1.587E-04
mixo253k	OECD-5	0.99984	0.00026	0.187	4	18	1.587E-04

Case ID	Reference	k_{eff}	sigma	EALF (eV)	Pu wt%	Pu 240 wt%	Am241/U238
mixo254k	OECD-5	0.99546	0.00026	0.137	4	18	1.587E-04
mixo255k	OECD-5	1.00068	0.00024	0.116	4	18	1.587E-04
mixo256k	OECD-5	1.00039	0.00022	0.095	4	18	1.587E-04
mixo257k	OECD-5	0.99846	0.00024	0.090	4	18	1.587E-04
saxtn104o	OECD-3	1.00023	0.00026	0.099	6.6	8.6	0.0
saxtn56bo	OECD-3	0.99971	0.00028	0.612	6.6	8.6	0.0
saxtn735o	OECD-3	0.99919	0.00028	0.182	6.6	8.6	0.0
saxtn792o	OECD-3	1.00015	0.00028	0.150	6.6	8.6	0.0
saxton52o	OECD-3	1.00000	0.00029	0.848	6.6	8.6	0.0
saxton56o	OECD-3	1.00067	0.00029	0.517	6.6	8.6	0.0
tca1	OECD-4	0.99609	0.00026	0.141	3.01	22.02	1.045E-04
tca10	OECD-4	0.99949	0.00024	0.079	3.01	22.02	9.307E-05
tca11	OECD-4	0.99960	0.00025	0.079	3.01	22.02	2.058E-04
tca2	OECD-4	0.99693	0.00023	0.140	3.01	22.02	1.993E-04
tca3	OECD-4	0.99716	0.00025	0.140	3.01	22.02	2.962E-04
tca4	OECD-4	0.99695	0.00027	0.117	3.01	22.02	9.878E-05
tca5	OECD-4	0.99768	0.00025	0.116	3.01	22.02	2.018E-04
tca6	OECD-4	0.99797	0.00026	0.115	3.01	22.02	3.903E-04
tca7	OECD-4	0.99826	0.00024	0.091	3.01	22.02	8.878E-05
tca8	OECD-4	0.99817	0.00023	0.091	3.01	22.02	2.034E-04
tca9	OECD-4	0.99906	0.00023	0.091	3.01	22.02	3.019E-04

A.5. Statistical Analysis of the Data

The statistical treatment used follows the guidance provided in NUREG/CR-6698 [A.2]. The NUREG approach weights the calculated k 's by the experimental uncertainty. This approach means the higher quality experiments (i.e.: lower uncertainty) affect the results more than the low quality experiments. The uncertainty weighting is used for the analysis of the set of experiments as a whole, as well as for the analysis for trends.

Spent fuel goes from having little plutonium to having about 1.5 wt% plutonium at discharge burnups. Since the bias is not the same for plutonium critical experiments as it is for uranium critical experiments, the bias would be expected to be a function of burnup. Rather than attempt to make the bias a function of burnup, analysis of the UO₂ and MOX critical experiments are separated and the most limiting bias and uncertainty from the two sets will be used in the analysis of the spent fuel pool. The fresh fuel storage uses only the UO₂ critical experiments.

Criticality Safety Evaluation Report

Serial No. 18-039

Docket No. 50-423

Attachment 6, Page 272 of 300

The set of MOX experiments is more limited in geometric variation. Because of this, the only trending parameter used for the analysis of the MOX fuel is the spectral index Energy of the Average Lethargy of the neutrons causing Fission, EALF.

A.5.1 Statistical Analysis of the UO2 Critical Experiments

This section follows closely NUREG/CR-6698 so in order to help matching with the NUREG the equation numbers from the NUREG are given in parentheses.

The first step of the analysis is force all the experiments to be critical so the analysis is consistent over the entire set. This is done by converting supercritical experiments to critical experiments using the following equation (9):

$$k_{\text{norm}} = k_{\text{calc}} / k_{\text{exp}}$$

NUREG/CR-6698 recommends weighting the data by its uncertainty. The combined error for each experimented is calculated (3):

$$\sigma_t = \sqrt{\sigma_{\text{calc}}^2 + \sigma_{\text{exp}}^2}$$

The weighted mean keff (6):

$$\bar{k}_{\text{eff}} = \frac{\sum \frac{1}{\sigma_i^2} k_{\text{eff},i}}{\sum \frac{1}{\sigma_i^2}}$$

The bias is calculated as follows (8):

$$\text{Bias} = \bar{k}_{\text{eff}} - 1; \text{ the bias is set to zero if calculated to be greater than zero.}$$

The variance about the mean (4):

$$s^2 = \frac{\frac{1}{n-1} \sum \left\{ \frac{1}{\sigma_i^2} [k_{\text{eff},i} - \bar{k}_{\text{eff}}]^2 \right\}}{\frac{1}{n} \sum \frac{1}{\sigma_i^2}}$$

The average total uncertainty (5):

$$\bar{\sigma}^2 = \frac{n}{\sum \frac{1}{\sigma_i^2}}$$

The square root of the pooled variance (7):

$$S_p = \sqrt{s^2 + \bar{\sigma}^2}$$

The uncertainty is calculated by multiplying the square root of the pooled variance by the one-sided lower tolerance factor. Since all the analysis has a sample size greater than 50, 2.065 is used as the single-sided lower tolerance factor.

The weighted mean is 0.9974 and the weighted standard deviation is 0.0024. The average uncertainty of the experiments (interpreted as one sigma) is 0.0022. Since the total one sigma standard deviation is only 0.0024, this suggests that the experimental uncertainty dominates the uncertainty and there is little to be gained with improved methods. Unless stated otherwise, all the results presented will come from the weighted analysis. The bias of the set as a whole is $1 - 0.9974 = 0.0026$. The uncertainty is the standard deviation multiplied by the single-sided lower tolerance factor (taken as 2.065 from Table 2.1 of Reference 2), so it is **0.0050**.

As recommended by NUREG/CR-6698, the results of the validation are checked for normality. Normality of the UO2 results were tested using the Shapiro-Wilks test via the J. P. Royston algorithm [A.9]. The Shapiro-Wilks test results indicates that the null hypothesis for the UO2 set being normally distributed should be rejected at the $\alpha=0.05$ significance level (i.e. 95% confidence). The p-value for the SW Test statistic is 0.02, which indicate that the SW Test would pass with 98% confidence. While we reject the null hypothesis for the UO2 set at the 95% confidence level, a qualitative visual examination of the histogram (Figure A.5.1) and the Q-Q plot (Figure A.5.2) does not show any obvious indication of non-normal behavior, trend, or skewness. Given no obvious non-normal behavior with the histogram and Q-Q plot, a normal distribution assumption may be acceptable. Notice that the calculated k's are a little closer to the mean and a little higher (conservative) than expected in a normal distribution. This means assuming a normal distribution may be conservative for this data.

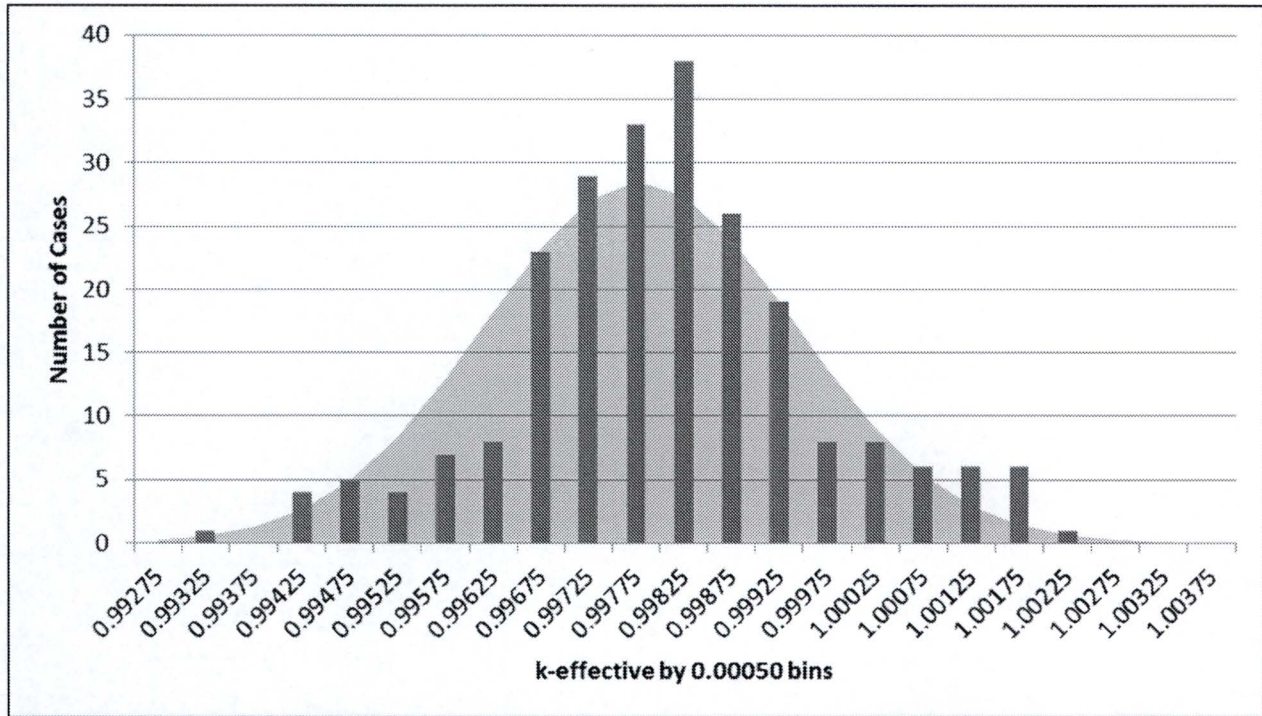


Figure A.5.1: Distribution of the Calculated k 's Around the Mean for All the UO2 Benchmarks

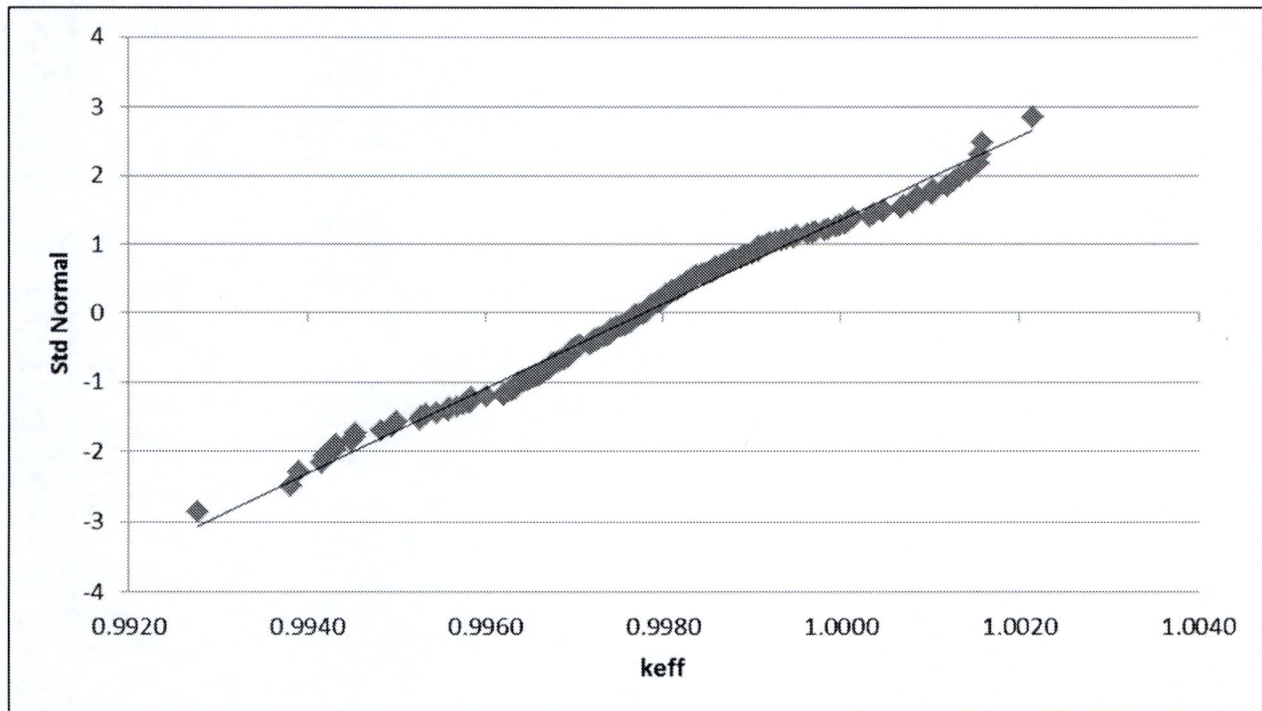


Figure A.5.2: Q-Q Normality Plot for the UO2 Benchmarks

Numerous sources [A.10, A.11, A.12] suggest that for the large sample size used here, normality testing is not important. For example, in the textbook, Statistics for Social Science by R. Mark Sirkin [A.12], it states:

"Law of large numbers. A law that states that if the size of the sample, n, is sufficiently large (no less than 30; preferably no less than 50), then the central limit theory will apply even if the population is not normally distributed along variable x ...

If: Then:

n ≥ 100 It is always safe to relax the normality assumption

50 ≤ n < 100 It is almost always safe

30 ≤ n < 50 It is probably safe."

The analysis in this validation assumes that the techniques used here are sufficiently robust for the limited normality data. In conclusion, although we reject the null hypothesis that the data is normally distributed at the 95% confidence level, there is no indication that the data is not normally distributed, thus it will be treated as normally distributed.

The next step in the analysis is to look for trends in the data. In the past it was assumed that unless there is a high confidence level (95%) that the slope was non-zero, the analysis would assume a zero slope (no trend) on the given parameter. The analysis will include consideration of the data as non-trended, however it is more conservative to assume there is a trend.

Inverting the statistical test to requiring a high confidence that the slope is zero will result in all cases having a trend. At this time, although a test on the confidence of the trend is performed, the analysis assumes all calculated trends are real.

Before presenting the results of the analysis the following provides the equations used for the analysis. For these equations parameter y is for the dependent variable (k_{eff}), and parameter x is for the independent variables (e.g., enrichment, EALF).

First, the linear equation for the fit(10):

$$Y(x) = a + bx$$

The coefficients are calculated using the next three equations (11, 12, 13):

$$a = \frac{1}{\Delta} \left[\sum \frac{x_i^2}{\sigma_i^2} \sum \frac{y_i^2}{\sigma_i^2} - \sum \frac{x_i}{\sigma_i^2} \sum \frac{x_i y_i}{\sigma_i^2} \right]$$

$$b = \frac{1}{\Delta} \left[\sum \frac{1}{\sigma_i^2} \sum \frac{x_i y_i}{\sigma_i^2} - \sum \frac{x_i}{\sigma_i^2} \sum \frac{y_i}{\sigma_i^2} \right]$$

$$\Delta = \sum \frac{1}{\sigma_i^2} \sum \frac{x_i^2}{\sigma_i^2} - \left[\sum \frac{x_i}{\sigma_i^2} \right]^2$$

The weighted mean for the independent parameter (15):

$$\bar{x} = \frac{\sum \frac{1}{\sigma_i^2} x_i}{\sum \frac{1}{\sigma_i^2}}$$

The bias is calculated as follows (23):

$$Bias = k_{fit}(x) - 1; \text{ the bias is set to zero if calculated to be greater than zero.}$$

Finally, the uncertainty is computed from (23):

$$S_{Pfit} \left\{ \sqrt{2F_a^{(2,n-2)} \left[\frac{1}{n} + \frac{(x - \bar{x})^2}{\sum (x_i - \bar{x})^2} \right]} + z_{2p-1} \sqrt{\frac{(n-2)}{\chi_{1-\gamma, n-2}^2}} \right\}$$

where p = desired confidence (0.95) and the remaining parameters are computed as follows (25, 30, 28):

$$\gamma = \frac{1-p}{2}$$

$$s_{fit}^2 = \frac{\frac{1}{n-2} \sum \left\{ \frac{1}{\sigma_i^2} [k_{eff,i} - k_{fit}(x_i)]^2 \right\}}{\frac{1}{n} \sum \frac{1}{\sigma_i^2}}$$

$$S_{Pfit} = \sqrt{s_{fit}^2 + \bar{\sigma}^2}$$

The width of the tolerance band is a function of the trending parameter. When the value for the independent variable is known, it is used in the calculation of the uncertainty. For simplicity sometimes the maximum width of the tolerance band over the range of data is taken as the uncertainty.

In the final analysis, the calculated k of the system must be less than the minimum of k(x) minus the uncertainty minus the administrative safety margin. The uncertainty in k from other independent uncertainties, such as the manufacturing tolerances, burnup, and depletion uncertainties can be statistically combined with the uncertainty in the criticality validation. Now

this section will evaluate the trends in k as a function of trending parameters using the methods described above.

Neutron spectrum

Trends in the calculated k of the benchmarks were sought as a function of the neutron spectrum. Since a large number of things can affect the spectrum, a single index calculated by SCALE is used. This index is the Energy (eV) of the Average Lethargy causing Fission (EALF). Figure A.5.3 shows the distribution of k's around the mean k, which is shown as the red line. Visual inspection of the graph and the statistical analysis of the results of the statistical analysis suggest that there is a statistically significant trend on neutron spectrum. Using NUREG/CR-6698 [A.2] equations from above and the data from Table A.4.1, the predicted mean k as a function of EALF is:

$$k(\text{EALF}) = 0.99853 - 0.00491 * \text{EALF}$$

The units for EALF are eV. The uncertainty (in terms of k) about the trend is a function of EALF and is shown on Table A.5-1.

Table A.5.1: Bias and Uncertainty Based on the EALF Trend from UO2 Critical Experiments

Fresh Fuel		
Maximum EALF (ev)	Bias	Uncertainty
0.35	0.0032	0.0047
0.40	0.0034	0.0047
0.45	0.0037	0.0047
0.50	0.0039	0.0048
0.55	0.0042	0.0049
0.60	0.0044	0.0050
0.65	0.0047	0.0051
0.70	0.0049	0.0052
0.75	0.0052	0.0053
0.80	0.0054	0.0054
0.85	0.0056	0.0055
0.90	0.0059	0.0057
0.95	0.0061	0.0058
1.00	0.0064	0.0059
1.05	0.0066	0.0060
1.10	0.0069	0.0061
1.15	0.0071	0.0062
1.20	0.0074	0.0063

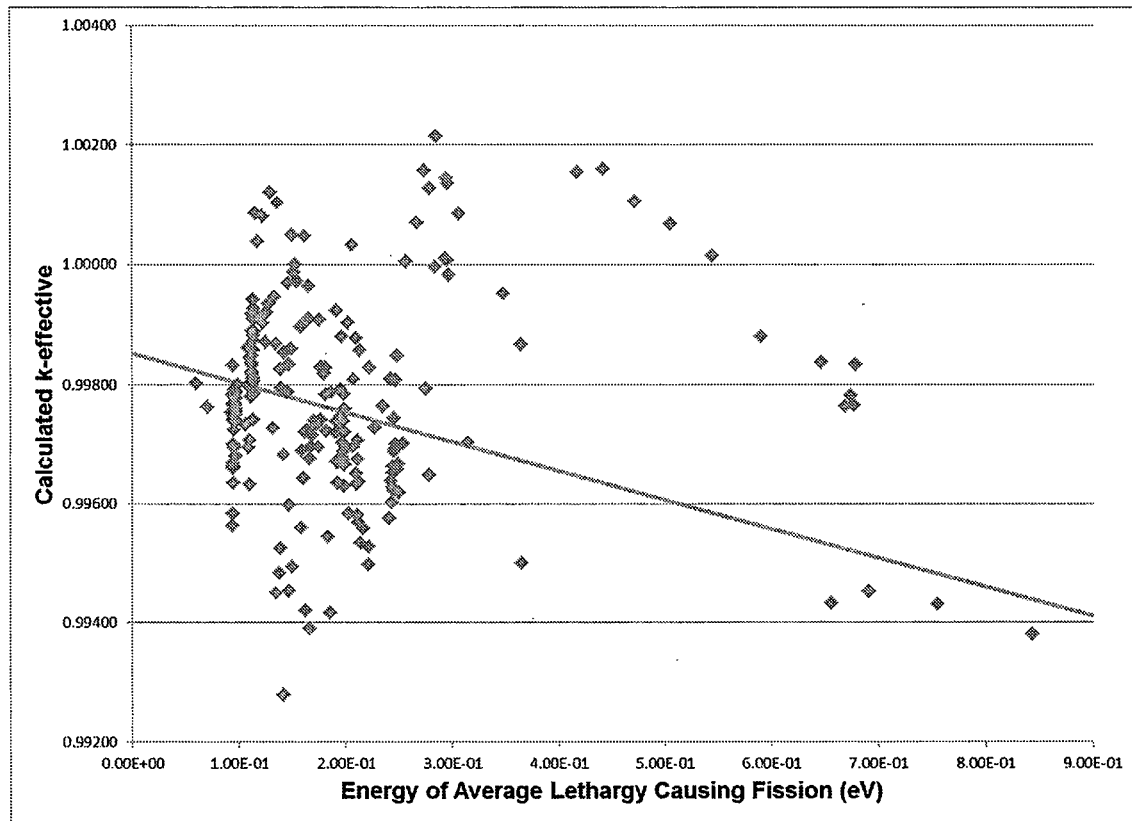


Figure A.5.3: Calculated k for the UO₂ Critical Benchmarks as a Function of EALF Geometry Tests

Two trend tests were performed to determine if lattice/geometric parameters are adequately treated by SCALE 6.0. The first parameter is the fuel pin diameter. A small, statistically significant trend was found when the critical experiment analysis results were correlated to the fuel pin diameter. The second lattice parameter tested is the lattice pitch. A statistically significant trend on lattice pitch was found. The trend on pitch or pin diameter could be caused by the spectral trend found in the previous subsection.

Using NUREG/CR-6698 [A.2] equations from above and the data from Table A.4.1, the predicted mean k as a function of pin diameter is:

$$k(\text{Pin Diameter}) = 0.99421 + (2.82\text{E-}03) \cdot \text{Pin Diameter}$$

where the pin diameter is in cm. The predicted mean k as a function of pitch is:

$$k(\text{Pitch}) = 0.99433 + (1.84\text{E-}03) \cdot \text{Pitch}$$

where lattice pitch is in cm.

The tolerance band widths, using the second term of NUREG/CR-6698 [2] equation 23, for the MPS3 pin diameter (0.95 cm) and the pin pitch (1.26 cm) are 0.0048 and 0.0047 respectively. Figures A.5.4 and A.5.5 graphically present k_{eff} as a function of the pin diameter and the lattice pitch.

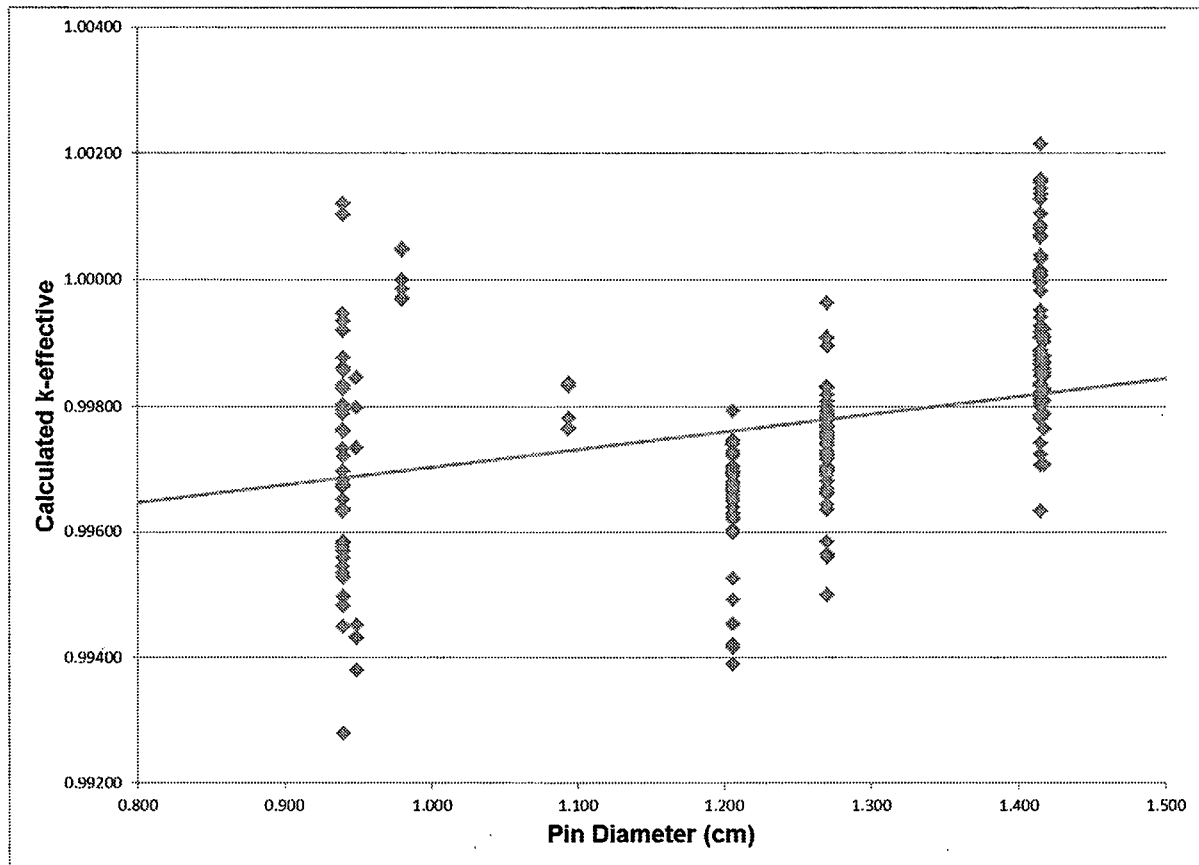


Figure A.5.4: Calculated k for the UO₂ Critical Benchmarks as a Function of Pin Diameter

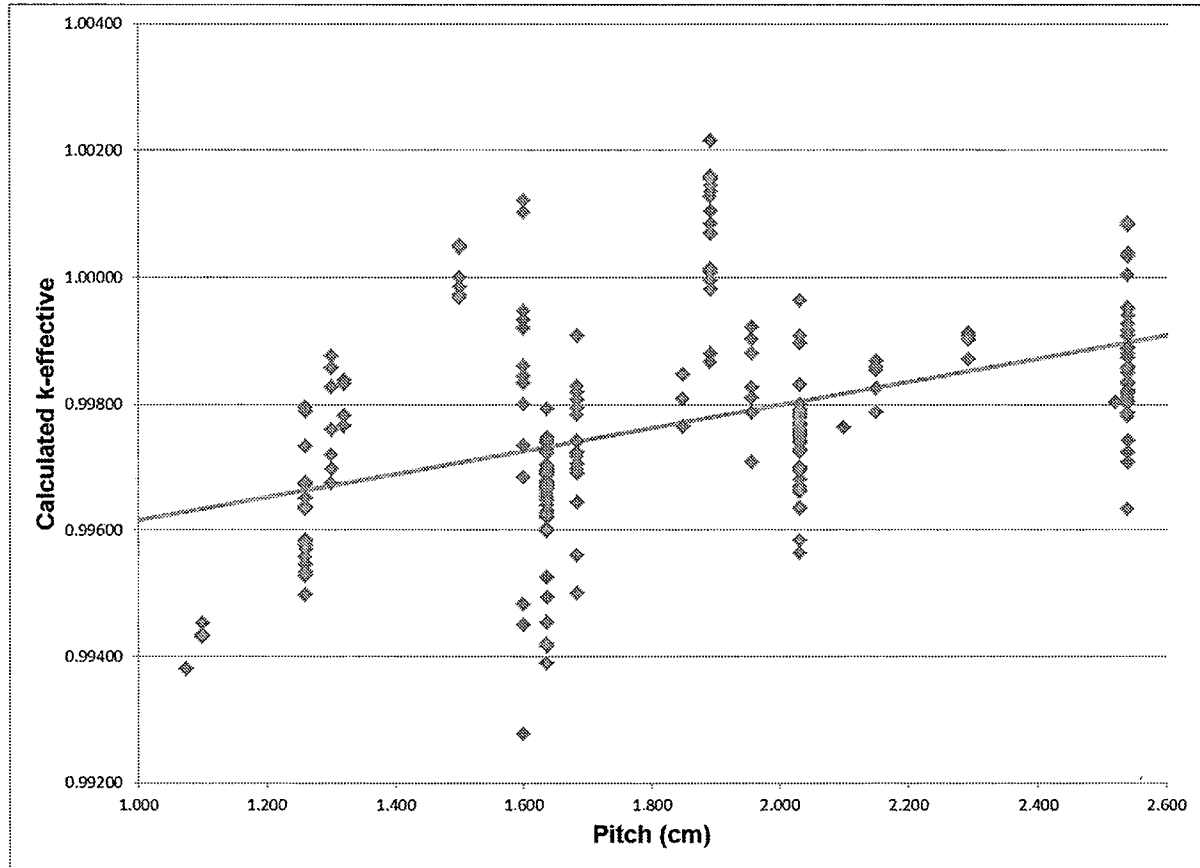


Figure A.5.5: Calculated k for the UO₂ Critical Benchmarks as a Function of Fuel Pin Pitch

Enrichment

The fuel to be stored in the racks may range in enrichment from 1.7 wt% ²³⁵U to 5 wt% ²³⁵U. It was determined that there is not a statistically significant trend on enrichment. However, to be conservative, both the zero slope and the calculated fit are used for determining the limiting k as a function of enrichment. Using NUREG/CR-6698 [A.2] equations from above and the data from Table A.4.1, the trend in the mean k is:

$$k(\text{Enrichment}) = 0.99699 + (0.01134) \cdot \text{Enrichment}$$

where Enrichment is wt% ²³⁵U.

The maximum tolerance band width is 0.0051. Figure A.5.6 graphically presents the results.

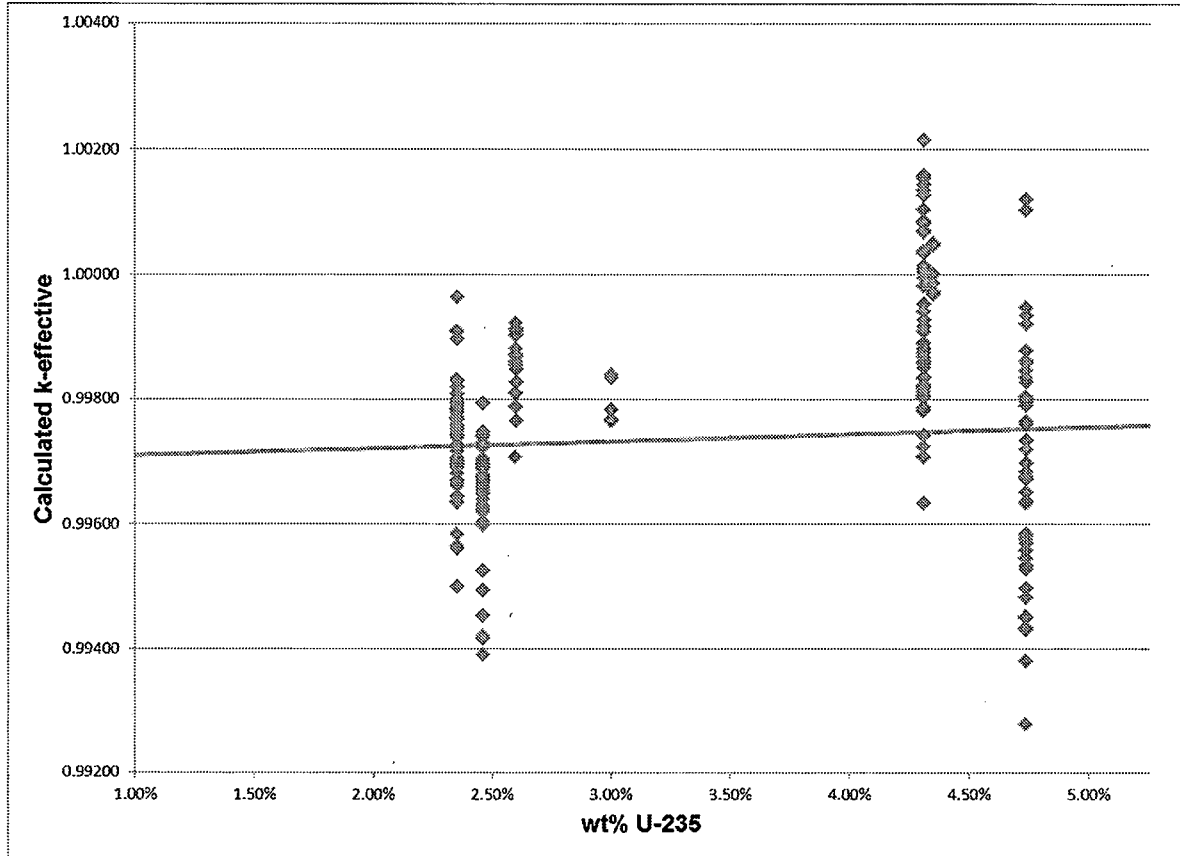


Figure A.5.6: Calculated k for the UO₂ Critical Benchmarks as a Function of Enrichment

Soluble Boron Content

A fit of the calculated k's as a function of the soluble boron ppm was performed using the data from Table A.4.1 and Table A.5.2. The trend on soluble boron concentration is not statistically significance test compared to a zero slope. However, to be conservative, both the zero slope and the calculated fit are used for determining the limiting k as a function of soluble boron content.

The following equation is the best fit of the data for k versus soluble boron. Figure A.5.7 shows the results of the analysis. The uncertainty around the mean value given in the following equation is 0.0053 at 2550 ppm (maximum uncertainty in the range of interest).

$$k(\text{ppm soluble boron}) = 0.99739 + (6.32\text{E-}08) \cdot \text{ppm}$$

Table A.5.2: UO2 Critical Experiment With Soluble Boron Results with SCALE 6.0 and ENDF/B-VII

Benchmark ID	Case No.	Soluble Boron ppm	Measurement Uncertainty (delta k)	k _{eff}
LCT-8	1	1511	0.0012	0.9965
	2	1334	0.0012	0.9969
	3	1337	0.0012	0.9974
	4	1183	0.0012	0.9969
	5	1181	0.0012	0.9962
	6	1034	0.0012	0.9966
	7	1031	0.0012	0.9965
	8	794	0.0012	0.9960
	9	779	0.0012	0.9964
	10	1245	0.0012	0.9966
	11	1384	0.0012	0.9970
	12	1348	0.0012	0.9970
	13	1348	0.0012	0.9969
	14	1363	0.0012	0.9967
	15	1362	0.0012	0.9962
	16	1158	0.0012	0.9973
	17	921	0.0012	0.9963
LCT-11	2	1037	0.0032	0.9963
	3	769	0.0032	0.9967
	4	764	0.0032	0.9974
	5	762	0.0032	0.9967
	6	753	0.0032	0.9968
	7	739	0.0032	0.9975
	8	721	0.0032	0.9968
	9	702	0.0032	0.9969
LCT-35	1	70	0.0018	0.9981
	2	147.7	0.0019	0.9971
LCT-50	3	822	0.0010	0.9970
	4	822	0.0010	0.9967
	5	5030	0.0010	0.9983
	6	5030	0.0010	0.9986
	7	5030	0.0010	0.9988
LCT-51	1 C10	143	0.0020	0.9960
	2 c11a	510	0.0024	0.9979
	3 c11b	514	0.0024	0.9973
	4 c11c	501	0.0024	0.9970
	5 c11d	493	0.0024	0.9974
	6 c11e	474	0.0024	0.9967
	7 c11f	462	0.0024	0.9972
	8 c11g	432	0.0024	0.9970
	9 c12	217	0.0019	0.9968

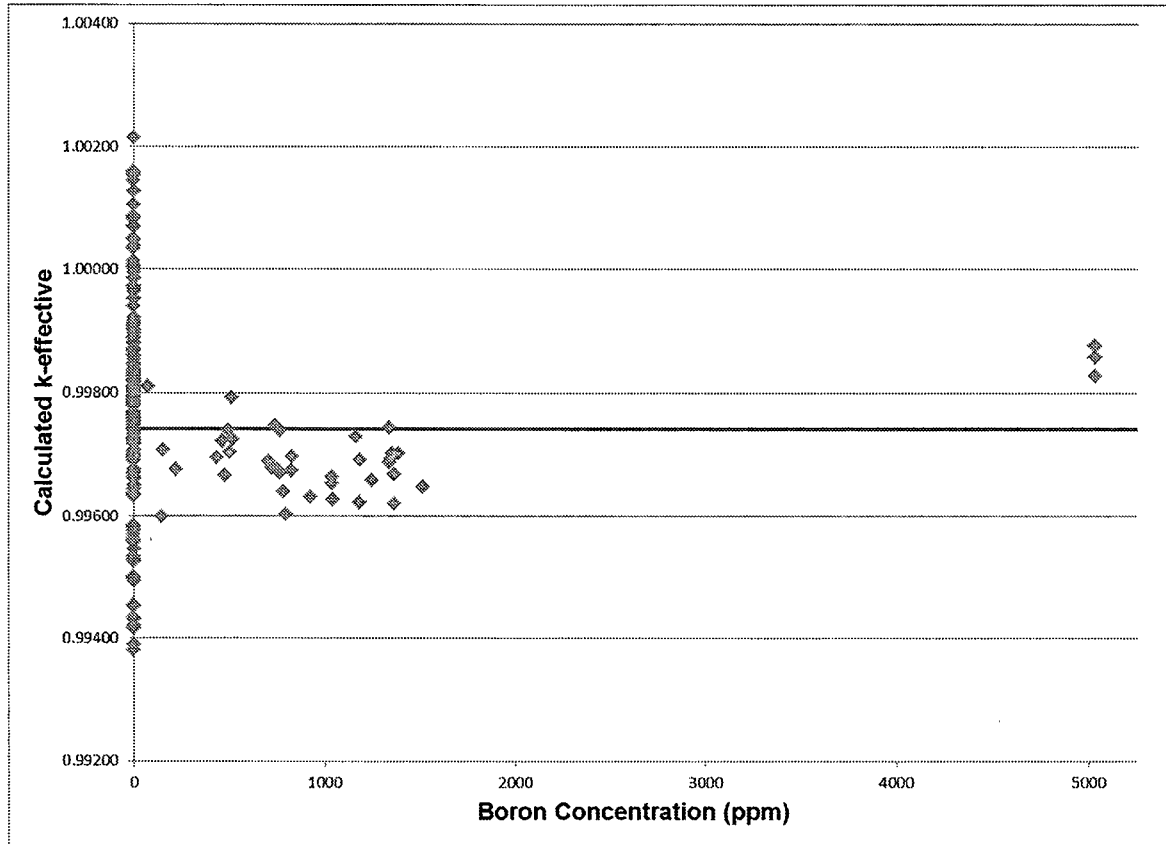


Figure A.5.7: Calculated k for the UO₂ Critical Benchmarks as a Function of Soluble Boron

Establishing the Bias and the Uncertainty

To make the incorporation of the bias and bias uncertainty in the criticality analysis conservative, the most limiting bias and bias uncertainty from the trends in the range of interest is used. The lattice pitch for Westinghouse fuel 17x17 fuel is 1.26 cm. The bias from the pitch trend is **0.0034**. The pin diameter is 0.95 cm where the bias as a function of pin diameter is 0.0031. The maximum bias from the enrichment trend is only 0.0027. The maximum bias as a function of soluble boron content is only 0.0026. There is a fairly strong trend with EALF. The bias increases with harder spectrum. The conservative bias to be used is 0.0034 or the EALF bias, whichever is greater. For all EALF less than 0.4 eV the bias from the pin pitch trend is most limiting. Table A.5.3 is a combination of the 0.0034 and the EALF values from Table A.5.1

The maximum uncertainty is also used but for this analysis if the bias is less than 0.0034, it is appropriate to subtract the differences in biases from the calculated uncertainties. The uncertainty from the pin pitch is 0.0047. Although the uncertainty from the pin diameter is slightly higher (0.0048) the bias for the pin diameter is less by 0.0003 so the pitch based uncertainty of 0.0047 should be used. Similarly, the uncertainty from the enrichment and soluble boron based trends are 0.0004 and 0.0006 higher than 0.0047 but the maximum bias from either of those two trends is less by 0.0007. Thus for all the trends other than the trend on EALF the conservative uncertainty is 0.0047. For all EALF under 0.4 eV the uncertainty based on the pin pitch is most limiting. Table A.5.3. provides the most limiting uncertainty using the uncertainties from Table A.5.1 (EALF) and 0.0047 from the other trends.

Table A.5.3: Bias and Uncertainty Based on the EALF Trend from UO2 Critical Experiments

Fresh Fuel		
Maximum EALF (ev)	Bias	Uncertainty
0.40	0.0034	0.0047
0.45	0.0037	0.0047
0.50	0.0039	0.0048
0.55	0.0042	0.0049
0.60	0.0044	0.0050
0.65	0.0047	0.0051
0.70	0.0049	0.0052
0.75	0.0052	0.0053
0.80	0.0054	0.0054
0.85	0.0056	0.0055
0.90	0.0059	0.0057
0.95	0.0061	0.0058
1.00	0.0064	0.0059
1.05	0.0066	0.0060
1.10	0.0069	0.0061
1.15	0.0071	0.0062
1.20	0.0074	0.0063

Impact of Cd and boron cases on the bias

The mean unweighted k of the 232 cases used for the bias and trends is 0.99775. The mean unweighted k of the 17 cases including Cd is 0.99821. The delta k between these means is

0.0005. This is less than one third of the standard deviation of the set as a whole. Including the Cd set of experiments has no significant effect on the set of experiments used in the analysis.

Likewise the mean unweighted k of the 69 cases that contain boron is 0.99745. This is well within the uncertainty of the set as a whole so including boron cases has no significant effect on the set of experiments used in the analysis.

A.5.2 Statistical Analysis of MOX Critical Experiments

Tables A.4.2 and A.4.3 provides the raw results of the analysis of the MOX critical experiments. From the calculated k 's provided in Tables A.4.1, A.4.2 and A.4.3 there is a trend on the plutonium content. Figure A.5.8 shows this trend.

Since there is a strong trend on plutonium content, the critical experiments with plutonium content out of the range of spent fuel have been excluded. Only the critical experiments with 2 wt% plutonium or less are included in the trending analysis for MOX critical experiments. This set of MOX experiments consists of 136 critical experiments. Similar to the UO₂ benchmark set, for the Shapiro-Wilks test, we reject the null hypothesis for the MOX benchmark set at the 95% confidence level. The SW Test statistic for the MOX benchmark set is 0.01, indicating that it would pass the SW test at the 99% confidence level. Figure A.5.9a shows the histogram for the 136 calculated k 's, and A.5.9b shows the Q-Q plot. The visual inspection of the Q-Q plot does not show any clear indication of non-normal behavior, such as non-linear trend, uneven data point groupings, or skewness. Therefore even though the null hypothesis for the SW test is rejected at the 95% level, it is reasonable to treat the data as normally distributed.

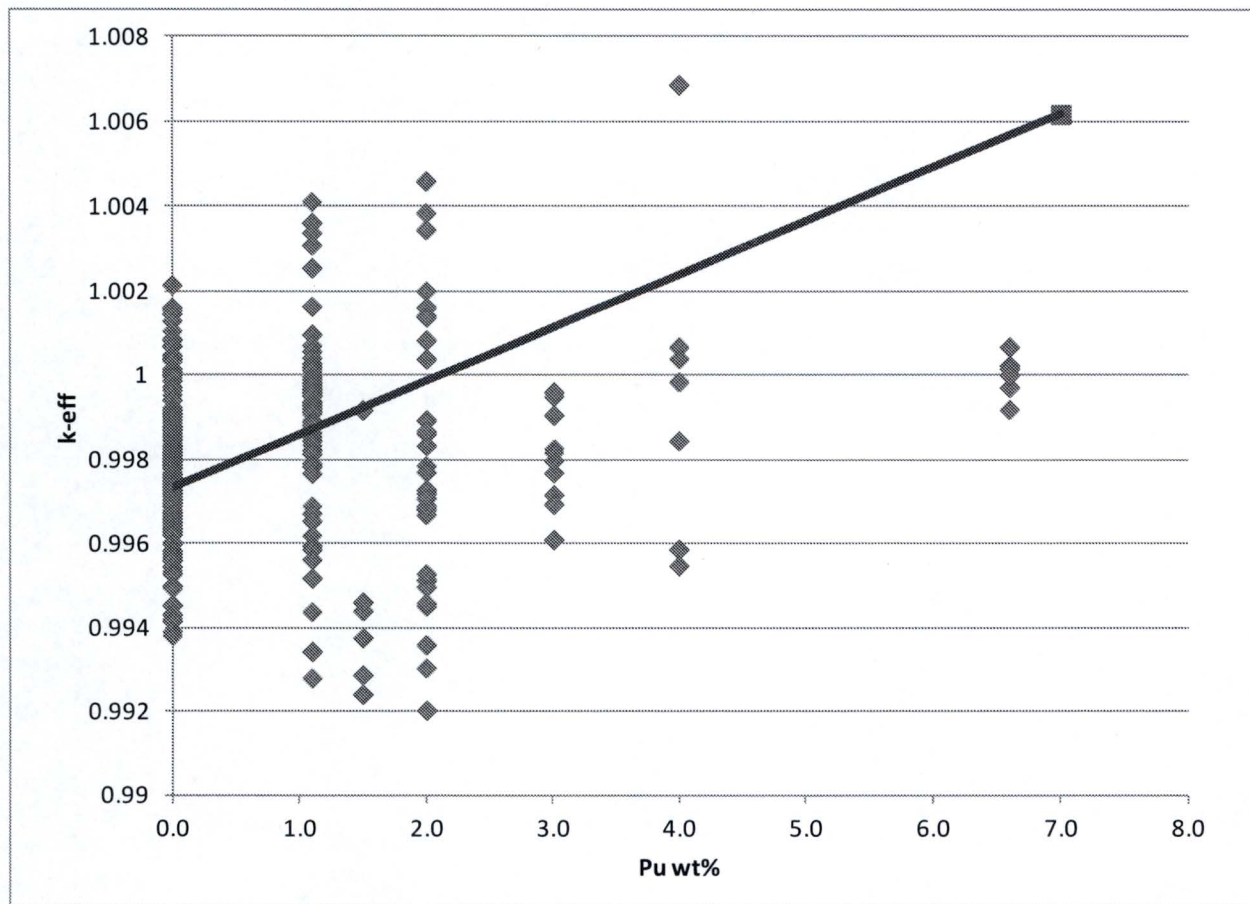


Figure A.5.8: Calculated k for the Critical Benchmarks as a Function of Plutonium Content

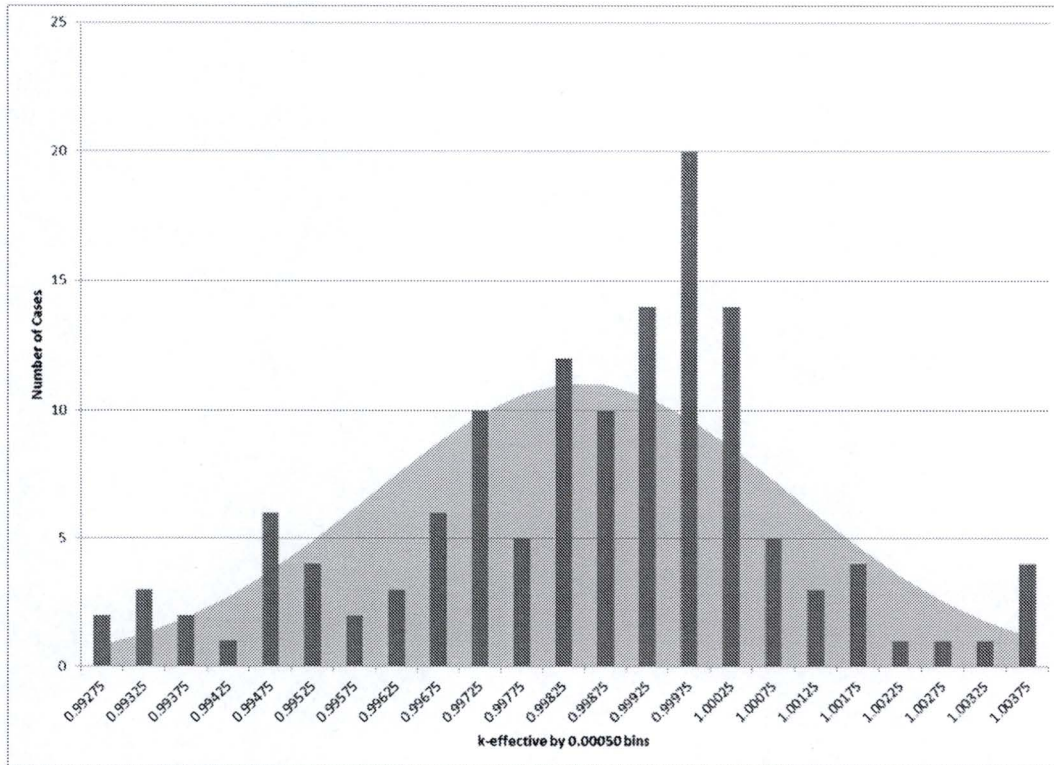


Figure A.5.9a: Distribution of Calculated ks for the MOX Critical Benchmarks

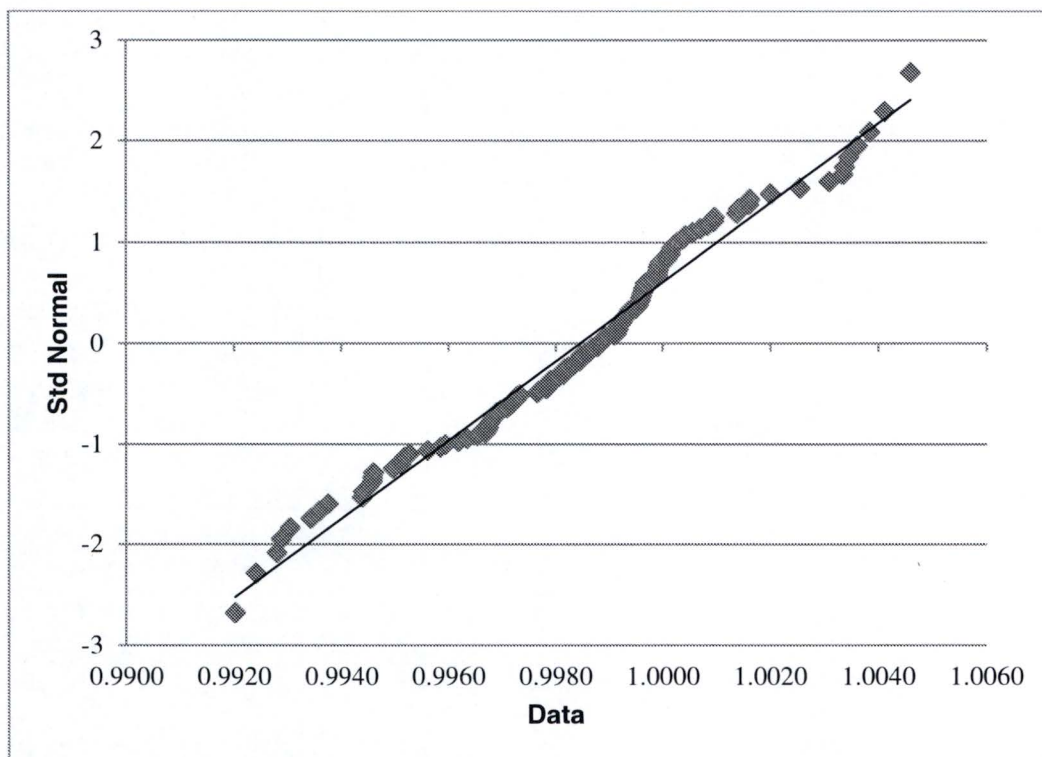


Figure A.5.9b: Q-Q Normality Plot for the MOX Benchmarks

As with the UO₂ set, the MOX set has a trend with EALF. That trend is:

$$k(\text{EALF}) = 0.999247 - 0.00264 * \text{EALF}$$

This is graphically presented as Figure A.5.10. The bias and uncertainty as a function of EALF is found on Table A.5.4. For accident conditions where the full Technical Specification soluble boron is used, the EALF may reach 0.75 eV. If partial voiding of the spent fuel pool is assumed, then the EALF approaches 1.0 eV but these cases are never close to limiting allowing a large amount of margin to cover the extrapolation above the highest EALF in the critical experiments of 0.71 eV. Spent fuel is not allowed in the new fuel vault, so the optimum moderation hard spectrum is not a concern for the burned fuel.

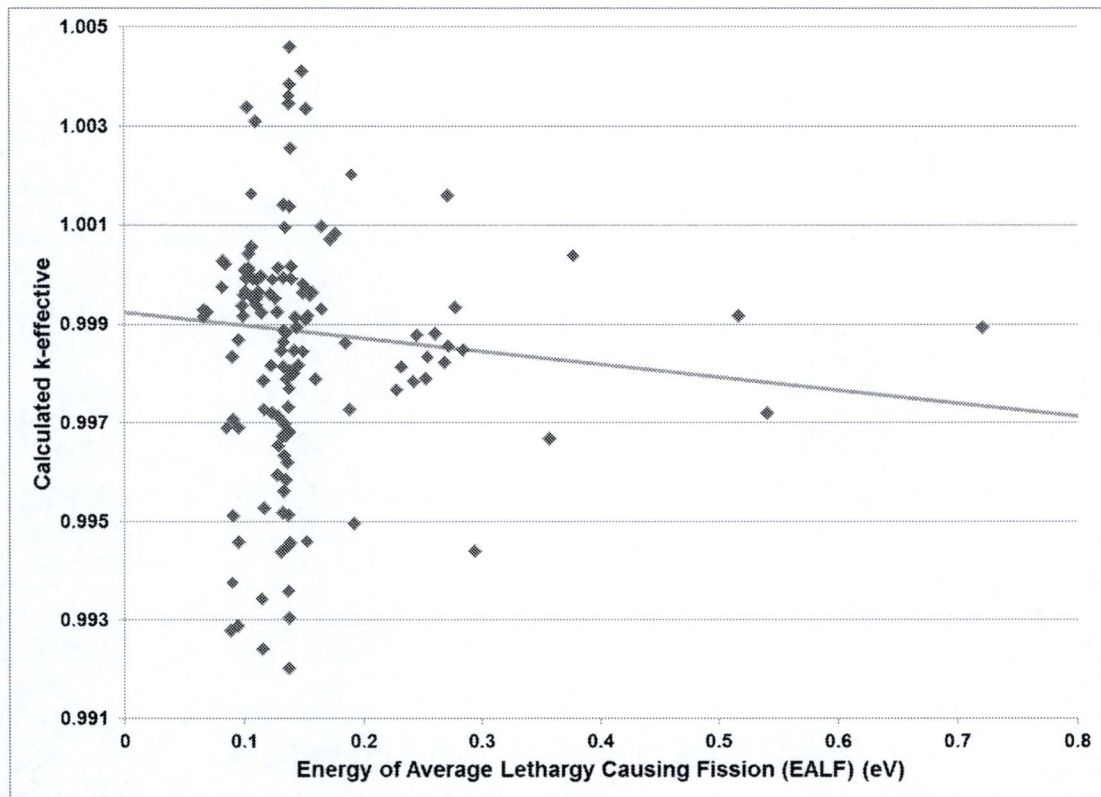


Figure A.5.10: Calculated MOX Critical k as a Function of EALF

Table A.5.4: Bias and Uncertainty Based on the EALF Trend from MOX Critical Experiments

Fresh Fuel		
Maximum EALF (ev)	Bias	Uncertainty
0.30	0.0017	0.0079
0.35	0.0017	0.0083
0.40	0.0018	0.0088
0.45	0.0019	0.0094
0.50	0.0021	0.0099
0.55	0.0022	0.0104
0.60	0.0023	0.0110
0.65	0.0025	0.0115
0.70	0.0026	0.0121
0.75	0.0027	0.0126
0.80	0.0029	0.0132
0.85	0.0030	0.0137
0.90	0.0031	0.0143
0.95	0.0033	0.0148
1.00	0.0034	0.0154
1.05	0.0035	0.0159
1.10	0.0037	0.0165
1.15	0.0038	0.0170
1.20	0.0039	0.0176

The bias for the MOX is smaller than the UO2 set, but the uncertainty for the MOX data is much higher. Until the combined total of the uncertainties in the spent fuel storage area is considered, it is not clear which is more limiting. Therefore both the UO2 and MOX bias and uncertainty are used in determining the most limiting condition.

A.5.3 Subcritical Margin

In the USA, the NRC has established subcritical margins for rack analysis. The subcritical margin for borated spent fuel pools, casks, and fully flooded dry storage racks is 0 when the analysis is performed with unborated water. This is actually saying the subcritical margin is contained in the uncredited soluble boron. To make sure there is sufficient soluble boron, analysis is also performed with soluble boron and a subcritical margin of 5% in k is required. For dry storage racks analyzed with optimum moderation, the subcritical margin is 2% and 5% with full moderation. In the analysis of 232 critical experiments, which generously cover the

range of expected conditions, the lowest calculated k was 0.9928. The additional 136 MOX experiments also support this subcritical margin since the lowest calculated k is 0.9920. The subcritical margin is more than sufficient.

A.6. Area of Applicability (Benchmark Applicability)

The critical benchmarks selected cover both the new fuel storage area and the spent fuel pool of MPS3. To summarize the range of the benchmark applicability (or area of applicability), Table A.6.1 is provided below.

Table A.6.1: Area of Applicability (Benchmark Applicability)

Parameter	Range	Comments
Fissionable Material/Physical Form	UO ₂	The fuel material is the same as in the benchmark experiments
Enrichment (wt% U-235)	2.35 to 4.74	The analysis covers possible low enrichments in the future that would require extrapolation. Assuming the enrichment is zero (y intercept of the trend) the selected bias generously covers the extrapolated bias. Therefore extrapolation of the bias to lower enrichments is justified. An extrapolation from 4.74 to 5 wt% will also be needed but in this direction the bias is decreasing so the data is adequate for this extrapolation.
Spectrum - EALF (eV)	0.0605 to 0.8485	Expected range in spent fuel pool applications without voiding: 0.1 to 0.8 eV Voided cases have significant margin to allow for the extrapolation. For optimum moderation in new fuel storage area the EALF may be as high as 1.1 eV for this some extrapolation of the data is done.
Lattice Characteristics Type Pin Pitch (cm)	Square 1.075 to 2.54	Hex lattices have been excluded. W 17x17 pin pitch is 1.26.

Parameter	Range	Comments
Assembly Spacing in Racks Distance between Assemblies (cm)	0 to 15.4	This covers all spacing. Neutron transport through larger than 15.4 cm has a small effect on k. Note that the spacing is assumed to be filled with full density water. If the water density is less, this separation effectively decreases. Therefore, optimum moderation cases of wide spaced racks are covered.
Absorbers Soluble Boron Concentration	0 to 5030 ppm	All designs are within this range.
Absorbers Cd (for Ag-In-Cd rods)	Cd Absorber panels	Although Cd is in panels, the inclusion or exclusion had no significant effect on the bias and uncertainty so credit for control rods is acceptable.
Reflector Experiments included water and steel	Reflectors adequately covered	Most racks are reflected by water, steel, and concrete which was covered in the set of experiments.
Temperature	Room Temperature (Reference 9 provides a bias for up to 85° C)	Most criticality calculations are performed with the fuel at low temperatures. A separate set of experiments are used for a temperature bias covered in Reference 9.
Moderating material	Water	The moderator in all benchmark experiments is water, therefore water as a moderating material is covered

A.7. Summary and Recommendations

232 UO₂ and 136 MOX critical experiments were analyzed with SCALE 6.0 and the 238 group ENFD/B-VII cross section set. The calculated k's were analyzed for trends using the statistical approach recommended in NUREG/CR-6698. Table A.7.1 provides the maximum bias and uncertainty for each trend.

For the spent fuel pool, the bias and uncertainty depends on the burnup since at low burnup the dominant fissile material is U-235 and at high burnup the dominant fissile material is Pu-239. In order to avoid trying to properly weight the critical experiments for the amount of U-235 and Pu-239, it is recommended to use two sets of bias and uncertainty, one from the fresh UO₂ critical experiments and one from the MOX critical experiments. The final bias and uncertainty to be used will be that which produces the highest 95/95 k. The UO₂ critical experiments have a

higher bias but lower uncertainty than the MOX experiments. Since the uncertainty is statistically combined with other uncertainties it is not possible to determine which set is more limiting until the other uncertainties due to factors such as manufacturing tolerances are determined.

Table A.7.2 provides the most limiting bias and uncertainty from both the UO₂ and MOX sets as a function of EALF.

Table A.7.1: Summary of the Trend Analysis

Trend	Equation	Maximum Bias	Maximum Uncertainty
UO₂ Critical Experiments			
No trend	n/a	0.0026	0.0050
EALF	$0.99853 - 0.00491 * \text{EALF (ev)}$	Bias as a function of EALF is used	Uncertainty as a function of EALF is used
Fuel Pin Diameter	$0.99421 + 2.82\text{E-}03 * \text{Pin Dia. (cm)}$	0.0031	0.0048
Lattice Pitch	$0.99433 + 1.84\text{E-}03 * \text{Pitch (cm)}$	0.0034	0.0047
Enrichment	$0.99699 + 0.01134 * \text{U235wt\%t}$	0.0027	0.0051
Soluble Boron	$0.99739 + 6.32\text{E-}08 * \text{ppm}$	0.0026	0.0053
MOX Critical Experiments			
No trend	n/a	0.0011	0.0065
EALF	$0.999247 - 0.00264 * \text{EALF (ev)}$	Used as a function of EALF	Used as a function of EALF

Table A.7.2: Final Bias and Uncertainty

(For burned fuel calculate with both UO2 and MOX bias and uncertainty and use the set that provides the highest 95/95 k)

UO2			MOX		
Maximum EALF (ev)	Bias	Uncertainty	Maximum EALF (ev)	Bias	Uncertainty
0.35	0.0034	0.0048	0.35	0.0017	0.0083
0.40	0.0034	0.0048	0.40	0.0018	0.0088
0.45	0.0037	0.0048	0.45	0.0019	0.0094
0.50	0.0039	0.0048	0.50	0.0021	0.0099
0.55	0.0042	0.0049	0.55	0.0022	0.0104
0.60	0.0044	0.0050	0.60	0.0023	0.0110
0.65	0.0047	0.0051	0.65	0.0025	0.0115
0.70	0.0049	0.0052	0.70	0.0026	0.0121
0.75	0.0052	0.0053	0.75	0.0027	0.0126
0.80	0.0054	0.0054	0.80	0.0029	0.0132
0.85	0.0056	0.0055	0.85	0.0030	0.0137
0.90	0.0059	0.0057	0.90	0.0031	0.0143
0.95	0.0061	0.0058	0.95	0.0033	0.0148
1.00	0.0064	0.0059	1.00	0.0034	0.0154
1.05	0.0066	0.0060	1.05	0.0035	0.0159
1.10	0.0069	0.0061	1.10	0.0037	0.0165
1.15	0.0071	0.0062	1.15	0.0038	0.0170
1.20	0.0074	0.0063	1.20	0.0039	0.0176

For unburned fuel in the spent fuel pool and new fuel vault use only the UO2 set from the above.

A.8. Temperature Bias

All of the critical experiments used thus far have been at room temperature. There could be a bias in k in the temperature range of interest to spent fuel pools and dry storage racks (0 to 100 C). There is one critical benchmark evaluation in the OECD/NEA handbook [A.3] that performed measurements with elevated temperatures in this range, LEU-COMP-THERM-046 (shortened to LCT-046). LCT-046 consists of 22 experiments but the last 5 experiments contain copper rods. Since copper is not in North Anna's spent fuel pool only the first 17 experiments are analyzed.

The 17 LCT-046 experiments have been analyzed using SCALE 6.0 [A.2] and the 238 ENDF/B-VII cross section library. Section 3 of LCT-046 provides the details for analysis of the critical

benchmark. The SCALE models used follows that specification. All the expansion factors from Table 29 of LCT-046 were applied to all the x-y dimensions. That means that the same stainless steel component expansion factor was applied to pitch and the inner and outer diameter of the clad. This is consistent with the MCNP samples given in the Appendix of LCT-046. For the axial expansion only the fuel was expanded. As with the MCNP sample input, the same expansion factor was used for the radius and the axial direction. The fuel column is 54.84 cm long (unexpanded). Due to the control rod bottom plug which hangs into the fuel region, the fuel is modeled as a 53.44 cm long (unexpanded) zone followed by two shorter zones. For this effort only the 53.44 cm long segment was expanded axially by the expansion factor. This approach assures that the axial position of the control rod and bottom plug is not changed.

Table A.8.1 shows the corrected SCALE 6.0 ENDF/B-VII results for the 17 critical experiments. Corrected results in this case means they were divided by the k of the benchmark which was not quite 1.0.

Table A.8.1: LCT-046 with Full Thermal Expansion Calculated with SCALE 6.0 and ENDF/B-VII

Case	Temperature (K)	Corrected SCALE k	SCALE sigma
1	297.05	0.998901	0.00007
2	310.41	0.998867	0.00007
3	315.43	0.998710	0.00007
4	319.96	0.998915	0.00007
5	324.93	0.998558	0.00007
6	332.53	0.998697	0.00007
7	287.22	0.999163	0.00007
8	315.91	0.998854	0.00006
9	330.27	0.998669	0.00007
10	337.44	0.998566	0.00007
11	351.99	0.998625	0.00007
12	303.60	0.998632	0.00007
13	312.95	0.998616	0.00007
14	321.16	0.998511	0.00007
15	328.24	0.998258	0.00007
16	338.26	0.998147	0.00007
17	358.31	0.998057	0.00007

Figure A.8.1 plots the results of the analysis as a function of case. As can be seen from this plot there does seem to be a small trend with temperature. Figure A.8.2 is the data plotted against temperature with the least squares linear fit. The nominal slope of the fit is $-1.14\text{E-}05$

$\Delta k/\Delta C$. Using the EXCEL regression function the most limiting slope with 95% certainty is $-1.7E-05 \Delta k/\Delta C$.

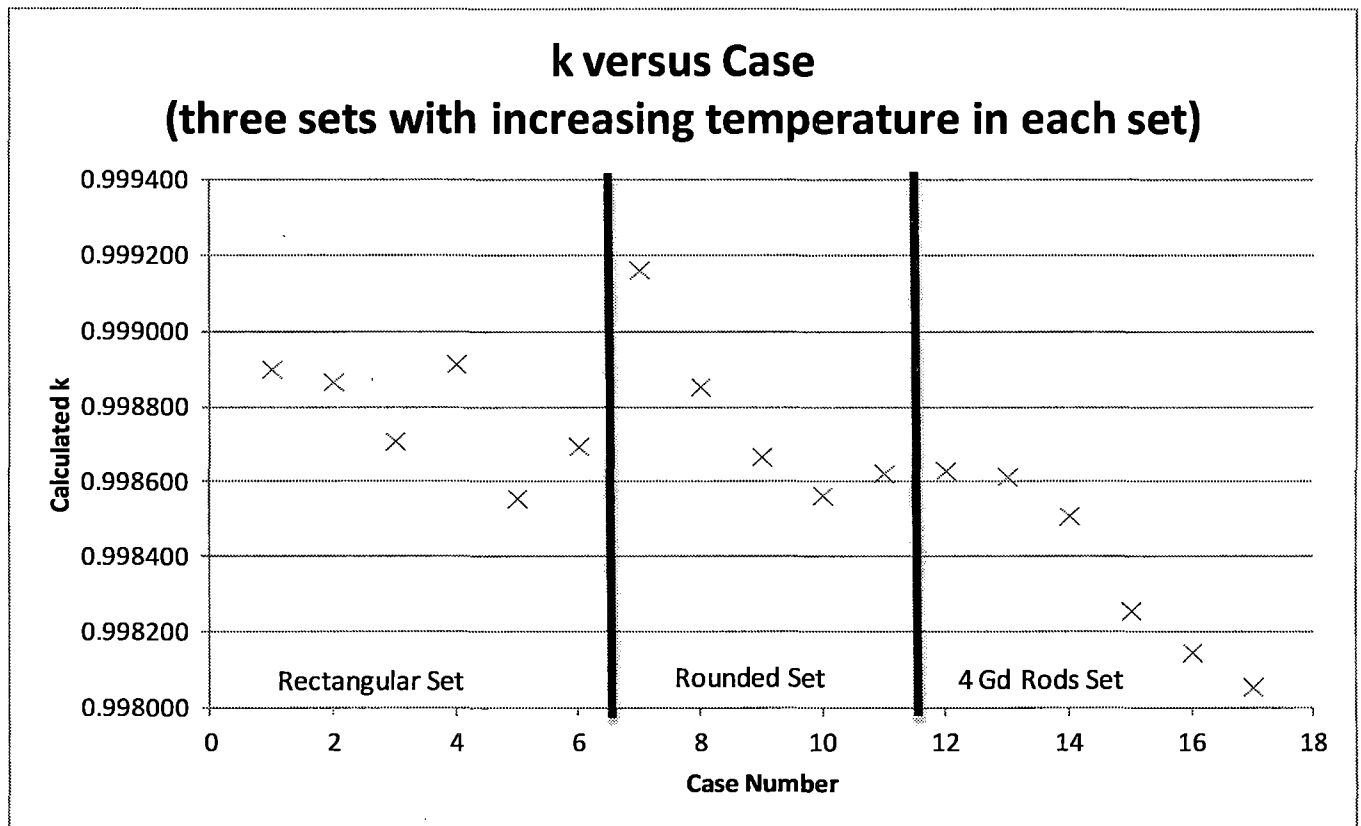


Figure A.8.1: LCT-046 Corrected Calculated k per Case

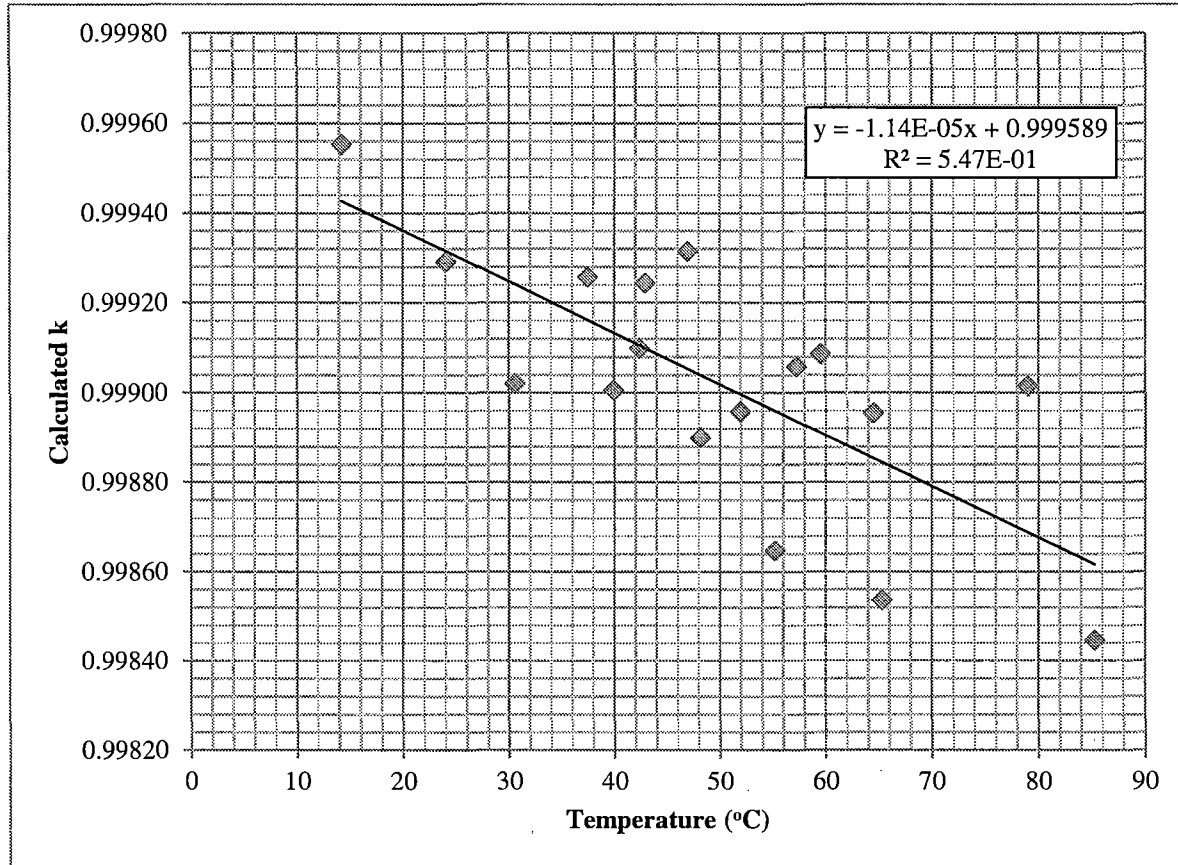


Figure A.8.2: LCT-046 k versus Temperature

The analysis of the only set of thermal critical experiments in the International Handbook that uses elevated temperatures in the range of 0 to 100 C has shown a small increase in the bias with temperature. This increase can be conservatively handled by a bias from room temperature of $1.7E-05 \Delta k/^\circ\text{C}$. This bias is the lower (most negative slope) 95% confidence slope of the fit line.

A.9. Appendix References

- [A.1] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39, Version 6, Volumes 1-3, January 2009.
- [A.2] J.C. Dean and R.W. Tayloe, Jr., *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, NUREG/CR-6698, Nuclear Regulatory Commission, Washington, DC January 2001
- [A.3] *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)3, Volumes IV and VI, Nuclear Energy Agency, OECD, Paris, September, 2010.
- [A.4] D. E. Mueller, K. R. Elam, and P. B. Fox, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979 (ORNL/TM-2007/083), prepared for the US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn., September 2008.
- [A.5] F. Fernex, "Programme HTC – Phase 1 : Réseaux de crayons dans l'eau pure (Water-moderated and reflected simple arrays) Réévaluation des expériences," DSU/SEC/T/2005-33/D.R., Institut de Radioprotection et de Sûreté Nucléaire, 2008.
- [A.6] F. Fernex, *Programme HTC – Phase 2 : Réseaux simples en eau empoisonnée (bore et gadolinium) (Reflected simple arrays moderated by poisoned water with gadolinium or boron) Réévaluation des expériences*, DSU/SEC/T/2005-38/D.R., Institut de Radioprotection et de Sûreté Nucléaire, 2008.
- [A.7] F. Fernex, *Programme HTC – Phase 3 : Configurations "stockage en piscine" (Pool storage) Réévaluation des expériences*, DSU/SEC/T/2005-37/D.R., Institut de Radioprotection et de Sûreté Nucléaire, 2008.
- [A.8] F. Fernex, *Programme HTC – Phase 4 : Configurations "chateaux de transport" Réévaluation des expériences*, DSU/SEC/T/2005-36/D.R., Institut de Radioprotection et de Sûreté Nucléaire, 2008.
- [A.9] J. P. Royston, "Approximating the Shapiro-Wilk W-test for non-normality", *Statistics and Computing*, Vol. 2,, September 1992.
- [A.10] Kleinbaum, Kupper, and Muller, *Applied Regression Analysis and Other Multivariable Methods*, Second Edition, page 48, PWS-KENT Publishing Company, Boston, MA 1988.
- [A.11] "PROPHET StatGuide: Examining normality test results," http://www.basic.northwestern.edu/statguidefiles/n-dist_exam_res.html, located on 6/8/09.
- [A.12] R. Mark Sirkin, *Statistics for the Social Sciences*, Third Edition, 2005, page 245, Sage Publications, Thousand Oaks, CA.

ATTACHMENT 7

SPENT FUEL POOL BORON DILUTION ANALYSIS

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Table of Contents

1.	Introduction.....	3
2.	Spent Fuel Pool and Related System Features	4
2.1.	Spent Fuel Pool	4
2.2.	Spent Fuel Pool Storage Racks	4
2.3.	Spent Fuel Pool Cooling	5
2.4.	Spent Fuel Pool Instrumentation.....	5
2.5.	Spent Fuel Pool Administrative Procedure.....	6
2.6.	Boration Sources	6
3.	Spent Fuel Pool Dilution Event	7
3.1.	Calculation of SFP Volume	7
3.2.	Plant Equipment Operator Response	7
3.3.	Calculation of SFP Boron Dilution Water Flow Rate	8
4.	Dilution Source Path Evaluation	10
4.1.	Unintentional Overfilling from Primary Grade Water (PGS) System	10
4.2.	Unintentional overfilling from normally isolated Service Water System	10
4.3.	SFP Cooling System Heat Exchanger Tube Rupture.....	10
4.4.	Excess Demineralized Water Addition from Rinsing Transfer Cask and Equipment ...	11
4.5.	Unintentional overfilling using RWST connection to SFP	11
5.	Pipe Breaks and Leaks.....	12
5.1.	Component Cooling, Fire Protection, and Hot Water Pre-Heating Water Line Breaks	13
5.2.	Hot Water Heating (HWH) Line Break	13
5.3.	Roof Drain Line Break	15
5.4.	Fuel Pool Cooling Lines.....	16
5.5.	Fire Water Hose Station Pipe Leakage.....	16
5.6.	Demineralized Water Pipe and Valve Leakage	16
5.7.	Hot Water Heating System leakage	16
5.8.	Domestic Water Pipe Leakage	16
5.9.	Roof Drain pipe leakage	17
6.	Securing the Dilution Source	18
7.	Conclusions	19
8.	References	21

1. Introduction

A Spent Fuel Pool (SFP) criticality reanalysis has been completed for crediting soluble boron in the Millstone Unit 3 (MPS3) SFP under both normal and accident conditions. As a result of the soluble boron credit, a boron dilution analysis is required and is presented herein. This analysis includes the following plant specific features and potential events:

- Level indicator instrumentation
- Administrative procedures
- Boration sources
- Dilution sources
- Dilution flow rates
- Boron dilution initiating events
- Boron dilution times and volumes
- Boron dilution mitigation actions

The boron dilution analysis, along with the criticality reanalysis, ensures that sufficient time is available to detect and mitigate the dilution before the design basis limit on the effective multiplication factor ($k_{\text{eff}} = 0.95$) is reached.

The new, proposed Technical Specifications will change the minimum SFP boron concentration to 2600 ppm. This dilution analysis will assume the dilution event starts at a boron concentration of 2300 ppm and the lower boron concentration limit is 700 ppm. The starting boron concentration of 2300 ppm was conservatively chosen prior to the completion of the criticality accident analysis that established 2600 ppm as the Technical Specifications minimum boron value.

2. Spent Fuel Pool and Related System Features

This section provides background information on the SFP and related systems.

2.1. Spent Fuel Pool

The spent fuel pool is an L-shaped structure located in the southwestern quadrant of the fuel building. Two adjacent areas, which are accessible from the spent fuel pool by means of sealable gates, are the transfer canal and the spent fuel shipping cask pit.

The spent fuel pool is designed to accommodate fuel racks that store new and spent fuel assemblies. The storage racks are located under water in the spent fuel pool. (UFSAR Section 9.1.2.2, Ref. 1).

2.2. Spent Fuel Pool Storage Racks

There are three different fuel storage regions in the spent fuel pool. The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 total available fuel storage locations. An additional Region 2 rack with 81 storage locations has been licensed by the NRC, but is not physically installed in the spent fuel pool. If this additional rack is installed, the Region 2 storage capacity is 754 storage locations. The total storage capacity of the spent fuel pool is limited to no more than 1860 fuel assemblies.

The Region 1 fuel storage racks are made up of 5 rack modules. Each rack module is free standing and is made up of a 7 by 10 array of storage cells. The Region 1 racks have a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center-to-center spacing between adjacent fuel storage locations. The Region 1 storage racks have a neutron flux trap design, which uses BORAL as the active neutron absorber. BORAL panels are included on all peripheral rack locations.

The Region 2 fuel storage racks are made up of 9 rack modules. Each rack module is free standing with several different storage array sizes. The Region 2 racks have a nominal 9.017 inch center-to-center spacing between adjacent fuel storage locations. Like the Region 1 racks, the Region 2 storage racks use BORAL as the active neutron absorber. The Region 2 storage

racks have a single BORAL panel between adjacent fuel assemblies. BORAL panels are included on all peripheral rack locations.

The Region 3 fuel storage racks are made up of 21 rack modules. Each rack module is free standing and is made up of a 6 by 6 array of storage cells. Each rack consists of cells welded to a grid base and welded together at the top through an upper grid to form an integral structure. The Region 3 racks have a nominal 10.35 inch center-to-center spacing between adjacent fuel storage locations. The Region 3 storage racks have a neutron flux trap design, which uses Boraflex as the active neutron absorber. However, Boraflex is no longer credited in the criticality analysis of these racks. Each rack module is provided with adjustable leveling pads at the center of the four corner cells within the module. (UFSAR Section 9.1.2.2, Ref. 1).

2.3. Spent Fuel Pool Cooling

Cooling for the spent fuel pool consists of two cooling mechanisms. The first is the active cooling provided by the fuel pool heat exchangers. The spent fuel pool water flows from the fuel pool discharge through either of the two fuel pool cooling pumps and through the tube side of a fuel pool cooler, and then returns to the fuel pool. One fuel pool cooling pump and cooler are normally in service. Cooling for the fuel pool coolers is provided by the reactor plant component cooling water system. The second mechanism is the passive cooling provided by evaporative cooling from the surface of the pool. The spent fuel pool cooling system has been analyzed to remove the decay heat load of up to 1960 fuel assemblies and maintain a bulk pool temperature at or below 150°F using a single train of spent fuel pool cooling. (UFSAR Section 9.1.3.2, Ref. 1).

2.4. Spent Fuel Pool Instrumentation

The fuel pool has redundant safety grade low level alarm lights and temperature indicators provided in the main control room. Non-safety grade level indication is provided locally and high and low level alarms are provided both locally and in the main control room. Continuous non-safety augmented quality wide range level indication is provided remotely in the Auxiliary Building. (UFSAR Section 9.1.3.5, Ref. 1).

If the pool were to overflow, the water would eventually collect into and auto-start the fuel building sump pump (FSAR Section 9.3.3.3, Ref. 1). The control room would get an indication

of the pump's auto-start. This analysis assumes that these automatic safety systems failed, and the only way operations can discover a dilution event is during operator rounds.

2.5. Spent Fuel Pool Administrative Procedure

Currently, Technical Specifications (TS) requires the soluble boron concentration in the SFP to always be greater than or equal to 800 ppm. The SFP soluble boron concentration must also be greater than or equal to 2600 ppm during refueling operations when fuel is being transferred between the fuel pool and refueling cavity to preclude uncontrolled dilution of the filled portion of the RCS (UFSAR Section 9.1.3.2, Ref. 1). Chemistry practice maintains the SFP greater than or equal to 2600 ppm since that is the boron concentration assumed in the "Fuel Pool Cooling and Purification" operating procedure. TS requirements are proposed to be changed to require the soluble boron concentration to be maintained ≥ 2600 ppm. The spent fuel pool boron concentration must be measured every 7 days in accordance with plant surveillance procedure.

Plant surveillance procedure states the shift readings must be checked and completed in the first 3 hours of each shift. These shifts start at 0700 and 1900 hours.

2.6. Boration Sources

Normal makeup to the fuel pool, necessitated by losses due to evaporation, is primary grade water from the primary grade water system. Borated water from the RWST can be used to fill the fuel pool at a concentration matching that used in the refueling cavity during refueling operations (UFSAR Section 9.1.3.2, Ref. 1). The RWST is maintained between 2700 ppm and 2900 ppm (LCO 3.5.4, Ref. 2).

3. Spent Fuel Pool Dilution Event

3.1. Calculation of SFP Volume

Excluding the Spent Fuel Shipping Cask Loading Area, the gate areas, Spent Fuel Shipping Cask Storage Area, and transfer tube, the SFP gross volume is 59,712.08 ft³ up to the low level alarm. This is conservative since the minimum gross volume results in the minimum volume required for dilution. The low level alarm is used for conservatism as this will have the least amount of boric acid to be diluted. The borated water volume is calculated by subtracting fuel assembly (FA) volume, RCCA volume, and Fuel Rack volume from the gross SFP volume. This calculation is as follows:

$$SFP_{borated} = SFP_{low\ level\ alarm} - SFP_{FA} - SFP_{RCCA} - SFP_{Fuel\ Rack} \quad (3.1)$$

$$SFP_{borated} = 59,712.08\ ft^3 - 5,040.6\ ft^3 - 582.9\ ft^3 - 1,511\ ft^3 = 52,577.6\ ft^3$$

Where:

- $SFP_{low\ level\ alarm} = 59,712.08\ ft^3$
- $SFP_{FA} = 5040.6\ ft^3$ (=1860 FAs * 2.71 ft³/FA)
- $SFP_{RCCA} = 582.9\ ft^3$ (=1860 * 0.313373 ft³/RCCA)
- $SFP_{Fuel\ Rack} = 1511\ ft^3$

Note: This calculation models the SFP with the same number of RCCAs as the number of FAs; this is conservative in that it reduces the volume of the SFP water.

This calculation results in a borated water volume of 52,577.6 ft³ or 393,308 gallons.

3.2. Plant Equipment Operator Response

As discussed in Section 2.4, the automatic safety systems that could detect a dilution event are assumed to have failed leaving operator rounds as the only method to discover the event. Plant surveillance procedure states the shift readings must be checked and completed in the first 3 hours of each shift. The ranges of times that the shift readings must be checked and completed are 0700-1000 and 1900-2200. The maximum amount of time that can pass between checking these procedural requirements would be 0700-2200 and 1900-1000 on a 2400 hour scale

(beginning of earlier or later shift to end of the next shift). The time transpired between these two times is 15 hours and is used as the conservative estimate of PEO response time.

3.3. Calculation of SFP Boron Dilution Water Flow Rate

The total amount of unborated water that can be added to the SFP to reduce the boron concentration from 2300 ppm to 700 ppm is determined using continuous dilution methodology (Ref. 3) consistent with the approach employed by Millstone Unit 2 SFP dilution calculation (Ref. 4 & 5). The continuous dilution method models unborated water being added at a constant rate with an equally constant rate of removal of a homogenous mixture of borated water. For conservatism, the SFP volume used excludes areas outside the SFP liner envelope (see Section 3.1 for further detail).

The equation used to calculate the volume required to dilute the SFP boron concentration to 700 ppm is derived in Air Contaminants and Industrial Hygiene Ventilation (Ref. 3).

$$V_{in} = V_{SFP} * \ln\left(\frac{C_o}{C_f}\right) \quad (3.2)$$

$$393,308 \text{ gal} * \ln\left(\frac{2300 \text{ ppm}}{700 \text{ ppm}}\right) = 467,873 \text{ gal}$$

Where:

- V_{in} = Required unborated water to dilute to C_f (gallons)
- V_{SFP} = Volume of SFP to the low level alarm calculated in Section 3.1 (393,308 gallons)
- C_o = Initial boron concentration in SFP (2300 ppm)
- C_f = Final boron concentration in SFP (700 ppm)

The maximum allowable dilution flow rate is determined by dividing the required unborated water volume to dilute the SFP (V_{in}) by the maximum response time by a Plant Equipment Operator (PEO) which is 15 hours.

$$Q_{dilution} = \frac{V_{in}}{t_{PEO}} * \frac{1 \text{ hr}}{60 \text{ min}} \quad (3.3)$$

$$\frac{467,873 \text{ gal}}{15 \text{ hrs}} * \frac{1 \text{ hr}}{60 \text{ min}} = 520 \text{ gpm}$$

Where:

- t_{PEO} = Maximum response time by the PEO (15 hrs)
- $Q_{dilution}$ = Minimum flow rate to dilute the SFP below 700 ppm (467,873 gpm)

Therefore, the minimum flow rate that could dilute the SFP from 2300 ppm to 700 ppm in 15 hours is 520 gpm. Note that the SFP would need to fill to overflowing prior to applying the feed and bleed methodology, however, the outlined approach is conservative since it results in boron exiting the SFP immediately.

4. Dilution Source Path Evaluation

This section evaluates the potential for dilution of the SFP from the SFP Cooling System and from external sources within the SFP building.

4.1. Unintentional Overfilling from Primary Grade Water (PGS) System

The rated flow from the PGS pumps is 450 gpm combined which is less than the maximum allowable dilution flow rate of 520 gpm (Section 3.3). Since a flow rate of 520 gpm dilutes the SFP to 700 ppm in 15 hours, a flow rate of 450 gpm would take 17.33 hours to dilute the pool the same amount.

$$\frac{15 \text{ hrs} * 520 \text{ gpm}}{450 \text{ gpm}} = 17.33 \text{ hours}$$

4.2. Unintentional overfilling from normally isolated Service Water System

Pipe 3-SFC-004-41-3(Z-), which provides access to the SFP from the Service Water System, is only opened during emergency operation. If standard operating procedures fail, this method is deemed acceptable for filling the SFP. Maximum operating conditions for the Service Water System are listed as 66 psig and 95/80°F per FSAR Tables 3.6-4 and 3.6-3 (Ref. 1). Using the methodology described in Section 5, the Service Water flow rate to the SFP is about 4000 gpm based on an open ended pipe with a driving pressure of 66 psig and a loss coefficient of 1 (pipe exit, Ref. 6). However, this method will only be used in emergency situations. Additionally, the water level of the SFP is monitored during this process of filling the pool and it is unreasonable for the PEO to be away from the pool for an extended period of time in an emergency situation. Therefore, it is not credible for the Service Water System to dilute the SFP beyond the critical lower boron limit of 700 ppm.

4.3. SFP Cooling System Heat Exchanger Tube Rupture

The SFP cooling heat exchangers have component cooling water (CCP) on the shell side and SFP Fuel Pool Cooling water on the tube side. Conservatively, the CCP (non-borated) water is considered to flow directly from high CCP pressure to atmosphere, giving the maximum possible flow through a potential tube break. FSAR Table 3.6-4 (Ref. 1) indicates that the CCP system is moderate energy and has maximum operating conditions of 186 psig and 137°F in the fuel building. However, for conservatism, the maximum CCP design pressure and temperature

(250 psig and 150°F) are used for this calculation. This state point is used with Equations 5.1 and 5.2 with a tube I.D. of 5/8". This produces a flow of 152 gpm per ruptured tube. Thus, 4 tubes would have to break and leak out directly to atmospheric conditions in the SFP to exceed the maximum allowable dilution flow rate of 520 gpm (Section 3.3). Assuming one ruptured tube, a flow rate of 152 gpm would take 51 hours to dilute the SFP to 700 ppm.

$$\frac{15 \text{ hrs} * 520 \text{ gpm}}{152 \text{ gpm}} = 51 \text{ hours}$$

4.4. Excess Demineralized Water Addition from Rinsing Transfer Cask and Equipment

Per procedure, purification demineralizer flow shall not exceed 60 gpm which is less than the maximum allowable dilution flow rate of 520 gpm (Section 3.3). A flow rate of 60 gpm would take 130 hours to dilute the SFP to 700 ppm.

$$\frac{15 \text{ hrs} * 520 \text{ gpm}}{60 \text{ gpm}} = 130 \text{ hours}$$

4.5. Unintentional overfilling using RWST connection to SFP

The minimum RWST borated concentration is 2700 ppm per Table 6.3-9 of the FSAR (Ref. 1) and would not dilute the SFP, as the RWST system minimum borated concentration of 2700 ppm is higher than the initial SFP boron concentration of 2300 ppm and the proposed T.S. limit of 2600 ppm.

5. Pipe Breaks and Leaks

This section evaluates potential pipe breaks that are in close proximity to the SFP.

The lines in piping systems near the SFP are considered moderate energy lines since their design pressure and temperature (or maximum normal operating pressure and temperature for the Hot Water Pre-Heating system) are both less than 200°F and 275 psig (high energy designation requires excess of 275 psig or 200°F; UFSAR Section 3.6.1.1.2, Ref. 1). In a moderate energy piping system, through-wall leakage cracks are modeled for these systems instead of double ended guillotine breaks per FSAR Sections 3.6.2.1.2 and 3.6.2.1.3 (Ref. 1). The area of a crack is considered equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. For conservatism, the outside diameter of the pipe is used in this calculation as the diameter is in the numerator, and therefore a larger value is more conservative. The flow rate through this crack area can be calculated using Equation 3-14 of Crane Technical Paper 410 (Ref. 6).

$$Q = \sqrt{\frac{\Delta P * d^4}{0.00001799 * K * \rho}} \quad (5.1)$$

Where:

- Q = Flow rate through the opening (gpm)
- ΔP = Pressure differential across the opening (psi)
- d = equivalent hydraulic diameter of round break area (in). Defined in Eq. 5.2.
- K = Loss coefficient of the pipe exit, in this evaluation a value of 1.5 (-) is used
(summation of exit and entrance losses from Ref. 6)
- ρ = Density of water before the pipe exit (lb/ft³)

$$d = \frac{4 * L * W}{2 * L + 2 * W} \quad (5.2)$$

Where:

- L = Length of rectangular crack (in)
- W = Width of rectangular crack (in)

5.1. Component Cooling, Fire Protection, and Hot Water Pre-Heating Water Line Breaks

The CCP, FPW, and HVG pipes are classified as moderate energy lines. Using Equations 5.1 and 5.2, the maximum flow value from these lines is 23.3 gpm which results from a postulated crack in line 3-FPW-006-252-4. A flow rate of 23.3 gpm would take approximately 335 hours to dilute the SFP to 700 ppm.

$$\frac{15 \text{ hrs} * 520 \text{ gpm}}{23.3 \text{ gpm}} = 335 \text{ hours}$$

5.2. Hot Water Heating (HVH) Line Break

The HVH system has a fixed initial inventory in the lines as well as a hot water expansion tank since it is a closed loop system. A break in this system would first release inventory and then fill the system with make-up pump flow.

To maintain 700 ppm of boron in the SFP, the maximum permissible amount of unborated water in the 15 hour period between PEO shift checks is 467,873 gal (Section 3.3). The hot water heating make-up pump has an approximate runout flow of 200 gpm. If the pump runs continuously for 15 hours, the total theoretical flow passed through is 180,000 gal (= 200 gpm * 15 hr * 60 min/hr). The remaining unborated water necessary to dilute the SFP, providing the make-up flow goes directly to the SFP, would be 287,873 gal (= 467,873 gal - 180,000 gal) or 38,483 ft³ (using 7.48052 gal/ft³).

The volume of the HVH tank is calculated using the three following equations from Ref. 7:

$$V_{cylinder} = \pi \frac{ID^2}{4} * H \quad (5.3)$$

$$V_{cylinder} = \pi \frac{(4.5 \text{ ft})^2}{4} * 11 \text{ ft} = 175 \text{ ft}^3$$

Where:

- ID = Internal Diameter (4'-6")
- H = Length of Cylinder (11')

$$V_{Dished \text{ Head}, Ellipsoidal} = \pi R^2 \left(\frac{2a}{3} \right) \quad (5.4)$$

$$V_{Dished\ Head, Ellipsoidal} = \pi * (2.25\ ft)^2 * \frac{2 * 1.125\ ft}{3} = 12\ ft^3$$

Where:

- R = Tank Radius (2'-3")
- a = Dished head height (1'-1.5" – radius to height ratio is 2:1)

$$V_{Tank} = V_{cylinder} + 2 * V_{Dished\ Head, Ellipsoidal} \quad (5.5)$$

$$V_{Tank} = 175\ ft^3 + 2 * 12\ ft^3 = 199\ ft^3$$

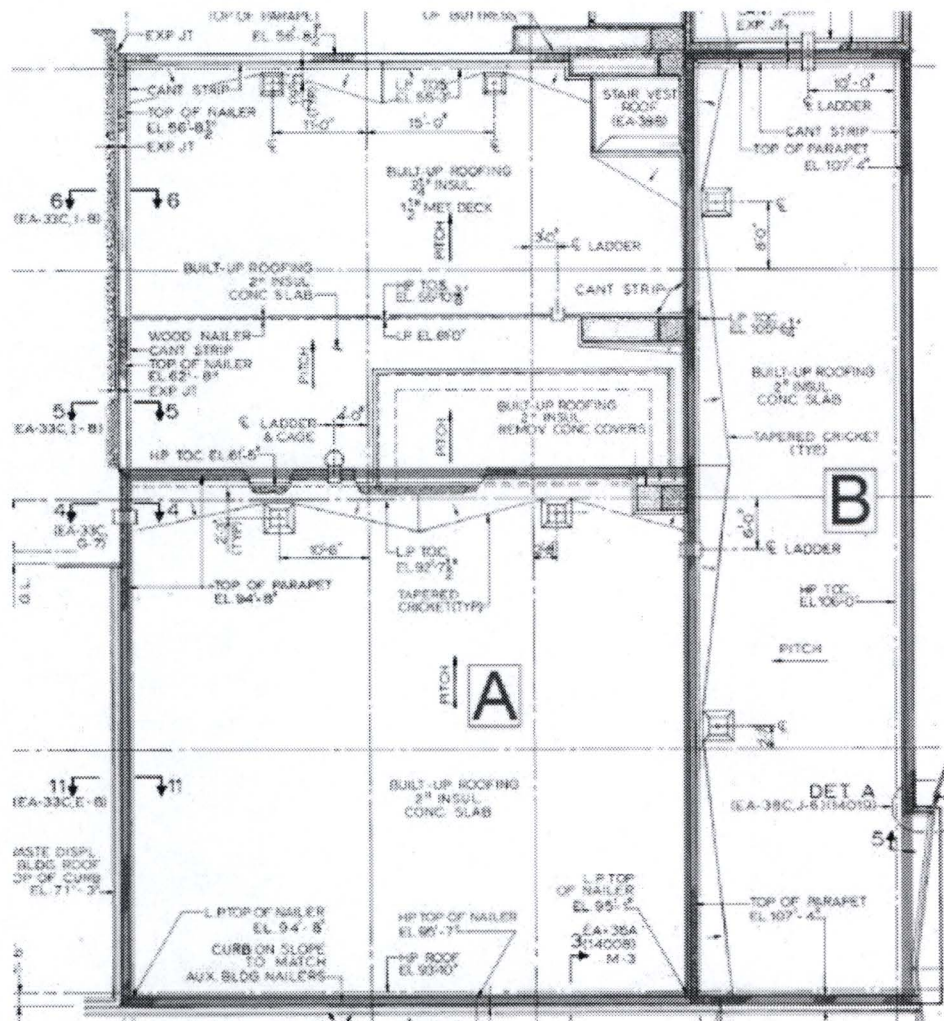
The largest pipe diameter associated with the HVH system is 10" with internal transverse area of 0.5475 ft² (at Standard Schedule) (Ref. 6). With this area, there would need to be 69,925 (=(38,483ft³-199ft³)/0.5475ft²) linear feet of 10" piping to release its inventory into the SFP to dilute to below 700 ppm.

Since 70,000 linear feet (more than 13 miles) of 10" piping does not exist in this system, it is reasonable to conclude that HVH line breaks would not result in an unacceptable bulk dilution of the SFP.

5.3. Roof Drain Line Break

There are roof drain lines that travel above the SFP to disperse environmental precipitation. These lines are shown on Figure 5-1. Using this drawing in conjunction with other drawings, roof areas to consider for rain precipitation can be determined. The roof areas considered are highlighted in Figure 5.1.

Figure 5-1: Roof Drain Surface Area



The roof areas outside the outlined sections A and B illustrated in Figure 5.1 are not considered to flow to the drain lines that flow within the SFP area because they either have a lower roof elevation, or are segregated due to a parapet. The surface areas of Section A and B are 4156.25 ft² and 2770.7 ft², respectively. The total roof drain (DNR) drainage surface area is

6927 ft². The rainfall required to collect a critical dilution volume is calculated by the following equation:

$$H_{Rain} = \frac{V_{in} (gal)}{SA_{DNR} * 7.48052 gal/ft^3} \quad (5.6)$$

The inches of rain required to dilute the SFP to 700 ppm is calculated using Equation 5.6 and is as follows:

$$H_{Rain} = \frac{467,873 gal}{6927 ft^2 * 7.48052 gal/ft^3} = 9 ft = 108 in$$

This amount of rain is far more than what would conservatively occur in 15 hours. Therefore, excessive dilution due to rain is not credible.

5.4. Fuel Pool Cooling Lines

Fuel Pool cooling lines are borated to the same value as the SFP as this system is fed from and exhausts to the pool itself. Breaks in these lines will not dilute the SFP or borate the system.

5.5. Fire Water Hose Station Pipe Leakage

A walk-down did not identify any fire water hose stations in the vicinity of the SFP. Further, fire protection pipe leakage would be bounded by the crack in the fire protection lines considered in Section 5.1.

5.6. Demineralized Water Pipe and Valve Leakage

A walk-down did not identify any demineralized water piping in the vicinity of the SFP. Further, per FSAR Section 9.2.8.2 (Ref. 1), the PGS consists of demineralized and deaerated water. As procedure fails to differentiate between PGS and Demineralized water pipes and valves, these lines are considered to be PGS lines and the leakage would be ultimately bounded by the pump flow discussed in Section 4.1.

5.7. Hot Water Heating System leakage

These lines are addressed in Section 5.2.

5.8. Domestic Water Pipe Leakage

A walk-down did not identify any domestic water pipes in the vicinity of the SFP. Further, per FSAR Section 9.2.2.1.2 (Ref. 1), the CCP system supplies make up water to reactor plant component cooling water piping and sill cock plumbing in various buildings. Thus the CCP piping has a domestic water function (plumbing). As procedure fails to differentiate between domestic water and CCP lines and valves, the domestic water is considered in the modeling of CCP lines justified in Section 5.1.

5.9. Roof Drain pipe leakage

These lines are addressed in Section 5.4.

6. Securing the Dilution Source

Two activities are needed to cease a dilution event: the dilution needs to be discovered, and then the dilution source needs to be secured. Sections 4 and 5 address the possible dilution sources and their associated flow rates. Table 6-1 shows the amount of time needed to dilute the pool to 700 ppm using the credible source flow rates discussed in the previous sections.

Table 6-1: Times to 700 ppm for Various Dilution Sources

Dilution Initialization Scenario	Flow Rate (gpm)	Time to 700 ppm (hrs)	Section Calculated
Unintentional Overfilling from Primary Grade Water System	450	17.33	4.1
SFP Heat Exchanger Tube Rupture	152	51	4.3
Excess demin water from rinsing transfer cask and equipment	60	130	4.4
CCP, FPW, and HVG Line Breaks	23.3	335	5.1

Table 6-1 shows that unintentionally overfilling the SFP from the PG water system is the bounding scenario. Using the assumptions from Sections 2.4 and 3.2, 15 hours are needed to discover the dilution event. Therefore, operations must be able to secure the PG water system in 2.33 hours.

When a fuel pool high level alarm is received locally or in the control room, station procedures direct operations to close the PG valve, PG bypass valve and to stop makeup to the spent fuel pool. While there is no formal time requirement on these procedural actions, discussions with Operations estimated that the process should take no more than 15 minutes from the time of discovery. Note that an operator will already be at the SFP since the event was assumed to be discovered via operator rounds. Since the estimated mitigation time has over 2 hours or 89% margin to the required mitigation time limit, it can be reasonably assumed that a dilution event from the PG water system can be discovered and secured before the SFP reaches 700 ppm.

The other dilution scenarios have much more time to determine and secure the source of the dilution; therefore they are bounded by the dilution scenario from the PG Water System.

7. Conclusions

This engineering analysis of potential scenarios which could dilute the boron concentration in the SFP demonstrates that sufficient time is available to detect and mitigate a boron dilution from 2300 ppm prior to reaching 700 ppm.

The systems which could dilute the spent fuel pool, either by direct connection to the spent fuel pool or by a potential pipe crack/break, have been analyzed via a bleed and feed methodology. However, in reality, the addition of unborated water to the SFP will lead to increased SFP water level and, if not controlled, an overflow of the SFP.

The ability to prevent the SFP soluble boron concentration from being diluted from 2300 ppm to a value of 700 ppm has been demonstrated by showing that each potential dilution source meets one of the following two criteria:

- Any dilution source not capable of supplying 467,873 gallons of unborated water will not be capable of diluting the SFP soluble boron concentration from 2300 ppm to 700 ppm.
- If the dilution flow rate of unborated water is < 520 gpm, then at least 15 hours will be needed for the SFP soluble boron concentration to be reduced from 2300 ppm to 700 ppm.

All dilution scenarios evaluated here will eventually cause SFP high water level alarms either detected directly by control room alarm, by the PEO detecting high SFP water levels or SFP overflow. The largest, credible dilution flow rate was 450 gpm (Unintentionally Overfilling from the Primary Grade Water System). The time to dilute the SFP from 2300 ppm to 700 ppm with a 450 gpm dilution source is 17.33 hours. Since the conservatively longest time between PEO rounds is 15 hours, operators need to be able to mitigate the dilution event within 2.33 hours. Discussions with operations estimate that securing the dilution source the dilution path would take no more than 15 minutes once discovered.

The postulated dilution break sources in the vicinity of the SFP have been conservatively evaluated. None of the sources result in a final SFP concentration less than 700 ppm of boron from an initial concentration of 2300 ppm. The major conservatisms of this evaluation are as follows:

- The proposed TS boron concentration limit is 2600 ppm rather than the 2300 ppm assumed initial condition.

- The SFP is assumed to contain the least amount of borated water possible when the dilution event initiates
 - Water level at the low level alarm set point
 - 1860 assemblies with 1860 RCCAs are stored in the SFP
 - The Spent Fuel Shipping Cask Loading Area, gate areas, Spent Fuel Shipping Cask Storage Area, and transfer tube are isolated from the SFP.
- Water level alarms do not detect the dilution event. It is assumed that this occurs if either the level indicators fail or the Bleed and Feed methodology keeps the water level constant.
- Operators are not notified of the dilution event via the Fuel Building Sump Pump auto start up.
- Operator rounds are the only way to detect a dilution event. It is assumed that no other workers or operators can discover a 467,873 gallon dilution event.
- The time interval between operator rounds is 15 hours (the maximum interval procedurally allowable).

8. References

External References:

- 1) MP3 UFSAR, Rev. 30.1, "Millstone Power Station Unit 3 Final Safety Analysis Report," 09/28/17.
- 2) MP3 T.S., Change No. 293, "Facility Operating License/Safety Technical Specifications," 03/09/17.
- 3) Air Contaminants and Industrial Hygiene Ventilation – A Handbook of Practical Calculations, Problems, and Solutions, Roger L. Wabeke, 1998 by CRC Press LLC.
- 4) Letter from J. Alan Price (DNC) to USNRC, "Millstone Power Station, Unit No. 2, Technical Specifications Change Request (TSCR) 2-10-01, Fuel Pool Requirements," November 6, 2001. (ADAMS Accession No. ML013510295).
- 5) Letter from Richard B. Ennis (NRC) to J. A. Price (DNC), "MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: SPENT FUEL POOL REQUIREMENTS (TAC NO. MB3386)," April 1, 2003. (ADAMS Accession No. ML030910485).
- 6) Crane Technical Paper No. 410, Flow of Fluids through Valves, Fittings, and Pipe, 1988.
- 7) Rules of Thumb for Chemical Engineers – fifth edition 2012, Stephen Hall.