Southern Nuclear

Regulatory Affairs

3535 Colonnade Parkway
Birmingham, AL 35243
205.992.5000 tel
205.992.7795 fax

September 21, 2022

10 CFR 50.90
NL-22-0289

Docket Nos.: 50-348
50-364

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant Units 1 and 2
License Amendment Request to
Revise Technical Specification 4.3 “Fuel Storage”
to Correct Tabulated Values from the Associated Spent Fuel Pool (SFP) Criticality Analysis

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) submits a license amendment request (LAR) to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License (NPF-2), and Unit 2 Renewed Facility Operating License (NPF-8) to revise Technical Specification (TS) 4.3, “Fuel Storage”. The purpose of this LAR is to revise TS 4.3 to correct tabulated values in the associated spent fuel pool (SFP) criticality analysis.

The Enclosure provides a description and assessment of the proposed TS changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides retyped TS pages. The TS Bases are not impacted other than changing the criticality safety analysis reference revision. Attachments 3 and 4 provide the proprietary and non-proprietary versions, respectively, of the criticality safety analysis WCAP-18414, “J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis”. Attachment 5 provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-22-039, accompanying Affidavit, and Proprietary Information Notice.

Approval of the proposed amendments is requested within one year. The proposed changes will be implemented within 60 days of issuance of the amendment.

This letter contains no NRC commitments.

In accordance with 10 CFR 50.91, SNC is notifying the state of Alabama of this license amendment request by transmitting a copy of this letter to the designated state official.

If you have any questions, please contact Amy Chamberlain at 205.992.6361.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 21, 2022.

C. A. Gayheart
Fleet Regulatory Affairs Director
Southern Nuclear Operating Company

CAG/efb/was/cbg

Enclosure: Description and Assessment of the Proposed Changes
Attachment 1: Technical Specification Page Markups
Attachment 2: Retyped Technical Specification Pages
Attachment 3: WCAP-18414-P, R3 "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary Version)
Attachment 4: WCAP-18414-NP, R3 "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Non-proprietary Version)
Attachment 5: Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-22-039, accompanying Affidavit, and Proprietary Information Notice

cc: NRC Regional Administrator
NRC NRR Project Manager – Farley 1&2
NRC Senior Resident Inspector – Farley 1 & 2
Alabama - State Health Officer for the Department of Public Health
SNC Document Control R-Type: CFA04.054

Southern Nuclear Operating Company
Joseph M. Farley Nuclear Plant – Units 1 and 2

License Amendment Request to
Revise Technical Specification 4.3 “Fuel Storage”
to Correct Tabulated Values from the Associated Spent Fuel Pool (SFP) Criticality Analysis

Enclosure

Description and Assessment of the Proposed Changes

ENCLOSURE

Description and Assessment of the Proposed Changes

Subject: Joseph M. Farley Nuclear Plant Units 1 and 2 Submittal of License Amendment Request to Revise Technical Specification (TS) TS 4.3 "Fuel Storage" to Correct Tabulated Values from the Associated Spent Fuel Pool (SFP) Criticality Analysis

1. SUMMARY DESCRIPTION

2. DETAILED DESCRIPTION
   - 2.1 System Design and Operation
   - 2.2 Current Technical Specifications Requirements
   - 2.3 Reason for the Proposed Change
   - 2.4 Description of the Proposed Change

3. TECHNICAL EVALUATION

4. REGULATORY EVALUATION
   - 4.1 Applicable Regulatory Requirements/Criteria
   - 4.2 Precedent
   - 4.3 No Significant Hazards Consideration Determination Analysis
   - 4.4 Conclusions

5. ENVIRONMENTAL CONSIDERATION

6. REFERENCES

ATTACHMENTS
1. Technical Specification Page Markups
2. Retyped Technical Specification Pages
3. WCAP-18414-P, R3 "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary Version)
4. WCAP-18414-NP, R3 "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Non-proprietary Version)
5. Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-22-039, accompanying Affidavit, and Proprietary Information Notice

1. SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS) to incorporate changes to TS 4.3 “Fuel Storage” to correct tabulated values in the associated spent fuel pool (SFP) criticality analysis.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

The SFP is made up of one fuel storage rack design (region) that maintains 10.75-inch center-to-center spacing between spent fuel assemblies. The Farley Units 1 & 2 SFPs each consist of two 6 x 7, nineteen 7 x 7, and seven 7 x 8 storage racks. The spent fuel racks are freestanding and are free to move on the pool liner floor during a seismic event.

The revised SFP criticality safety analysis, “J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis” (WCAP-18414-P, R3 (Proprietary Version) and WCAP-18414-NP, R3 (Non-Proprietary Version) evaluates the SFP storage racks for the placement of fuel within the storage arrays defined in the technical specifications. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time) for assemblies which have been operated in the reactor.

2.2 Current Technical Specifications Requirements

Technical Specification (TS) 4.3 “Fuel Storage” for FNP Unit 1 and Unit 2 is based upon the SFP criticality safety analysis.

2.3 Reason for the Proposed Change

A review of WCAP-18414-P, Revision 0 discovered a misapplication of depleted isotopics to standard fuel in storage configurations with burnup requirements in “Burnup Bin 1.” As a result, Table 6-7 Burnup Requirement Coefficients were revised for 0 and 5 year decay times to reflect a correction to the applied depleted isotopics and ultimately the fitting coefficients at 0 and 5 years for Standard Fuel Assembly (STD)/ Robust Fuel Assembly (RFA) fuel. The corresponding Table 6-8 Example STD/RFA Burnup Requirements were revised accordingly using the revised STD/RFA burnup requirement coefficients from Table 6-7. The corresponding bias and uncertainty rackup table has not been revised as it remains conservative for this issue.

An extent-of-condition review of calculations discovered that the burnup measurement uncertainty for 4.0 weight percent U-235 Optimized Fuel Assembly (OFA) fuel in WCAP-18414-P Revision 0 Table 5-7 was found to be incorrectly calculated. Additionally, Table 5-7 was revised to reflect the response to an NRC request for additional information (RAI) on Revision 0 of WCAP-18414-P (See Reference 1) which included a revised temperature bias and rackup for the associated maximum fresh enrichment without burnup.

The changes to WCAP-18414-P necessitated a revision to TS 4.3 Tables 4.3-3 and 4.3-4.

2.4 Description of the Proposed Changes

TS 4.3, "Fuel Storage"

The proposed changes to TS 4.3 are based on the revised SFP criticality safety analysis. Specifically, the proposed changes to TS 4.3 include:

- The corrected Table 4.3-3 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for Standard Fuel Assembly (STD)/Robust Fuel Assembly (RFA) fuel.
- The corrected Table 4.3-4 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for Optimized Fuel Assembly (OFA) fuel.

Redline/strikeout copies of TS 4.3 are included in Attachment 1. Retyped copies of TS 4.3 are provided in Attachment 2.

3. TECHNICAL EVALUATION

Westinghouse WCAP-18414-P,-NP, R3 J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis are included in Attachment 3 and Attachment 4 and will be referred to as the revised SFP criticality safety analysis in this technical evaluation. Relevant revised pages include: pages ii, 3-3, 4-12, 5-22, 5-23, 6-5, 6-7, and 6-8.

A review of WCAP-18414-P, Revision 0 discovered a misapplication of depleted isotopics for STD/RFA fuel in storage configurations with burnup requirements in "Burnup Bin 1." As a result, WCAP-18414-P Table 6-7 STD/RFA Burnup Requirement Coefficients were revised for 0 and 5 year decay times to reflect a correction to the applied depleted isotopics and ultimately the fitting coefficients at 0 and 5 years for STD/RFA fuel. The corresponding Table 6-8 Example STD/RFA Burnup Requirements were revised accordingly using the revised STD/RFA burnup requirement coefficients from Table 6-7. The corresponding bias and uncertainty rackup table has not been revised as it remains conservative for this issue.

TS Table 4.3-3 was corrected to reflect the values of Table 6-7 of WCAP-18414-P at 0 and 5 years decay time for STD /RFA Fuel Category 3. The corrected TS Table 4.3-3 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for STD/RFA fuel. There is no current STD/RFA fuel inventory at Farley Units 1 and 2 that have between 0 and 5 years of decay time.

An extent-of-condition review of calculations discovered that the burnup measurement uncertainty for 4.0 weight percent U-235 OFA fuel in Table 5-7 was incorrectly calculated in WCAP-18414-P Revision 0. The target k-eff for 4.0 weight percent U-235 in Table 5-7 was revised to correct the burnup measurement uncertainty. The burnup measurement uncertainty value affects the fitting coefficients (WCAP-18414-P Table 6-13) and by extension the calculated example burnup requirements (WCAP-18414-P Table 6-14). The changes to burnup measurement uncertainty in Table 5-7 for 4.0 weight percent U-235 OFA fuel resulted in WCAP-18414-P Tables 6-13 and 6-14 being revised to reflect the effect on fitting coefficients for Fuel Category 4 OFA fuel.

Additionally, Table 5-7 was revised to reflect the response to NRC RAI No. 5 for WCAP-18414 (See Reference 1):

> WCAP-18414-P, Section 5.2.3.1.14, "SFP [Spent Fuel Pool] Temperature Bias," discusses the determination of the most reactive temperature in the Farley SFP. With respect to the SFP Temperature Bias analysis, please provide the SFP keff at each temperature analyzed.

The increase in the SFP temperature bias for the cases that impacted Tables 5-6 and 5-7 from WCAP-18414 Revision 0 was provided with the response to NRC RAI No. 5. The changes to Tables 5-6 and 5-7 from the RAI response were also included in the revision to WCAP-18414.

TS Table 4.3-4 was corrected to reflect the values of Table 6-13 of WCAP-18414-P, R3 at all decay times for OFA Fuel Category 4. The corrected TS Table 4.3-4 provides the fitting coefficients to calculate the minimum required fuel assembly burnup for fuel categories 3 and 4 for OFA fuel.

Conclusion

The proposed changes to TS 4.3 correct tabulated values and allow for continued safe storage of spent fuel at Farley Units 1 and 2. The existing stored fuel is not impacted by the SFP criticality analysis revision to correct tabulated values, and administrative controls are in place to ensure the stored fuel stays within a safe storage configuration.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

The requirements for system operability during movement of irradiated fuel are included in the TSs in accordance with 10 CFR 50.36(c)(2), Limiting Conditions for Operation. As required by 10 CFR 50.36(c)(4), design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36. This amendment request concerns 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(4).

General Design Criterion (GDC) 61 – *Fuel storage and handling and radioactivity control*, "The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

GDC 62 – *Prevention of criticality in fuel storage and handling,* “Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.”

Additional guidance is available in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," particularly Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, issued March 2007. Section 9.1.1 provides the existing recommendations for performing the review of the nuclear criticality safety analysis of SFPs.

The proposed changes ensure that compliance is maintained with these regulations, that equipment functions as required, and fuel storage operations are conducted safely. The existing stored fuel is not impacted by the SFP criticality analysis revision to correct tabulated values, and administrative controls are in place to ensure the stored fuel stays within a safe storage configuration.

4.2 Precedent

None.

4.3 No Significant Hazards Consideration Determination Analysis

Southern Nuclear Operating Company (SNC) requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS) to incorporate changes to TS 4.3 “Fuel Storage” to correct tabulated values in the associated spent fuel pool (SFP) criticality analysis.

SNC has evaluated whether a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

   Response: No.

   The proposed amendment was evaluated for impact on the following criticality events and accidents and no impacts were identified: (1) loss of spent fuel pool cooling system, (2) dropping a fuel assembly into an already loaded storage cell, and (3) the misloading of a single fuel assembly or multiple fuel assemblies into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

   Operation in accordance with the proposed amendment will not change the probability of a loss of spent fuel pool cooling because the changes in the criticality safety analysis have no bearing on the systems, structures, and components involved in initiating such an event. A criticality safety analysis of the limiting fuel loading configuration confirmed that the condition would remain subcritical for a range of normal and accident conditions. The effects of the accident conditions are bounded by the multiple fuel assembly misload accident.

Operation in accordance with the proposed amendment will not change the probability of a fuel assembly being dropped into an already loaded storage cell because fuel movement will continue to be controlled by approved fuel handling procedures. The consequences of a dropped fuel assembly are not changed; there will continue to be significant separation between the dropped fuel assembly and the active regions of the fuel assemblies. The effects of this accident are bounded by the multiple fuel assembly misload accident.

Operation in accordance with the proposed amendment will not change the probability of a fuel assembly misloading because fuel movement will continue to be controlled by approved fuel selection and fuel handling procedures. These procedures continue to require identification of the initial and target locations for each fuel assembly and fuel assembly insert that is moved. The consequences of a fuel misloading event are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for the multiple fuel assembly misload accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The potential for criticality in the spent fuel pool is not a new or different type of accident. Storage configurations allowed by Technical Specifications 3.7.15 and 4.3 have been analyzed to demonstrate that the pool remains subcritical.

The new criticality safety analysis includes analysis of a multiple misload accident scenario; only single misload events were previously analyzed. The inclusion of this analysis does not imply the creation of the possibility of a new accident, but simply expands the boundaries of the analyzed accident conditions to ensure that all potential accidents are properly considered.

There is no significant change in plant configuration, equipment design or usage of plant equipment. The revised SFP criticality safety analysis assures that the pool will continue to remain subcritical.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change was evaluated for its effect on current margins of safety as they relate to criticality. The revised SFP criticality safety analysis confirms that operation in accordance with the proposed amendment continues to meet the required subcriticality margin.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents 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 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.

6. REFERENCES

1. Agencywide Documents and Access Management System (ADAMS) Package Accession No. ML19275E393, License Amendment Request to Update the Spent Fuel Pool Criticality Safety Analysis, dated September 30, 2019, as supplemented by Response to RAI dated April 13, 2020 (ADAMS Accession No. ML20104C140).

Attachment 1

Existing Technical Specification Page Markups

Pages: 4.0-9
4.0-10
Insert A for 4.0-10 (1 Page)

Table 4.3-3

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for STD/RFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.3997 ~~0.2251~~	-4.4670 ~~-2.5199~~	28.2780 ~~21.4065~~	-44.1204 ~~-36.6115~~
	5	~~0.3002~~ 0.3637	~~-3.4376~~ -4.1462	~~24.0978~~ 26.6011	~~-38.9002~~ -41.6405
	10	0.1856	-2.3309	20.2704	-34.6503
	15	0.0892	-1.3905	17.0683	-31.1550
	20	0.0388	-0.9253	15.5082	-29.4500
4	0	-0.6112	4.6655	6.7127	-21.8911
	5	-0.3326	2.0713	12.8468	-26.1880
	10	-0.1305	0.0505	18.3242	-30.7080
	15	0.1360	-2.6856	26.5239	-38.3300
	20	0.2321	-3.7177	29.5977	-41.1200

Table 4.3-4

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for OFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.1692	-1.8852	18.5219	-32.7830
	5	0.0191	-0.4154	13.4482	-27.1777
	10	-0.0705	0.4300	10.5987	-24.0722
	15	-0.1420	1.1146	8.2825	-21.5440
	20	-0.1959	1.6375	6.5093	-19.6130
4	0	0.4957	-6.0715	37.2851	-49.1282
	5	0.7476	-8.7581	45.3241	-56.5172
	10	Replace with Insert A		50.3246	-61.0800
	15	1.0799	-12.2326	55.7508	-66.1820
	20	1.2541	-13.9154	60.5977	-70.5720

Table 4.3-4 Insert A

4	0	**0.3726**	**-4.8740**	**33.7329**	**-45.9288**
	5	**0.6544**	**-7.8532**	**42.6520**	**-54.1346**
	10	**0.8557**	**-9.9883**	**49.1073**	**-60.1446**
	15	**0.9692**	**-11.1551**	**52.5353**	**-63.2522**
	20	**1.1873**	**-13.2641**	**58.6586**	**-68.8379**

Attachment 2

Retyped Technical Specification Pages

Table 4.3-3

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for STD/RFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.3997	-4.4670	28.2780	-44.1204
	5	0.3637	-4.1462	26.6011	-41.6405
	10	0.1856	-2.3309	20.2704	-34.6503
	15	0.0892	-1.3905	17.0683	-31.1550
	20	0.0388	-0.9253	15.5082	-29.4500
4	0	-0.6112	4.6655	6.7127	-21.8911
	5	-0.3326	2.0713	12.8468	-26.1880
	10	-0.1305	0.0505	18.3242	-30.7080
	15	0.1360	-2.6856	26.5239	-38.3300
	20	0.2321	-3.7177	29.5977	-41.1200

Table 4.3-4

Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Initial Enrichment (En) for OFA Fuel

Fuel Category	Decay Time (years)	Coefficients			
		A_1	A_2	A_3	A_4
3	0	0.1692	-1.8852	18.5219	-32.7830
	5	0.0191	-0.4154	13.4482	-27.1777
	10	-0.0705	0.4300	10.5987	-24.0722
	15	-0.1420	1.1146	8.2825	-21.5440
	20	-0.1959	1.6375	6.5093	-19.6130
4	0	0.3726	-4.8740	33.7329	-45.9288
	5	0.6544	-7.8532	42.6520	-54.1346
	10	0.8557	-9.9883	49.1073	-60.1446
	15	0.9692	-11.1551	52.5353	-63.2522
	20	1.1873	-13.2641	58.6586	-68.8379

Attachment 4

WCAP-18414-NP, R3
J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis
(Non-proprietary Version)

WCAP-18414-NP
Revision 3

August 2022

J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

WCAP-18414-NP
Revision 3

J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis

Christopher M. Briggs*
Core Engineering & Software Development

August 2022

Reviewers: Michael T. Wenner*
Core Engineering & Software Development

Approved: Stephanie Y. Harsche*, Manager
Core Engineering & Software Development

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

REVISION HISTORY

Revision	Description and Impact of the Change	Date
0	Original Issue	09/2019
1	Revision 1 contains multiple corrections. • Tables 5-6 and 5-7 were updated to reflect the Reference 18 U.S. NRC RAI responses which indicated a revised temperature bias and rackup for the associated maximum fresh enrichment without burnup. • Table 6-7 Burnup Requirement Coefficients were updated for 0 and 5 year decay times as well as the corresponding Example Burnup Requirements in Table 6-8 when considering the burnup dependent nature of the bounding Criticality Fuel Design for Burnup Bin 1 with STD/RFA fuel. The corresponding bias and uncertainty rackup is conservative and has not been updated. • Table 6-12 is updated to reflect the coefficients in Table 6-11 (which is unchanged) as the Example Burnup Requirements were incorrectly calculated. • The burnup measurement uncertainty term is updated in Tables 6-13 and 6-14 for 4.0 wt. % OFA fuel. Table 5-7 is updated to reflect the updated burnup measurement uncertainty value itself in the corresponding rackup. • Table 5-8 contains an update to the 2.15 wt. % total bias and uncertainty term. • Table 3-3 (footnote) and Section 4.3.2 contain updated text to explain the burnup dependent bounding Criticality Fuel Design for STD/RFA fuel within Burnup Bin 1. • Additional minor editorial updates are included.	04/2022
2	Revision 2 contains the changes below. Note that the revision bars from Revision 1 and Revision 2 are shown in Revision 2. • The current reactor power was updated in Table 3-1. • Figure 5-5 was updated to correct an unacceptable configuration. • Minor rounding updates were made to Tables 6-6, 6-8, 6-10, 6-12, and 6-14.	06/2022
3	Revision 3 includes the changes below. Note that the revision bars from Revision 1, Revision 2, and Revision 3 are shown in Revision 3. • Remove Appendix A. • Add Appendix A of Revision 0 of WCAP-18414-NP as Reference 19 in Section 7. • Update any text that refers to Appendix A to refer to Reference 19.	08/2022

TRADEMARK NOTICE

TABLE OF CONTENTS

LIST OF TABLES v

LIST OF FIGURES vii

LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS viii

1 INTRODUCTION 1-1

2 OVERVIEW 2-1
- 2.1 ACCEPTANCE CRITERIA 2-1
- 2.2 DESIGN APPROACH 2-1
- 2.3 COMPUTER CODES 2-2
  - 2.3.1 Two-Dimensional Transport Code PARAGON 2-2
  - 2.3.2 Scale Code Package 2-3
  - 2.3.3 Scale 238 Group Cross-Section Library 2-5

3 FARLEY UNITS 1 & 2 NUCLEAR POWER PLANT 3-1
- 3.1 REACTOR DESCRIPTION 3-1
- 3.2 FUEL STORAGE DESCRIPTION 3-4

4 DEPLETION ANALYSIS 4-1
- 4.1 DEPLETION MODELING SIMPLIFICATIONS & ASSUMPTIONS 4-1
- 4.2 FUEL DEPLETION PARAMETER SELECTION 4-2
  - 4.2.1 Fuel Isotopic Generation 4-2
  - 4.2.2 Reactor Operation Parameters 4-2
  - 4.2.3 Axial Profile Selection 4-7
  - 4.2.4 Burnable Absorber Usage 4-9
  - 4.2.5 Fuel Assembly Physical Changes with Depletion 4-10
- 4.3 DESIGN BASIS FUEL SELECTION 4-10
  - 4.3.1 Fuel Design and Management Modeling Considerations 4-10
  - 4.3.2 Reactivity Comparison Methodology 4-11
- 4.4 FINAL DEPLETION PARAMETERS 4-13

5 CRITICALITY ANALYSIS 5-1
- 5.1 KENO MODELING APPROACH, SIMPLIFICATIONS & ASSUMPTIONS 5-1
  - 5.1.1 Description of Fuel Assembly and Storage Racks for KENO 5-3
  - 5.1.2 Impact of Structural Materials on Reactivity 5-5
- 5.2 KENO MODELING ANALYSIS 5-6
  - 5.2.1 Array Descriptions and Fuel Categories 5-6
  - 5.2.2 Target k_{eff} Calculation Description 5-11
  - 5.2.3 Bias & Uncertainty Calculations 5-11
- 5.3 INTERFACE CONDITIONS 5-26
- 5.4 NORMAL CONDITIONS 5-27
  - 5.4.1 Type 1 Normal Conditions 5-27
  - 5.4.2 Type 2 Normal Conditions 5-27

5.4.3 Type 3 Normal Conditions 5-29
5.4.4 Type 4 Normal Conditions 5-30
5.4.5 Type 5 Normal Conditions 5-31
5.5 SOLUBLE BORON CREDIT 5-31
5.5.1 Soluble Boron Requirements for Normal Conditions 5-32
5.5.2 Soluble Boron Requirements for Accident Conditions 5-33
5.5.3 Soluble Boron Requirements....5-36
5.6 RODDED OPERATION....5-37

6 ANALYSIS RESULTS & CONCLUSION 6-1
6.1 BURNUP AND IFBA REQUIREMENTS FOR STORAGE ARRAYS....6-1
6.2 ANALYSIS AREA OF APPLICABILITY 6-9
6.3 SOLUBLE BORON CREDIT 6-13

7 REFERENCES 7-1

LIST OF TABLES

Table 2-1 Isotopes Used in the Nuclear Criticality Safety Analysis 2-3
Table 3-1 Reactor General Specifications....3-1
Table 3-2 Fuel Design Mechanical Specifications....3-2
Table 3-3 Non-Mechanical Specifications and Operating History 3-3
Table 3-4 Fuel Storage Rack Specifications 3-6
Table 3-5 Fuel Rod Storage Canister 3-7
Table 3-6 Loose Pellet Transport Canister 3-7
Table 4-1 Cycle Average Soluble Boron Concentration (ppm)....4-4
Table 4-2 Fuel Design Operating Power....4-6
Table 4-3 Burnable Absorber Specifications....4-9
Table 4-4 Design Basis Comparison Modeled Fuel Design Parameters....4-11
Table 4-5 Parameters Used in Depletion Analysis....4-13
Table 4-6 Limiting Axial Burnup and Moderator Temperature Profiles....4-14
Table 5-1 IFBA Criticality Modeling Specifications 5-2
Table 5-2 Design Basis Fuel Assembly Design Modeling Parameters 5-4
Table 5-3 Fuel Categories Ranked by Reactivity....5-8
Table 5-4 Biases & Uncertainties for Array A with STD Fuel....5-19
Table 5-5 Biases & Uncertainties for Array A with OFA Fuel....5-20
Table 5-6 Biases & Uncertainties for Array B with STD Fuel....5-21
Table 5-7 Biases & Uncertainties for Array B for OFA Fuel 5-22
Table 5-8 Biases & Uncertainties for Array C with STD Fuel....5-23
Table 5-9 Biases & Uncertainties for Array C with OFA Fuel....5-24
Table 5-10 Biases & Uncertainties for Array D with STD Fuel 5-25
Table 5-11 [ ]^{a,c} 5-33
Table 5-12 [ ]^{a,c}....5-33
Table 5-13 [ ]^{a,c}....5-34
Table 5-14 [ ]^{a,c}....5-36
Table 6-1 Fuel Category 2: STD/RFA IFBA Fitting Coefficients....6-2
Table 6-2 Fuel Category 2: Example STD/RFA IFBA Requirements (# of IFBA Rods)....6-2

Table 6-3 Fuel Category 2: OFA IFBA Fitting Coefficients 6-3
Table 6-4 Fuel Category 2: Example OFA IFBA Requirements (# of IFBA Rods) 6-3
Table 6-5 Fuel Category D: STD/RFA Burnup Requirement Coefficients 6-4
Table 6-6 Fuel Category D: Example STD/RFA Burnup Requirements (GWd/MTU) 6-4
Table 6-7 Fuel Category 3: STD/RFA Burnup Requirement Coefficients 6-5
Table 6-8 Fuel Category 3: Example STD/RFA Burnup Requirements (GWd/MTU) 6-5
Table 6-9 Fuel Category 3: OFA Burnup Requirement Coefficients 6-6
Table 6-10 Fuel Category 3: Example OFA Burnup Requirements (GWd/MTU) 6-6
Table 6-11 Fuel Category 4: STD/RFA Burnup Requirement Coefficients 6-7
Table 6-12 Fuel Category 4: Example STD/RFA Burnup Requirements (GWd/MTU) 6-7
Table 6-13 Fuel Category 4: OFA Burnup Requirement Coefficients 6-8
Table 6-14 Fuel Category 4: Example OFA Burnup Requirements (GWd/MTU) 6-8
Table 6-15 Criticality [ ]a,c 6-10

LIST OF FIGURES

Figure 3-1 Farley SFP (Unit 1) Layout....3-4

Figure 3-2 Farley SFP (Unit 2) Layout....3-5

Figure 5-1 KENO Array Rack Model Planar (x-y) View: Top Left Array A, Top Right Array B, Bottom Left Array C, Bottom Right Array D (All Models: x-y Periodic Boundary Conditions)....5-3

Figure 5-2 Allowable Storage Arrays....5-9

Figure 5-3 Currently Used Damaged Fuel Assembly Configuration (Farley Unit 1)....5-10

Figure 5-4 [ ][a,c]....5-17

Figure 5-5 [ ][a,c]....5-26

Figure 5-6 Schematic View of Modeled Failed Fuel Rod Storage Canister....5-29

Figure 5-7 [ ][a,c]....5-30

LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS

1-D	One-Dimensional
2-D	Two-Dimensional
3-D	Three-Dimensional
AEG	Average Energy Group of Neutrons Causing Fission
AoA	Area of Applicability
at%	Atom Percent
B&U	Sum of Biases and Uncertainties
B&W	Babcock and Wilcox
BA	Burnable Absorber
BONAMI	Bondarenko AMPX Interpolator
Boraflex	Neutron Absorber Material Comprised of Silicone Polymer and Boron Carbide Powder
C.E.A.	Commissariat à l'Energie Atomique et aux Energies Alternatives
Decay time	Post-irradiation cooling time
EALF	Energy of Average Lethargy causing Fission
En	Enrichment
ENDF/B	Evaluated Nuclear Data File
EPRI	Electric Power Research Institute
FHE	Fuel Handling Equipment
FOSAR	Foreign Object Search and Retrieval
FRSC	Fuel Rod Storage Canister
GT	Guide Tube
GWd	Gigawatt-days
HTC	Haut Taux de Combustion
ID	Inner Dimension
IFBA	Integral Fuel Burnable Absorber
ISG	Interim Staff Guidance
IT	Instrumentation Tube
k_{eff}	Effective neutron multiplication factor
LWR	Light Water Reactor
MTU	Metric Ton Uranium
MWt	Megawatts-thermal
NPM	Non-Parametric Margin
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Lab
Optimized ZIRLO	Optimized ZIRLO® High Performance Cladding Material
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SFP	Spent Fuel Pool
SRSC	Service de Recherche en Sûreté Criticité, now called Service de Recherche en Neutronique et Sûreté Criticité
SS	Stainless Steel

LIST OF ACRONYMS, INITIALISMS, AND TRADEMARKS (cont.)

STD	Standard Fuel Assembly
TD	Percentage of Theoretical Density
WABA	Wet Annular Burnable Absorber
Westinghouse	Westinghouse Electric Company LLC
wt%	Weight Percent
ZIRLO	ZIRLO® High Performance Fuel Cladding Material

1 INTRODUCTION

The purpose of this report is to document the criticality safety analysis performed to support the operation of the J. M. Farley Nuclear Power Plant Units 1 and 2 (hereafter, Farley Units 1 & 2) spent fuel pools (SFPs). The report considers past, current, and planned future operating history and fuel design of Farley Units 1& 2.

The main report details the SFP criticality safety analysis. Reference 19 details the validation of the code used for pool eigenvalue calculations.

2 OVERVIEW

The existing SFP storage racks are evaluated for the placement of fuel within the storage arrays described in Section 5.2.1. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time) for assemblies which have been operated in the reactor. Fuel assemblies which have not been operated in the reactor may take credit for the presence of zirconium diboride (IFBA) (hereafter referred to as IFBA). While the Farley Units 1 & 2 SFP storage racks may contain Boraflex® absorber inserts, no credit is taken for the presence of Boraflex absorber. Additionally, credit is taken for the presence of soluble boron in the SFPs.

2.1 ACCEPTANCE CRITERIA

This SFP criticality safety analysis ensures that the SFPs operate within the bounds discussed here.

1. The effective neutron multiplication factor (k_{eff}) of all permissible storage arrangements at a soluble boron concentration of 0 parts per million (ppm) shall be less than 1.0 including a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

2. The k_{eff} of all permissible storage arrangements when crediting soluble boron shall yield results not exceeding 0.95, including a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

3. The k_{eff} when crediting soluble boron shall not exceed 0.95 under all postulated accident conditions, including a margin for all applicable biases and uncertainties with 95 percent probability at a 95 percent confidence level.

2.2 DESIGN APPROACH

For the SFPs, compliance is demonstrated by establishing the minimum burnup requirements as a function of enrichment and decay time and minimum number of unirradiated IFBA rods as a function of enrichment for storage arrays A, B, C, and D seen in more detail in Figure 5-2. The fuel storage arrays have been analyzed to determine separate burnup requirements for two major fuel designs considered, the Standard Fuel Assembly (STD), and the Optimized Fuel Assembly (OFA). Note that the burnup requirements developed for the STD fuel design are applicable to the Robust Fuel Assembly (RFA) design since their neutronically important characteristics are the same.

A conservative combination of best estimate and bounding values have been selected as input for modeling in this analysis to ensure that fuel represented by the proposed Farley Units 1 & 2 SFP storage Technical Specifications is less reactive than the fuel modeled for this analysis. Therefore, burnup requirements generated here will conservatively bound all fuel to be stored in the Farley Units 1 & 2 SFPs.

The acceptability of the storage arrays developed in this analysis is ensured by controlling the assemblies that can be stored in each array. Assemblies are divided into Fuel Categories 1 through 4, and D (assemblies meeting the requirements of the damaged fuel array, Array D), based on assembly reactivity determined as a function of assembly average burnup, initial enrichment[1], IFBA loading[2], and decay time. An assembly's fuel category determines in which storage arrays it may be stored. Fuel Category 1 defines the most reactive assemblies, i.e. fresh 5 weight percent (wt%) ^{235}U assemblies without IFBA and Fuel Category 4 defines the least reactive assemblies, i.e., representing low reactivity assemblies that can be stored in Array B (see Table 5-3).

2.3 COMPUTER CODES

The analysis methodology employs the following computer codes and cross-section libraries: (1) the two dimensional (2-D) transport lattice code PARAGON Version 1.2.0, as documented in WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (Reference 1) and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3), and (2) Scale Version 6.2.3, as documented in ORNL/TM-2005/39, "Scale: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation" (Reference 2), with the ENDF/B-VII 238-group cross-section library.

2.3.1 Two-Dimensional Transport Code PARAGON

PARAGON is used in this application to simulate in-reactor fuel assembly depletion to generate isotopics for burnup credit. PARAGON is the Westinghouse Electric Company LLC state-of-the-art 2-D lattice transport code for pressurized water reactor (PWR) applications. It is part of the Westinghouse core design package and provides lattice cell data for three dimensional (3-D) core simulator codes.

This data includes macroscopic cross-sections, microscopic cross-sections for feedback adjustments, pin factors for pin power reconstruction calculations, and discontinuity factors for a 3-D nodal method solution of the diffusion equation. PARAGON uses the collision probability theory within the interface current method to solve the integral transport equation. Throughout the calculation, PARAGON uses the exact heterogeneous geometry of the assembly and the same energy groups as in the cross-section library to compute the multi-group fluxes for each micro-region location of the assembly. In order to generate the multi-group data, PARAGON goes through four steps of calculations: resonance self-shielding, flux solution, burnup calculation, and homogenization. The 70-group PARAGON cross-section library is based on the ENDF/B-VI.3 basic nuclear data. It includes explicit multigroup cross-sections and other nuclear data without any lumped fission products or pseudo cross-sections. PARAGON and its 70-group cross-section library are benchmarked, qualified, and licensed both as a standalone transport code and as a nuclear data source for a core simulator in a complete nuclear design code system for core design, safety, and operational calculations. The list of fuel isotopes modeled in PARAGON and subsequently modeled in the criticality analysis are given in Table 2-1.

---

1. Initial enrichment is the enrichment of the central zone region of fuel, excluding axial cutbacks\blankets and prior to reduction in ^{235}U content due to fuel depletion. If the fuel assembly contains axial regions of different ^{235}U enrichment values, such as axial cutbacks or low enriched blankets, the maximum initial enrichment value is to be used.
2 IFBA loading restrictions only apply to fresh fuel being stored as Fuel Category 2.

Table 2-1 Isotopes Used in the Nuclear Criticality Safety Analysis

a,c

Additional qualification of PARAGON for use in spent fuel pool applications has been performed at Westinghouse. The Electric Power Research Institute (EPRI) has developed PWR reactivity depletion benchmarks using a large set of measured flux data (flux maps) in EPRI Report 3002010613, "Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty" (Reference 3).

A guide for application of the EPRI depletion benchmarks for use in burnup credit calculations is given by way of example in Reference 4, with this methodology repeated by Westinghouse in "EPRI Depletion Benchmark Calculations Using PARAGON" (Reference 5). Results of this analysis provide additional confidence in the usage of PARAGON for SFP reactivity calculations, and provide a sound basis for usage of the 5% decrement approach for depletion uncertainty (See Section 5.2.3.1.5), showing that the depletion isotopics generated with PARAGON, and input into CSAS5 input models in Scale is conservative for determining depletion uncertainty.

PARAGON is generically approved for depletion calculations (Reference 1). PARAGON has been chosen for this spent fuel criticality analysis because it has all the attributes needed for burnup credit applications. There are no Safety Evaluation Report limitations for the use of PARAGON in UO_2 criticality analysis.

2.3.2 Scale Code Package

The Scale system was developed for the U.S. Nuclear Regulatory Commission (NRC) to standardize the method of analysis for evaluation of nuclear fuel facilities and shipping package designs (Reference 2). In this SFP criticality analysis, the Scale code package is used to calculate the reactivity of fissile systems in SFP conditions. Specifically, the Scale package is used to analyze infinite arrays for all storage arrays in the SFPs, finite rack modules and SFP representations to evaluate interfaces, soluble boron requirements, and postulated accident scenarios to demonstrate that the requirements in Section 2.1 are met.

The Scale package includes the control module Criticality Safety Analysis Sequence with KENO V.a (CSAS5), which provides reliable and efficient means of performing k_{eff} calculations for systems that are routinely encountered in engineering practice, especially in the calculation of k_{eff} of 3-D system models. Updated structurally from prior versions, CSAS5 implements the modern material and cross section processing module XSProc to process material input and provide a temperature resonance-corrected cross section library based on the physical characteristics of the problem being analyzed. XSProc calls several lower level functional modules, some of which perform simple functions that were not called out as separate from CSAS5 in past versions.

XSProc was developed for the Scale 6.2 release to prepare data for continuous-energy and multigroup calculations. XSProc expands material input from Standard Composition Library definitions into atom number densities (calling the integrated MixMacros module) and, for multigroup calculations, performs cross section resonance self-shielding, energy group collapse, and spatial homogenization. XSProc implements capabilities for problem-dependent temperature interpolation, calculation of Dancoff factors (calling the integrated Dancoff module), resonance self-shielding using Bondarenko factors with full-range intermediate resonance treatment, as well as use of continuous energy resonance self-shielding in the resolved resonance region. XSProc integrates and enhances the capabilities previously implemented independently in BONAMI, CENTRM, PMC, WORKER, ICE, and XSDRNPM, along with some additional capabilities that were provided by MIPLIB and SCALELIB in prior Scale release. For this work XSProc utilizes the following modules in addition to MixMacros and Dancoff (CENTRM and PMC are called via the CentrmPmc module):

- BONAMI: The BONAMI module is used to perform Bondarenko calculations for resonance self-shielding. BONAMI obtains problem-independent cross sections and Bondarenko shielding factors from a multigroup (MG) AMPX master library, and it creates a MG AMPX working library of self-shielded, problem-dependent cross sections. Several options may be used to compute the background cross section values using the narrow resonance or intermediate resonance approximations, with and without Bondarenko iterations. A novel interpolation scheme is used that avoids many of the problems exhibited by other interpolation methods for the Bondarenko factors. BONAMI is most commonly used in automated SCALE sequences and is fully integrated within the Scale cross section processing module, XSProc. During the execution of a typical Scale computational sequence using XSProc, Dancoff factors for uniform lattices of square- or triangular-pitched units are calculated automatically for BONAMI by numerical integration over the chord length distribution. Heterogeneous effects are treated using equivalence theory based on an “escape cross section” for arrays of slabs, cylinders, or spheres.

- CENTRM: CENTRM computes continuous-energy neutron spectra for infinite media, 1-D) systems, 2-D unit cells in a lattice, by solving the Boltzmann transport equation using a combination of pointwise and multigroup nuclear data. CENTRM is primarily used to calculate problem-specific fluxes on a fine energy mesh to generate self-shielded multigroup cross sections for subsequent radiation transport computations. Several calculation options are available, including a slowing-down computation for homogeneous infinite media, 1-D discrete ordinates in slab, spherical, or cylindrical geometries; a simplified two-region solution; and 2D method of characteristics for a unit cell within a square-pitch lattice.

- PMC: PMC generates problem-dependent multigroup cross-sections from an existing AMPX multigroup cross-section library, a point wise nuclear data library, and a pointwise neutron flux file produced by the CENTRM continuous-energy transport code. In the Scale sequences, PMC is used primarily to produce self-shielded multigroup cross-sections over a specified energy range such as the resolved resonance energy range of individual nuclides in the system of interest. The self-shielded cross-sections are obtained by integrating the point wise nuclear data using the CENTRM problem-specific, continuous-energy flux as a weight function for each spatial zone in the system.

- KENO: The KENO module is a Monte Carlo criticality program used to calculate the k_{eff} of 3-D models using continuous energy or multigroup cross-sections and is called by CSAS5 once XSProc is complete. Flexible geometry features and the availability of various boundary condition prescriptions in KENO allow for accurate and detailed modeling of fuel assemblies in storage racks, either as infinite arrays or in actual SFP models. The version used in this work, KENO V.a, contains a simplified geometry package appropriate for use here. Anisotropic scattering is treated by using discrete scattering angles using P_n Legendre polynomials. KENO uses problem-specific cross-section libraries, processed for resonance self-shielding and for the thermal characteristics of the problem.

For this work, the option parm=centrm is used as input, for which the CENTRM/PMC modules are executed to process shielded multi-group cross sections using continuous energy flux spectra calculated with the recommended type of continuous energy transport solver for the designated type of cell. An infinite homogeneous medium calculation is used for those materials not called out for special processing, uses 2-D Method of Characteristics for a LATTICECELL consisting of cylindrical fuel rods in a square lattice, and uses 1-D discrete S_n transport for all other LATTICECELLs and MULTIREGION cells.

The criticality sequence of Scale 6.2.3 is validated using fresh UO_2 critical experiments and Haut Taux de Combustion (HTC) critical experiments to form an experiment benchmark suite applicable to fresh and spent fuel criticality calculations. See NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (Reference 16) for an overview of the HTC criticals. Additional details of the validation are found in Reference 19. The validation shows that Scale 6.2.3 is an accurate tool for calculation of k_{eff} for SFP applications. The benchmark calculations use the same computer platform and cross-section libraries that are used for the design basis calculations. The validation considers both fresh UO_2 and fuel with plutonium designed to have an actinide composition similar to burned fuel.

2.3.3 Scale 238 Group Cross-Section Library

The 238-group ENDF/B-VII library included in the Scale package is available for general purpose criticality analyses. The group structure is the same as the 238-group ENDF/B-V and ENDF/B-VI libraries in Scale, and the same weighting spectrum as for the ENDF/B-VI. As with the 238-group ENDF/B-VI library, the ENDF/B-VII library cannot be used with the NITAWL-III module for resonance self-shielding calculations in the resolved range.

The 238-group and continuous-energy ENDF/B-VII libraries have 417 nuclides that include 19 thermal-scattering moderators. The validation of the ENDF/B-VII 238-group library with the Scale Version 6.2.3 CSAS5 module is documented in Reference 19.

3 FARLEY UNITS 1 & 2 NUCLEAR POWER PLANT

This section describes the physical characteristics of Farley Units 1 & 2 that are important to SFP criticality safety. Pertinent reactor characteristics and associated fuel design and fuel management history are discussed in Section 3.1. The physical characteristics of the SFPs are discussed in Section 3.2.

3.1 REACTOR DESCRIPTION

The Farley Units 1 & 2 Nuclear Power Plant is a Westinghouse PWR utilizing fuel with a 17 x 17 lattice. Farley Units 1 & 2 have used multiple fuel designs from Westinghouse. All fuel assemblies used at Farley Units 1 & 2 incorporate a 17 x 17 square array of 264 fuel rods with 24 guide tubes (GT) and 1 instrument tube (IT). The fuel rod cladding material is Zircaloy cladding and its variants, such as ZIRLO High Performance Fuel Cladding Material. Each fuel rod contains a column of enriched UO_2 fuel pellets. The pellets are pressed and sintered, and are dished on both ends.

Section 3.1 provides data on the design and operation of Farley Units 1 & 2 as well as the fuel designs and fuel management of the plant. Table 3-1 provides basic data on the type of reactor and the fuel types that comprise Farley Units 1 & 2. The neutronically important mechanical features of the three fuel designs are listed in Table 3-2.

Table 3-1 Reactor General Specifications

Reactor type	Westinghouse
Historic & current reactor power[1] (MWt)	2652-2821
Fuel lattice	17 x 17
Fuel design 1	Westinghouse Standard Fuel Assembly
Fuel design 2	Westinghouse Optimized Fuel Assembly
Fuel design 3[2]	Westinghouse Robust Fuel Assembly

Note:
1. Reactor power will be analyzed in this work up to 2831 MWt with current fuel management strategy to address future operation.
2. The RFA fuel design has not been used at Farley, and there are no current plans to transition to this fuel design. However, it is included in this analysis to support potential future use provided the RFA design is operated within the analysis area of applicability.

Table 3-2 Fuel Design Mechanical Specifications

Assembly type	STD	RFA	OFA
Rod array size	17 x 17	17 x 17	17 x 17
Rod pitch, in	0.496	0.496	0.496
Active fuel length, in	144	144	144
Total number of fuel rods	264	264	264
Fuel cladding outer dimension (OD), in	0.374	0.374	0.36
Fuel cladding inner dimension (ID), in	0.329	0.329	0.315
Fuel cladding thickness, in	0.0225	0.0225	0.0225
Pellet diameter, in	0.3225	0.3225	0.3088
Number of GT/IT	24/1	24/1	24/1
GT/IT OD, in	0.482	0.482	0.474
GT/IT ID, in	0.450	0.442	0.442
Percent theoretical density, nominal	95.0 – 96.5	95.0 – 96.5	95.0 – 96.5

Non-mechanical fuel features which are important to criticality safety and how they impact the number of distinct fuel designs are considered in this analysis. Operational characteristics of every cycle operated at Farley Units 1 & 2 were reviewed. All cycles can be categorized conservatively into one of the following Criticality Fuel Designs. Table 3-3 outlines the key non-mechanical features and fuel management history of each of the fuel designs.

Table 3-3 Non-Mechanical Specifications and Operating History

Criticality Fuel Design	1	2	3	4	5	6	7
Assembly type	STD	STD	STD	OFA	OFA	OFA	OFA
Max. TD[1]	96.5	96.5	98.0	96.5	96.5	96.5	98.0
Max. operating power, MWt	2652	2652	2831	2775	2652	2775	2831
Axial blanket enrichment	No Blanket	No Blanket	Annular, Fully Enriched	No Blanket	No Blanket	No Blanket	Annular, Fully Enriched
Axial blanket length, in	N/A	N/A	6	N/A	N/A	N/A	6
Burnable absorber (BA) Type	Pyrex	WABA	IFBA	IFBA / WABA	IFBA / WABA	IFBA	IFBA
BA material	B_2O_3-SiO_2	B_4C	ZrB_2	ZrB_2 / B_4C	ZrB_2/ B_4C	ZrB_2	ZrB_2
BA max. loading	12.5 wt%	6.03 mg^{10}B/cm	1.50X	1.00X / 6.03 mg^{10}B/cm	1.00X / 6.03 mg^{10}B/cm	1.50X	1.50X
Max BA length, in	144	134	132	IFBA: 132 WABA: 132	IFBA: 132 WABA: 134	132	132
Maximum number of rods / fingers	24	20	200	156 / 8	104 / 12	156	200

Note:

1. Percentage of Theoretical Density (TD).
2. The Max. TD for Criticality Fuel Designs 3[1] and 7 are chosen to bound potential future operation and are greater than current experienced at Plant Farley as seen in Table 3-2.

Criticality Fuel Design 1 and 7 are selected as the design basis fuel designs for STD/RFA and OFA fuel designs respectively. See Section 4.3.2 for methodology details for determination of the design basis criticality fuel designs.

[1] See Section 4.3.2 for additional details. Bounding design is burnup dependent in Burnup Bin 1.

3.2 FUEL STORAGE DESCRIPTION

The physical characteristics of the Farley Units 1 & 2 SFPs are described in this section. The SFPs are made up of one fuel storage rack design (region). The Farley Units 1 & 2 SFPs each consist of two 6 x 7, nineteen 7 x 7, and seven 7 x 8 storage racks. The storage racks are of flux trap style with an uncredited Boraflex neutron absorber panel on every side (in the x and y-axis directions) of each storage cell. This results in a flux trap between any two assembly storage locations. A schematic layout of the unit 1 and unit 2 SFP is given in Figure 3-1 and Figure 3-2, respectively. See Section 5.1.1 for modeling details. The specifications for the storage racks are given in Table 3-4.

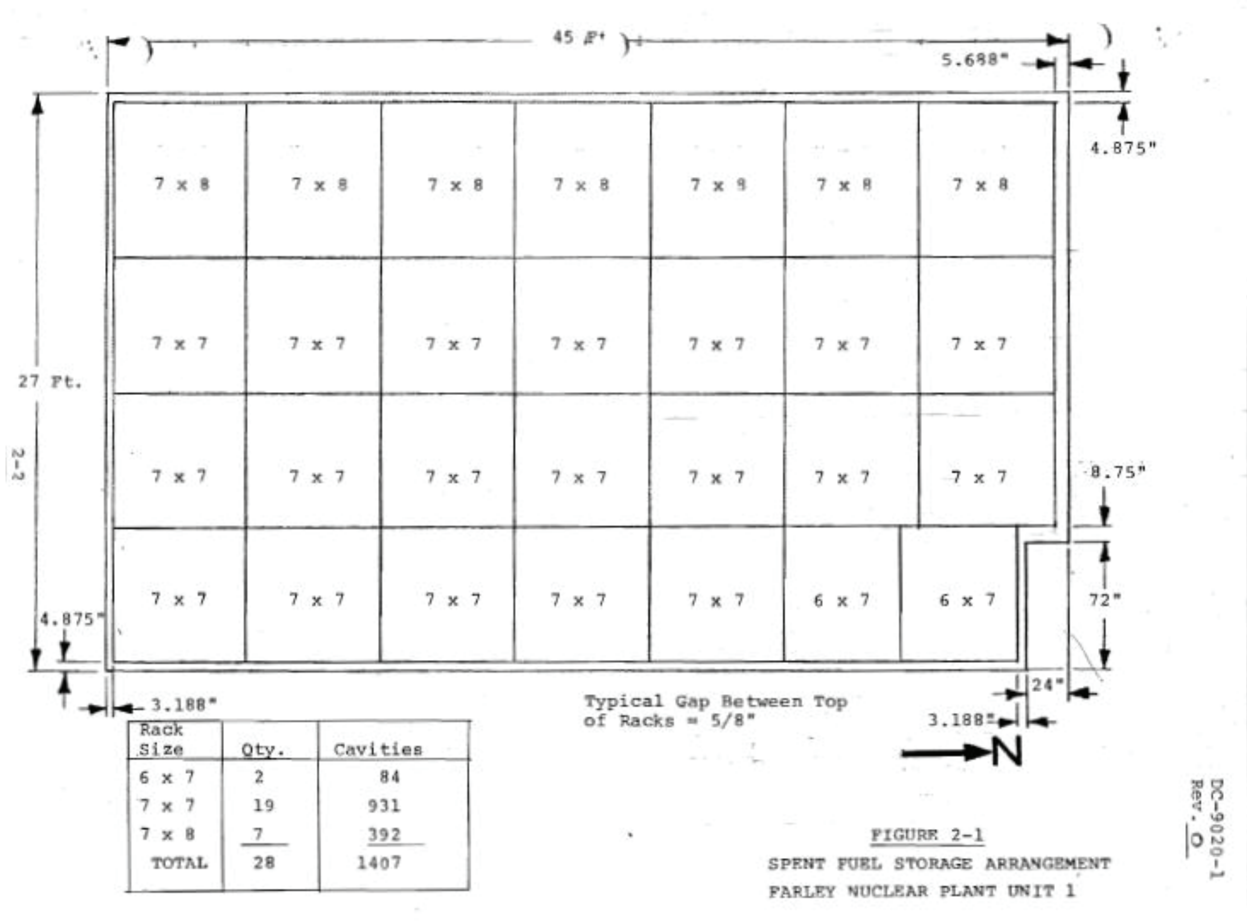

Figure 3-1 Farley SFP (Unit 1) Layout

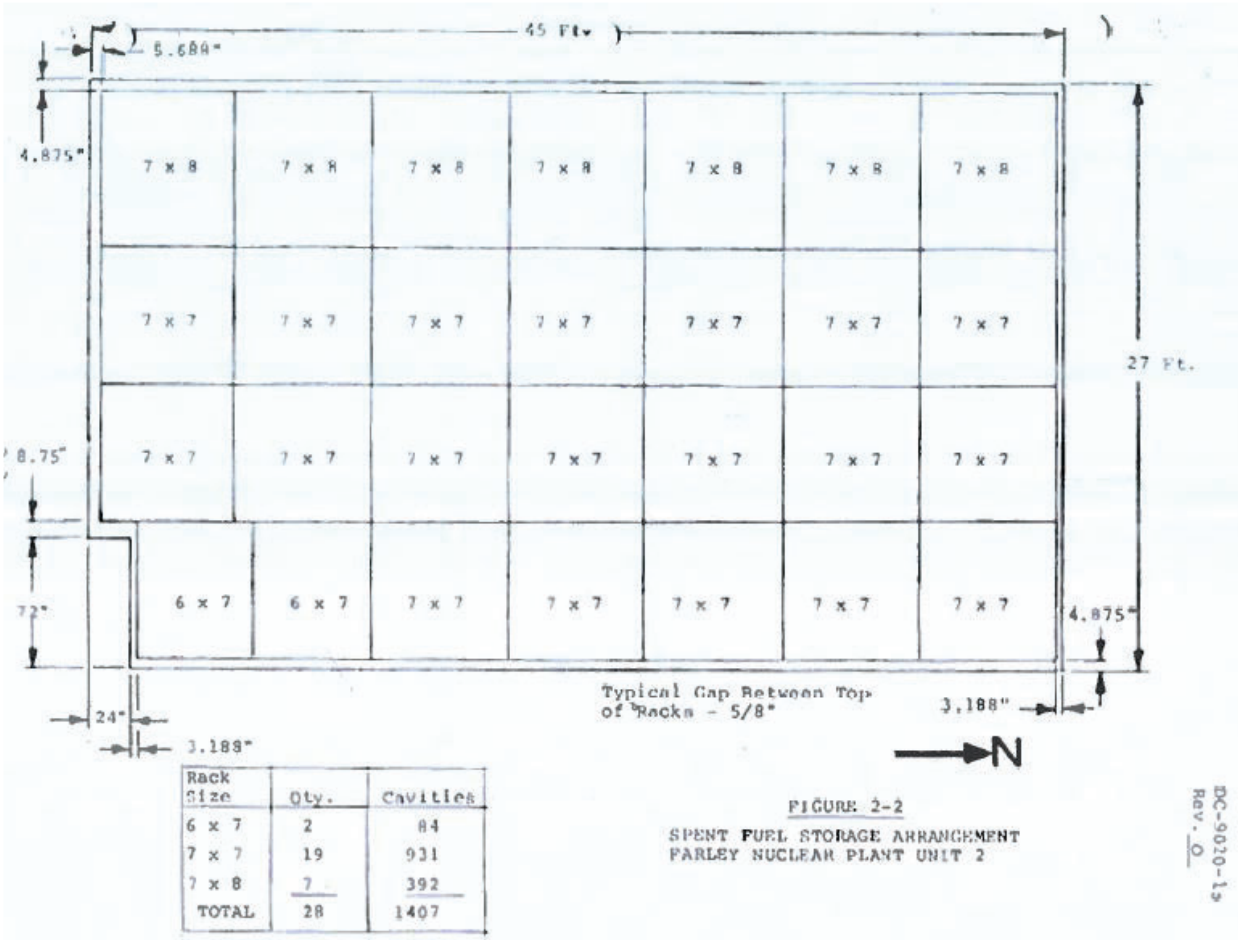

Figure 3-2 Farley SFP (Unit 2) Layout

Table 3-4 Fuel Storage Rack Specifications

	Value	Tolerance
Cell pitch, in	10.75	±0.06
Cell ID, in	8.9	±0.045
Cell wall thickness, in	0.12	± 0.012
BA[1] Type	Boraflex	N/A
BA cavity width, in	8	±0.06
BA cavity thickness, in	0.07	N/A
BA wrapper thickness, in	0.024	±0.003

Note:

1. No credit is taken for the presence of any residual Boraflex. The BA cavity is assumed to be filled with water of the same composition as the water elsewhere in the storage racks.

Also present in the SFPs are fuel rod storage canisters (FRSCs) and loose pellet transport canisters (LPTCs). The fuel rod storage canister at Farley Units 1 & 2 is a rectangular lattice of storage tubes for failed fuel rods arranged in an 8 x 8 pattern. Design details of the FRSCs are given in Table 3-5. Section 5.4.3 contains additional modeling details.

The loose pellet transport canisters (LPTCs) are stainless steel (SS) canisters designed to store up to 5000 loose fuel pellets within individual loose pellet canisters stored within the LPTCs. Design details are given in Table 3-6. Section 5.4.3 contains additional modeling details.

Table 3-5 Fuel Rod Storage Canister	
Maximum rod loading of FRSC	52 rods with fresh 5 wt% ^{235}U loading
Length of FRSC, in	155.75
Lattice of storage locations	8 x 8
Fuel rod storage tubes per row	4, 6, 8, 8, 8, 8, 6, 4
Fuel rod storage tube OD, in	0.750
Fuel rod storage tube thickness, in	0.120
Fuel rod storage tube material	SS-304 (not modeled)
Fuel rod pitch, in	0.937

Table 3-6 Loose Pellet Transport Canister	
Maximum Loading of LPTC	5000 pellets at 5 wt%, no burnup
Length of LPTC, in	258
Outer shell material	SS-304
Outer shell thickness, in	0.375
Loose pellet canister outer dimensions, in	7 x 5
Loose pellet canister material	SS-304
Loose pellet canister SS thickness, in	0.078125

4 DEPLETION ANALYSIS

This section describes the methods used to determine the conservative and bounding inputs for the generation of isotopic number densities, which are then used in subsequent Monte Carlo simulations.
[

]a,c

4.1 DEPLETION MODELING SIMPLIFICATIONS & ASSUMPTIONS

There are several different combinations of fuel designs including differing mechanical designs, operating conditions, and BA types that need to be considered when performing the analysis. To facilitate the analysis, two bounding design basis fuel assembly types are determined, one for fuel designs with a nominal rod outer diameter of 0.374 inches (STD) and one with a nominal rod outer diameter of 0.360 inches (OFA).

[

]a,c

- Depletion isotopics for use in the Criticality Analysis are generated every 2000 MWd/MTU.

4.2 FUEL DEPLETION PARAMETER SELECTION

4.2.1 Fuel Isotopic Generation

This section outlines how parameters are selected for use in the fuel depletion calculations to generate isotopic number densities. For the purposes of this analysis, the isotopic number densities generated are differentiated by fuel design, fuel enrichment, burnup, and decay time after discharge.

[

]a,c

Based on the Farley Units 1 & 2 fuel management, the fuel has isotopic number densities which are calculated at enrichments of 3, 4, and 5 wt% ^{235}U and decay times of 0, 5, 10, 15, and 20 years. Fresh fuel modeled in this analysis conservatively excludes ^{234}U and ^{236}U.

4.2.2 Reactor Operation Parameters

The reactivity of the depleted fuel in the SFP is determined by the in-reactor depletion conditions. The conditions experienced in the reactor impact the isotopic composition of fuel being discharged to the SFP. NUREG/CR-6665, “Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel” (Reference 6) provides discussion on the core operation parameters important to SFP criticality. NEI-12-16, Revision 3, “Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants” (Reference 14) provides practical guidance for criticality safety analysts in line with current recommendations. This section outlines the parameters used in generating the fuel isotopics and why they are appropriate for use in this analysis. The operating conditions of the fuel selected for modeling are provided in Table 4-5, which provides both the nominal values and the values assumed in the analysis.

4.2.2.1 Soluble Boron Concentration

The soluble boron concentration in the reactor during operation impacts the reactivity of fuel being discharged to the SFP. Because boron is a strong thermal neutron absorber, its presence hardens the neutron energy spectrum in the core, creating more plutonium.

Based on guidance from Reference 6, “establishment of a bounding value for the maximum average boron per cycle based on boron let-down curves would enable more straightforward application of the depletion analyses,” a constant cycle average soluble boron concentration (Equation 4-1) which assumes 19.9 at% ^{10}B in place of a soluble boron letdown curve is considered appropriately conservative. To determine the maximum cycle average soluble boron concentration, fuel management strategies for

Farley Units 1 & 2 have been reviewed. Table 4-1 provides the cycle average soluble boron concentration for cycles 1 through 29 of Unit 1 and for cycles 1 through 26 of Unit 2.

[ ]a,c Equation 4-1

[

]a,c

Table 4-1 Cycle Average Soluble Boron Concentration (ppm)

Cycle #	Unit 1	Unit 2
Cycle 1	567.3	547.5
Cycle 2	491.6	482.6
Cycle 3	507.8	653.3
Cycle 4	493.3	703.7
Cycle 5	461.3	685.8
Cycle 6	669.6	806.7
Cycle 7	783.0	769.3
Cycle 8	720.1	724.3
Cycle 9	777.1	851.7
Cycle 10	777.2	694.1
Cycle 11	789.9	856.0
Cycle 12	762.5	807.9
Cycle 13	754.4	722.3
Cycle 14	828.0	684.8
Cycle 15	771.6	783.7
Cycle 16	714.8	767.1
Cycle 17	755.9	800.7
Cycle 18	785.3	771.1
Cycle 19	797.0	803.9
Cycle 20	785.3	751.1
Cycle 21	783.9	801.1
Cycle 22	765.4	805.4
Cycle 23	783.2	777.5
Cycle 24	746.0	768.2
Cycle 25	786.0	798.8
Cycle 26	777.9	788.9
Cycle 27	771.3	N/A
Cycle 28	787.1	N/A
Cycle 29	780.6	N/A

4.2.2.2 Fuel Temperature

The fuel temperature during operation impacts the reactivity of fuel being discharged to the SFP. Increasing fuel temperature increases resonance absorption in ^{238}U due to Doppler broadening which leads to increased plutonium production, increasing the reactivity of the discharged fuel. Therefore, utilizing a higher fuel temperature is more conservative.

The temperature input for this analysis is calculated by the FIGHTH code documented in Westinghouse WCAP-9522, "FIGHTH – A Simplified Calculation of Effective Temperatures in PWR Fuel Rods for Use in Nuclear Design" (Reference 7), which determines the fuel temperatures used as input to PARAGON for depletion calculations. FIGHTH calculates the steady state radial temperature distribution at each burnup, given the local value of the heat generation rate in the rod, the moderator temperature, and coolant flow rate. The FIGHTH model accounts for radial variations of the heat generation rate, thermal conductivity, thermal expansion in the fuel pellet, elastic deflection for the cladding, and pellet-clad gap conductance. The FIGHTH code is used in the development of cross-sections for in-core calculations as part of the standard reload methodology.

As discussed, the important input parameters used by FIGHTH for determining fuel temperature are power level, moderator temperature, and coolant flow rate. [

]^{a,c} Selection of moderator temperature is performed as discussed in Section 4.2.3.2. [

]^{a,c}

4.2.2.3 Operating History and Specific Power

The analysis assumes constant full power operation consistent with a bounding assembly average power. For fission product credit analyses, the conservative direction for specific power varies with burnup (see Reference 6). However, assuming a bounding assembly average power (therefore high specific power) ensures high fuel temperatures which is conservative throughout life. Interim Staff Guidance (ISG) DSS-ISG-2010-001 (Reference 8) states:

> "It may be physically impossible for the fuel assembly to simultaneously experience two bounding values (i.e., the moderator temperature associated with the "hot channel" fuel assembly and the minimum specific power). In those cases, the application should maximize the dominate parameter and use the nominal value for the subordinate parameter."

As anticipated by the ISG and consistent with sensitivity study results reported in Reference 6, the fuel temperature impact on reactivity is greater than the impact from specific power. Guidance in Reference 14 corroborates this assessment. This makes the selection of a high operating power, and therefore specific power to maximize fuel temperature, appropriate as the subordinate parameter is more conservative than nominally chosen and the 0.002 Δk bias recommended for a bounding treatment in Reference 6 is unnecessary. For additional conservatism, a 0.002 Δk uncertainty is taken on operational history.

4.2.2.4 Maximum Average Assembly Power

[

]^{a,c}

Table 4-2 Fuel Design Operating Power

a,c

Parameter	Value						

[

]^{a,c}

4.2.3 Axial Profile Selection

This section discusses the selection of bounding axial burnup and moderator temperature profiles. [

]a,c

4.2.3.1 Axial Burnup Profile Selection

This section describes the methods used to determine the limiting distributed axial burnup profiles. These profiles will be used along with the uniform axial burnup profile as one of the conservative input parameters to develop isotopics to ultimately calculate the minimum burnup requirements provided in Section 6.1.

As discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses" (Reference 9), as fuel is operated in the reactor, the axial center of each assembly generates more power than the ends. This leads to the burnup of each assembly varying along its length. Because the axial center of each assembly generates most of the power, the burnup in the axial center of the assembly is greater than the assembly average. At the same time, the ends of the assembly are less burned than the assembly average. When the burnup difference between the axial center and end of an assembly is large enough, reactivity becomes driven by the end of the assembly rather than the axial center, as the under depletion of the ends (the end-effect) overcomes the reactivity loss due to neutron leakage.

As driven by the end-effect, the following methodology was used to ensure that the appropriate axial burnup profiles were selected for this analysis. Fuel management calculations containing readily available data from 25 cycles of operation were utilized to develop a database of axial burnup profiles specific to Farley Units 1 & 2. [

]a,c

[

]a,c

For the reasons discussed above, it is typical for fuel modeled assuming a uniform axial burnup profile to be more reactive early in life than fuel modeled with a distributed profile. To address this, isotopics were created for fuel assuming both a uniform profile and distributed profile (for the design basis fuel). These isotopics were used during the Monte Carlo calculations to determine the minimum burnup requirements to ensure the limiting profile (uniform vs design basis distributed) has been used.

4.2.3.2 Axial Moderator Temperature Profile Selection

This section describes the methods used to determine the limiting axial moderator temperature profiles. These profiles will be used together with axial distributed and uniform burnup profiles to calculate the isotopics used in generating the burnup requirements provided in Section 6.1.

Selecting an appropriate moderator temperature profile is important as it impacts the moderator density and therefore the neutron spectrum during depletion as discussed in Reference 6. An appropriate moderator temperature ensures the impact of moderator density on the neutron spectral effects is bounded, conservatively biasing the isotopic inventory of the fuel.

[

]a,c

4.2.4 Burnable Absorber Usage

Burnable absorber usage at Farley Units 1 & 2 has been considered for this analysis and conservative assumptions have been used to bound the effects of BAs on fuel isotopics. The BAs that have been evaluated include both discrete and integral BAs. The BA rod parameters are shown in Table 4-3.

Table 4-3 Burnable Absorber Specifications

Parameter[1]	**Pyrex**	**WABA**	**IFBA**
BA material	B_2O_3-SiO_2	Al_2O_3-B_4C	ZrB_2
BA type	Discrete	Discrete	Integral
B_4C TD, %	N/A	70	N/A
Boric Oxide Content, wt%	12.5	N/A	N/A
^{10}B abundance or loading	19.9 at%	19.9 at%	2.35 (STD/RFA) / 2.25 (OFA) mg/in
BA thickness, in	0.073	0.02	0.0002[2]
BA ID, in	0.1900	0.2780	N/A
BA OD, in	0.3360	0.3180	N/A
BA clad material	Stainless Steel	Zirc-4	N/A
BA inner clad OD, in	0.1810	0.2670	N/A
BA inner clad thickness, in	0.0070	0.0210	N/A
BA outer clad OD, in	0.3810	0.3810	N/A
BA outer clad thickness, in	0.0185	0.0260	N/A
BA length, in	See Table 3-3	See Table 3-3	N/A
Max. BA exposure, MWd/MTU[3]	40000	40000	N/A

Notes:

1.) Additional BA information is contained in Table 3-3 for each Criticality Fuel Design considered. The maximum BA length and loading are modeled for each Criticality Fuel Design.

2.) Coating on the fuel pellet. IFBA depletion input captures the desired absorber per unit length with 0.2 mils coating thickness. Specific criticality analysis input is given in Table 5-1.

3.) Pyrex and WABA exposure is conservatively modeled to 40000 MWd/MTU. IFBA residual ^{10}B is removed from the fuel for spent fuel pool criticality calculations.

Criticality Fuel Designs 2, 4, and 5 contain WABA. The WABA length and number of WABA rodlets present will impact the final assembly reactivity, as they are directly related to the amount and location of absorber present within the assembly. For past operation with these designs, a minimum of a 5” cutback is observed, however less conservative 6” cutbacks are utilized in the depletion calculations. The maximum number of WABA rodlets used for past operating cycles falling within these designs is 20 yet 24 rodlets were analyzed. The additional rodlets ensures any reactivity impact lost from the additional inch of

WABA during operation is accounted for. For conservatism, discrete BAs (Pyrex and WABA) were not removed from the core until 40 GWd/MTU of operation.

4.2.5 Fuel Assembly Physical Changes with Depletion

Reference 14 discusses fuel assembly physical changes with depletion, and specifically calls out the need to address the potential reactivity impact from fuel rod changes (clad creep, fuel densification/swelling) and material dependent grid growth. Appendix B of Reference 14 is based on Westinghouse methodology and indicates that holistically the impact of fuel rod changes with depletion are conservative. Fuel assembly grid growth impact has been shown to be negligible during depletion and is conservatively addressed for the in-pool storage reactivity impact in Section. 5.2.3.1.11.

4.3 DESIGN BASIS FUEL SELECTION

4.3.1 Fuel Design and Management Modeling Considerations

To develop conservative storage requirements for the Farley Units 1 & 2 SFPs, the different unique fuel designs (criticality fuel designs) and the conditions in which those designs were operated or are planned to be operated were considered. All the criticality fuel designs considered are discussed in Section 3.1. [

]a,c

Table 4-4 Design Basis Comparison Modeled Fuel Design Parameters

	Fuel Design	
Parameter	**STD/RFA**	**OFA**
Rod pitch, in	0.496	0.496
Active fuel length, in	144	144
Total number of fuel rods	264	264
Pellet OD, in	0.3225	0.3088
Clad OD, in	0.374	0.360
Clad ID, in	0.329	0.315
Number of GT/IT	24/1	24/1
GT/IT OD, in	0.482	0.474
GT/IT ID, in	0.442	0.442

Notes:
1. Blanket/Cutback, BA type and loading information is given in Table 3-2
2. RFA fuel model GT ID is used. Section 6.2 discusses the applicability of the RFA fuel design.

[

]^{a,c}

At Farley Units 1 & 2, the STD/RFA and OFA fuel designs have used the Pyrex, WABA and IFBA BAs. As discussed in NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for the PWR Burnup Credit" (Reference 11), the presence of discrete burnable absorbers such as Pyrex and WABA displace water and absorb thermal neutrons, thereby hardening the neutron spectrum and creating more plutonium isotopes. Therefore, Pyrex and WABA must be modeled in the depletion analysis. As discussed in NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit" (Reference 10), the presence of integral absorbers during depletion hardens the neutron spectrum, resulting in lower ^{235}U depletion and higher production of plutonium isotopes. As a result, the IFBA integral absorber must also be modeled in the depletion analysis for determination of the bounding criticality fuel design.

4.3.2 Reactivity Comparison Methodology

The final bounding assembly design is the determination of a limiting combination of fuel type and conservative depletion input parameters, denoted as a criticality fuel design. The design basis conservatively covers past, current, and expected future spent fuel operation for Farley Units 1 & 2. This section outlines the methodology used to determine a bounding assembly design, including the selected criticality fuel type. For this analysis, a bounding fuel type is determined for the STD/RFA and OFA fuel designs. The potentially limiting axial burnup profiles identified using the methodology approach described in Section 4.2.3.1 are implemented together with the limiting depletion parameters for each Criticality Fuel Design.

Reactivity comparisons were performed across all burnup bins at 3, 4, and 5 wt% ^{235}U with all potentially limiting axial burnup profiles and the limiting moderator temperature profiles for each criticality fuel design with STD/RFA and OFA fuel assemblies. Comparisons were performed for both Array B and Array C (See Section 5.2.1 for a description of storage arrays). Based on the reactivity comparison Criticality Fuel Design 1 was chosen as the limiting STD/RFA criticality fuel design throughout Burnup Bins 2, 3 and for the majority of Burnup Bin 1, while Criticality Fuel Design 7 was chosen as the limiting OFA criticality fuel design. Burnup Bin 1 results indicate a burnup dependent bounding design within the burnup bin range, such that Criticality Fuel Design 3 is limiting at lower burnup values within the burnup bin, with a shift to Criticality Fuel Design 1 at higher burnups within Burnup Bin 1. Note that when necessary, reactivity comparisons focused on a range of reactivity of interest to spent fuel pool criticality to ensure that the limiting design was appropriate.

In general, Criticality Fuel Design 1 was selected as limiting over Criticality Fuel Design 2 and 3 for STD (RFA) fuel. It is the combination of Criticality Fuel Design 1 input (including the burnable absorber usage during operation) that leads to the bounding nature, despite the lower fuel percent of theoretical density (96.5 vs 98.0) when compared with Criticality Fuel Design 3. See Section 6.2 for the analysis area of applicability.

4.4 FINAL DEPLETION PARAMETERS

This section outlines the parameters used in the final depletion calculations. The depletion parameters discussed in this section are:

- Core Operation Parameters
- Fuel Assembly Dimensions
- Axial Burnup and Moderator Temperature Profiles

The fuel isotopics used in the reactivity calculations were generated based on the data presented in Table 4-4, Table 4-5, and Table 4-6.

Table 4-5 Parameters Used in Depletion Analysis

Parameter	Nominal Values	Depletion Analysis
Maximum cycle average soluble boron concentration, ppm	461.3 – 856.0	[ ]a,c
Rated thermal power[1], MWt	2652 - 2775	[ ]a,c
Average assembly power, MWt	16.90 – 18.04 [2]	[ ]a,c
Soluble boron ^{10}B atom percent, %	19.9	[ ]a,c
Minimum core loading, kg U	72443 (STD), 66417 (OFA)	[ ]a,c
System pressure, psia	2250	[ ]a,c
Core outlet moderator temperature, °F	618.9 (max)	[ ]a,c
Core inlet moderator temperature, °F	541.1 (max)	[ ]a,c
Minimum RCS flow rate (thermal design flow), gpm/coolant pump	86000	[ ]a,c
Fuel designs	STD/RFA, OFA	[ ]a,c
Fuel assembly cutback/blanket region	See Table 3-3	[ ]a,c
Blanket type	See Table 3-3	[ ]a,c
TD	94.0 – 96.5	[ ]a,c
BA	Pyrex, WABA, IFBA	[ ]a,c
Max BA lengths, in	144 (STD), 134 (STD/OFA), 132 (STD/OFA)	[ ]a,c

Notes:
1. The current rated thermal power is 2775 MWt, with a proposed uprate power of 2831 MWt.
2. This number is calculated by dividing the rated thermal power by the number of fuel assemblies.
3. [ ]a,c
4. See Section 6.2 for overall applicability of fuel designs covered by the analysis area of applicability.

Table 4-6 Limiting Axial Burnup and Moderator Temperature Profiles

a,c

5 CRITICALITY ANALYSIS

This section describes the reactivity calculations and evaluations performed in developing the burnup requirements for fuel storage in the Farley Units 1 & 2 SFPs. The section also confirms continued safe SFP operation during both normal and accident conditions.

5.1 KENO MODELING APPROACH, SIMPLIFICATIONS & ASSUMPTIONS

As discussed in Section 2.3.2, KENO is the criticality code used to support this analysis. KENO is used to determine the absolute reactivity of burned and fresh fuel assemblies loaded in storage arrays. Additionally, KENO is used to determine the reactivity sensitivity of these storage arrays to effects such as manufacturing tolerances, fuel depletion, eccentric positioning, and the allowable temperature range of the SFPs. KENO is also used to model accident scenarios and confirm there is sufficient soluble boron to meet the requirements of Section 2.1.

The methods used to model the fuel in normal and accident scenarios are discussed in the following sections. [

□

□

□

□

□

□

□

□

]a,c

Table 5-1 IFBA Criticality Modeling Specifications

Parameter	**1.00X**	**1.25X**	**1.50X**

a,c

- [

]a,c

- Acceptable storage arrays are described in Section 5.2.1. Figure 5-1 shows a planar view (x-y) of each storage array as modeled (periodic boundary conditions applied in the x-y directions).

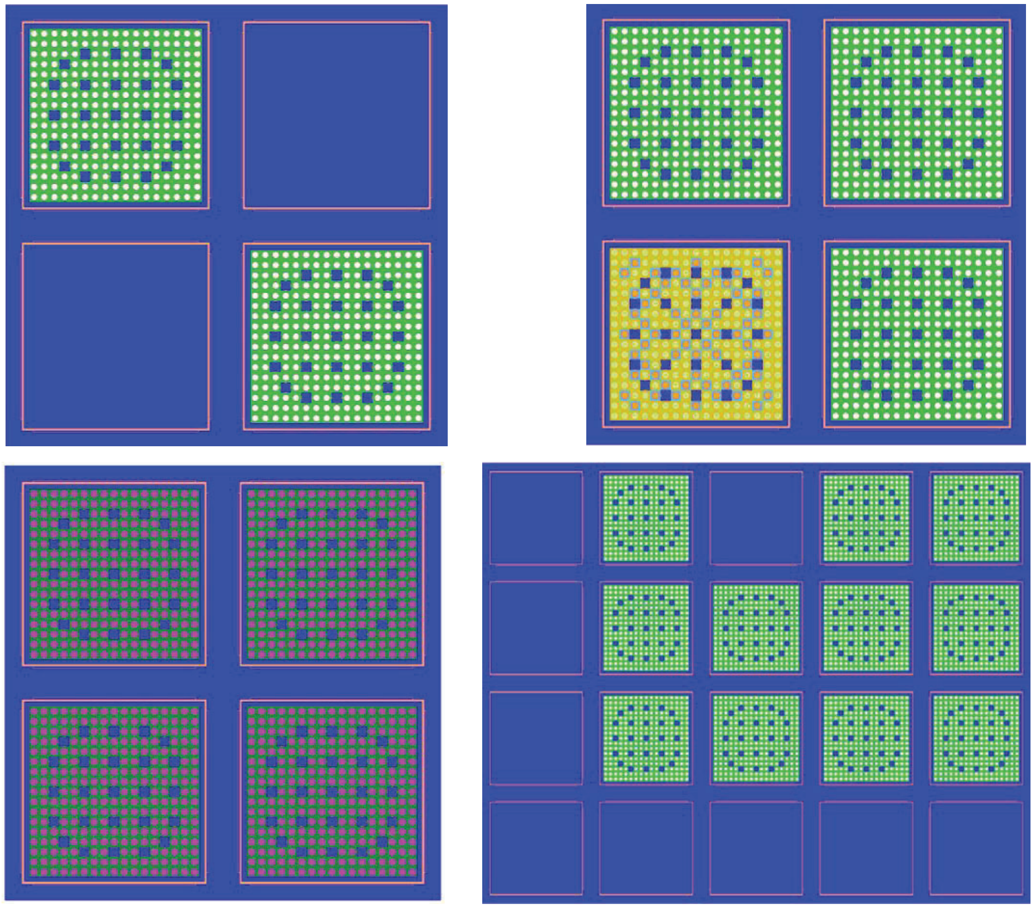

Figure 5-1 KENO Array Rack Model Planar (x-y) View: Top Left Array A, Top Right Array B, Bottom Left Array C, Bottom Right Array D (All Models: x-y Periodic Boundary Conditions)

5.1.1 Description of Fuel Assembly and Storage Racks for KENO

This section outlines the dimensions and tolerances of the design basis fuel assembly designs and the fuel storage racks. These dimensions and tolerances are the input basis for the KENO models used to determine the burnup requirements for each fuel storage array and to confirm the safe operation of the SFPs under normal and accident conditions.

5.1.1.1 Fuel Assembly Dimensions and Tolerances

This section provides the dimensions and tolerances for the design basis fuel assembly designs. Table 5-2 provides this data for STD/RFA and OFA fuel as modeled. Selection of these fuel designs is discussed in Section 4.3. As identified in Section 4.3.2, Criticality Fuel Design 1 was selected as limiting over Criticality Fuel Design 2 and 3 for STD (RFA) fuel. It is the combination of Criticality Fuel Design 1 input (including the burnable absorber usage during operation) that leads to the overall bounding nature, despite the lower fuel percent of theoretical density (96.5 vs 98.0) when compared with Criticality Fuel Design 3. See Section 6.2 for the analysis area of applicability.

Table 5-2 Design Basis Fuel Assembly Design Modeling Parameters		
STD (RFA) Fuel Assembly		
Parameter	**Value**	**Tolerance**
Rod array size	17 x 17	N/A
Rod pitch, in	0.496	[ ]^{a,c}
Active fuel length, in	144	N/A
Nominal fuel theoretical density, % TD	96.5	N/A
Maximum pellet enrichment, wt% ^{235}U	5	[ ]^{a,c}
Total number of fuel rods	264	N/A
Fuel cladding OD, in	0.374	[ ]^{a,c}
Fuel cladding ID, in	0.329	[ ]^{a,c}
Pellet diameter, in	0.3225	[ ]^{a,c}
Number of GT/IT	24/1	N/A
GT/IT OD, in	0.482	[ ]^{a,c}
GT/IT ID, in	0.442	[ ]^{a,c}
OFA Fuel Assembly		
Parameter	**Value**	**Tolerance**
Rod array size	17 x 17	N/A
Rod pitch, in	0.496	[ ]^{a,c}
Active fuel length, in	144	N/A
Nominal fuel theoretical density, % TD	98.0	N/A
Maximum pellet enrichment, wt% ^{235}U	5	[ ]^{a,c}
Total number of fuel rods	264	N/A
Fuel cladding OD, in	0.360	[ ]^{a,c}
Fuel cladding ID, in	0.315	[ ]^{a,c}
Pellet diameter, in	0.3088	[ ]^{a,c}
Number of GT/IT	24/1	N/A
GT/IT OD, in	0.474	[ ]^{a,c}
GT/IT ID, in	0.442	[ ]^{a,c}

Notes:
1. [ ]a,c
2. The maximum pellet enrichment tolerance is used for all enrichments evaluated as identified in Section 5.2.3.1.2.

5.1.1.2 Fuel Storage Cell Rack Dimensions and Tolerances

The storage racks used at Farley Units 1 & 2 SFPs are described in Section 3.2. The fuel storage cell characteristics, as they are modeled in the criticality analysis, are shown in Section 3.2. Dimensions including tolerances are given in Table 3-4. Tolerance models were created and the reactivity impacts were accounted for in the form of uncertainties added to the final reactivity calculation as shown in Table 5-4 through Table 5-10.

5.1.2 Impact of Structural Materials on Reactivity

Over the years, different fuel types have been developed to meet the needs of the utilities. Differences between the fuel types include changes in rod pitch, fuel rod dimensions such as pellet and cladding dimensions, and structural components such as grid material and volumes.

Each of the fuel types which have been or are planned to be operated at the plant need to be considered. For Farley Units 1 & 2, the determination of the design basis fuel assembly designs for the analysis has been performed as outlined in Section 4.3. The structural materials of each fuel type do not need to be considered in the determination of the bounding fuel assembly design as discussed in Reference 14 in regards to grid material where 50 ppm is added to soluble boron requirements as recommended to neglect modeling grids[1].

5.1.2.1 Composition of Structural Materials

Various zirconium-based materials and SS have traditionally been used as structural materials for fuel assembly designs. [

]a,c

5.1.2.2 Top and Bottom Nozzles

[

]a,c

[1] Reference 14 indicates this 50 ppm is also sufficient to offset the change in reactivity effect of tolerances under borated conditions (if modeling only unborated conditions for bias and uncertainty calculations).

5.1.2.3 Grids and Sleeves

A generic study determined the impact of grids and sleeves being present in an assembly both during core operation and storage in the SFP shows that the impact of grids and sleeves is negligible. The study was based on the [ ]^{a,c} The study incorporated a variety of depletion parameters and several different [ ]^{a,c} Additionally, Reference 14 indicates for depletion that 50 ppm additional soluble boron in determination of normal and accident condition soluble boron requirements is acceptable if grids are not explicitly modeled in the SFP criticality analysis. An additional 50 ppm was used.

5.2 KENO MODELING ANALYSIS

KENO models generated for STD/RFA and OFA bounding criticality fuel designs were evaluated for different storage arrays. The enrichment, assembly average burnup, and decay time input were varied to determine appropriate storage limits based on resulting reactivity. Reactivity margin is added to the KENO reactivity calculations for the generation of burnup requirements as discussed in Section 5.2.2 to account for manufacturing deviations.

5.2.1 Array Descriptions and Fuel Categories

Assembly storage is controlled through the storage arrays defined in this section. For storage arrays A, B and C, separate requirements are determined by fuel type (STD/RFA as one type, OFA as the second type). Each storage array contains assemblies which are defined by a fuel category as given in Table 5-3. A fuel category is a ranking of assemblies by the maximum allowable reactivity of the individual assembly in a storage cell within each storage location. A lower fuel category is more reactive than a higher fuel category. Unique storage locations were determined for which to assign fuel categories using the desired storage patterns for use in the Farley Units 1 and 2 SFPs. Figure 5-2 shows the allowable storage arrays, including the fuel categories discussed in this section.

Reactivity as discussed in this paragraph pertains to the maximum allowable reactivity in a storage cell. Fuel Category 1 locations can contain the highest reactivity fuel assemblies, up to 5 wt% ^{235}U assemblies with no burnup, IFBA, or decay time credit required. Fuel Category 2 locations can contain fresh fuel assemblies with up to 5 wt% ^{235}U but are subject to IFBA requirements (thereby reducing the reactivity compared to Fuel Category 1 fuel assemblies). Additionally, Fuel Category 2 assemblies must have accumulated at least 10 GWd/MTU of burnup once they have been exposed to ensure peak reactivity is considered for IFBA burnout, with the exception of a Fuel Category 3 or 4 assembly (discussed in this section). Fuel Category 3 locations are storage cells defined within Storage Array C, an "all-cell" storage cell. Fuel Category 3 locations can contain fuel assemblies of up to 5 wt% ^{235}U and have minimum burnup and/or decay time requirements which determine acceptability for storage in Array C.

To determine that Fuel Category 3 assemblies are less reactive than Fuel Category 2 assemblies (which require IFBA credit), a comparison of the burnup requirements of Fuel Category 4 assemblies that are stored with the Fuel Category 2 assemblies (in Array B) is necessary. If the three Fuel Category 4 assemblies are individually less reactive than Fuel Category 3 assemblies, then Fuel Category 2 assemblies will be allowed a higher reactivity than Fuel Category 3 assemblies because the overall storage array reactivities are the same (iso-reactive). Additionally, the 10 GWd/MTU required for (5 wt% ^{235}U fuel assemblies) Fuel Category 2 which have burnup is significantly less than the burnup requirements for 5 wt% ^{235}U Fuel Category 3 assemblies.

As can be seen by comparing the storage requirements of Fuel Category 4 assemblies and the storage requirements of Fuel Category 3 assemblies, the Fuel Category 3 assemblies require less burnup for the same enrichment, indicating the individual assemblies are allowed to be more reactive.

An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category (e.g., Fuel Category 3 assemblies can be stored in locations for Fuel Categories 1, 2, or 3, but cannot be stored in Fuel Category 4 locations). If lower reactivity fuel category requirements are met for an assembly, they need not meet the requirements of the fuel category cell for which they are stored). This is a unique occurrence because Fuel Category 2 requirements for exposed fuel were generated for the bounding case (10 GWd/MTU for 5 wt% ^{235}U assemblies) and applied for all burned fuel stored in Fuel Category 2 locations.

In addition to these defined fuel categories, Array D contains 11 fuel storage locations generically evaluated for storage within the Farley SFPs with STD fuel. This allows assemblies which meet the damaged fuel array storage requirements to be stored anywhere in the SFPs for which an array of this size is met. Array D assemblies were given the Fuel Category D label, indicating these storage cells are for damaged fuel. An example of Array D, which is currently employed at the Farley Unit 1 SFP, is shown in Figure 5-3 including the assembly ID used for storage. All 11 damaged assemblies meet the requirements of Array D.

Table 5-3 Fuel Categories Ranked by Reactivity

Fuel Category 1	High Reactivity
Fuel Category 2	
Fuel Category 3	
Fuel Category 4	Low Reactivity

Notes:
1. Assembly storage is controlled through the storage arrays defined in Figure 5-2.
2. Fuel Categories are ranked in order of decreasing reactivity, e.g., Fuel Category 2 is less reactive than Fuel Category 1, etc.
3. Each storage cell in an array can only be populated with assemblies of the fuel category defined in the array definition or a lower reactivity fuel category.
4. Fuel Category 1 contains fuel with an initial maximum enrichment up to 5 wt% ^{235}U. Neither burnup nor IFBA is required.
5. Fuel Category 2 contains fuel with an initial maximum enrichment up to 5 wt% ^{235}U. Storage of fresh fuel is determined from the minimum IFBA equation and coefficients provided in Table 6-1 for STD/RFA fuel and Table 6-3 for OFA fuel. Fuel Category 2 fuel which has been operated in the reactor requires at least 10.0 GWd/MTU of burnup with the exception of a Fuel Category 3 or 4 assembly.
6. Fuel Categories 3 and 4 are determined from the minimum burnup equation and coefficients provided in Table 6-7 and Table 6-11 for STD/RFA fuel, and in Table 6-9 and Table 6-13 for OFA fuel, respectively. Example burnup requirements at several initial enrichments and decay times are provided for Fuel Categories 3 and 4 in Table 6-8 and Table 6-12 for STD/RFA fuel, and Table 6-10 and Table 6-14 for OFA fuel, respectively.
7. Example IFBA requirements at several initial enrichments for IFBA thicknesses of 1.0X, 1.25X, and 1.50X are provided in Table 6-2 for STD/RFA fuel and in Table 6-4 for OFA fuel.

<table>
<tr><td rowspan="2" colspan="5">Array A

Two Category 1 assemblies with two empty storage locations. The Category 1 fuel assemblies must only be face adjacent to an empty storage location.</td><td>1</td><td>X</td></tr>
<tr><td>X</td><td>1</td></tr>
<tr><td rowspan="2" colspan="5">Array B

One Category 2 assembly with three Category 4 assemblies.</td><td>4</td><td>4</td></tr>
<tr><td>4</td><td>2</td></tr>
<tr><td rowspan="2" colspan="5">Array C

Four Category 3 assemblies.</td><td>3</td><td>3</td></tr>
<tr><td>3</td><td>3</td></tr>
<tr><td rowspan="5">Array D

Eleven Category D assemblies arranged in an array of four assemblies by three assemblies. One storage cell along the four storage cell wide side of the outside of the Array must remain empty. The storage array must have at least one row of empty cells between it and any other array (Array A, B, C, or D). A row of empty cells are not needed on any section of the configuration face adjacent to the SFP wall.</td><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>X</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>D</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>D</td><td>D</td><td>D</td><td>D</td><td>X</td></tr>
<tr><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td colspan="7">Notes:
1. Any storage array location designated for a fuel assembly may be replaced with non-fissile material or an empty (water-filled) cell.
2. Empty locations designated with an X must remain completely empty.
3. Storage requirements are determined for different fuel types (RFA/STD and OFA) for Arrays A through C. Only RFA/STD fuel is evaluated for storage in Array D.</td></tr>
</table>

Figure 5-2 Allowable Storage Arrays

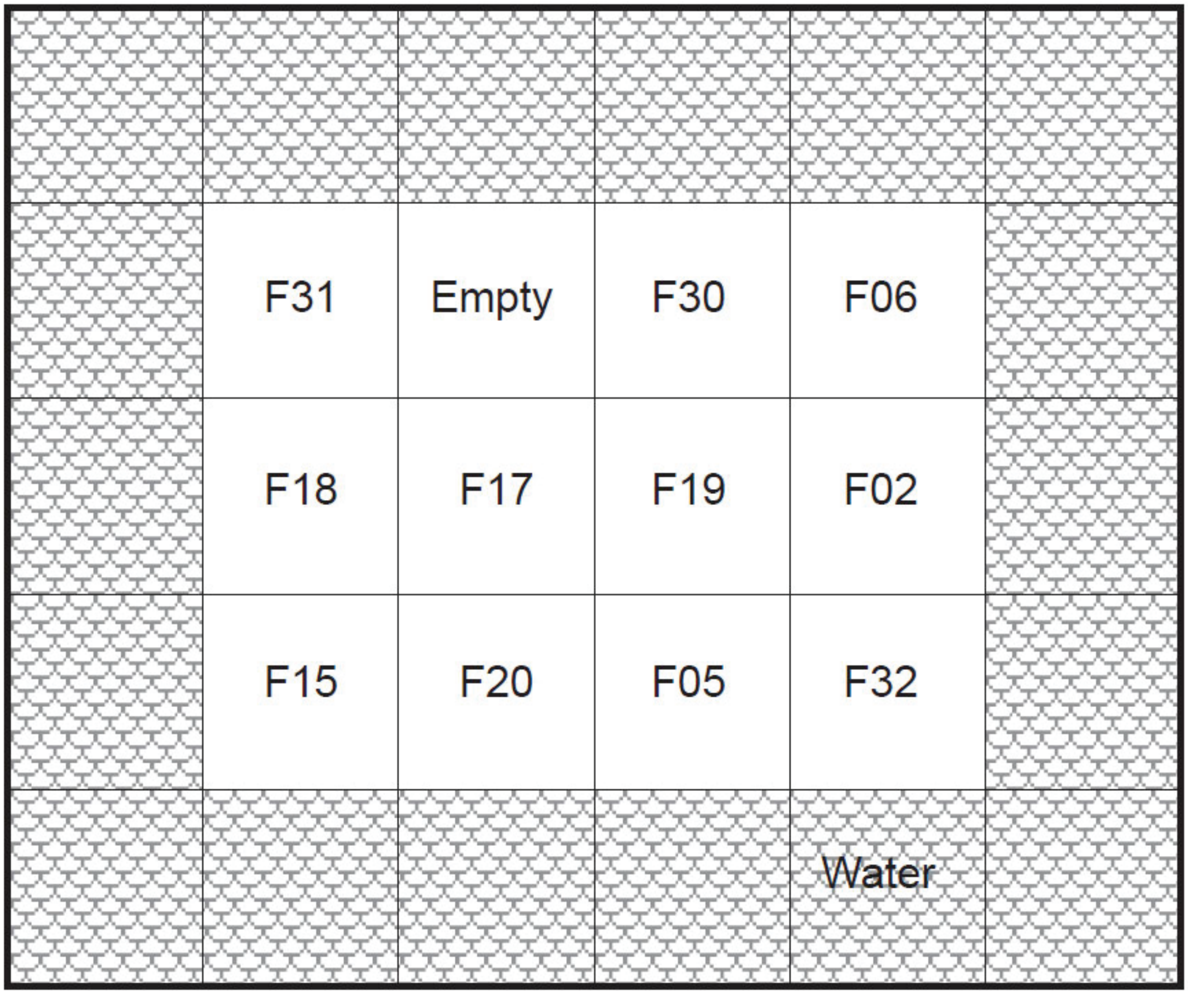

Figure 5-3 Currently Used Damaged Fuel Assembly Configuration (Farley Unit 1)

5.2.2 Target k_{eff} Calculation Description

As discussed in Section 2.1, this analysis provides burnup requirements such that the Farley Units 1 & 2 SFPs remain subcritical in unborated conditions. To ensure that the burnup requirements generated are appropriate, a target k_{eff} value is created for each array at different enrichments (maximum fresh, 3, 4 and 5 wt%, ^{235}U). The target k_{eff} value accounts for the reactivity effect of applicable biases and uncertainties and includes administrative margin to ensure safety as shown in Equation 5-1.

$$\text{Target } k_{eff} = Acceptance\ Criterion - Admin\ Margin - \sum(Biases\ \&\ Uncertainties) \quad \text{Equation 5-1}$$

where,

Acceptance Criterion	=	the maximum allowable k_{eff} for a storage array (see Section 2.1)
Admin Margin	=	the administrative margin (0.005 Δk) taken to provide additional certainty of safe operation
Σ(Biases & Uncertainties)	=	the amount of reactivity that accounts for biases and uncertainties in the reactivity calculation for each storage array

The sum of biases are simply additive while the sum of uncertainties are statistically added as the root sum square of the individual reactivity uncertainties.

5.2.3 Bias & Uncertainty Calculations

Reactivity biases are known variations between the real and analyzed system and their reactivity impact is added directly to the calculated k_{eff}. Examples include the SFP temperature and code validation biases. Uncertainties account for allowable variations within the real model whether they are physical (manufacturing tolerances), analytical (depletion uncertainty and validation bias uncertainty), or measurement related (burnup measurement uncertainty). Biases have a greater impact due to their direct addition to the total sum of bias and uncertainty. Uncertainties are statistically added as the root sum square of the individual reactivity uncertainties.

[

]^{a,c}

5.2.3.1 Bias & Uncertainty Descriptions including Manufacturing Tolerances

Reactivity biases and uncertainties as a result of manufacturing tolerances and other SFP characteristics are discussed in this section and the following subsections. KENO is used to quantify reactivity effects.

[

-
-
-
-
-
-
-
-
-
  -
  -
  -

]a,c

5.2.3.1.1 Cladding Tolerance Reactivity Uncertainty

[

]^{a,c}

5.2.3.1.2 Initial Fuel Enrichment Reactivity Uncertainty

[

]^{a,c}

5.2.3.1.3 Guide Tube and Instrument Tube Reactivity Uncertainty

[

]^{a,c}

5.2.3.1.4 Burnup Measurement Uncertainty

[

]^{a,c}

5.2.3.1.5 Depletion Uncertainty

The depletion uncertainty takes into account the potential reactivity misprediction of the depletion code.
[

]a,c

5.2.3.1.6 Operational Uncertainty

[

]a,c

5.2.3.1.7 Flux Trap Gap Reactivity Uncertainty

The flux trap gap tolerance worth is not explicitly calculated; however the rack cell pitch tolerance cases explicitly include the effect of cell pitch and flux trap gap tolerance since the rack pitch would change the flux gap width and vice versa. The rack pitch tolerance was chosen as it is the larger change.

5.2.3.1.8 Borated Sheath Width Reactivity Uncertainty

Reference 14 indicates that for flux-trap rack designs, the uncertainty due to the manufacturing tolerance on the sheathing width is small but cannot generically be declared negligible. Despite not crediting the borated insert, sheath width tolerance reactivity uncertainty is determined.

5.2.3.1.9 Borated Insert Cavity Width Uncertainty

In addition to the borated sheath width reactivity uncertainty, the borated insert cavity width uncertainty is determined.

5.2.3.1.10 Other Uncertainties

An uncertainty in the predictive capability of Scale 6.2.3 and the associated cross-section library is considered in the analysis. The uncertainty from the validation of the calculational methodology is discussed in detail in Reference 19.

5.2.3.1.11 Assembly Envelope Expansion Bias

The assembly envelope expansion bias is comprised of [

]a,c

[

]a,c

5.2.3.1.12 Fission Product and Minor Actinide Worth Bias

A common approach to the validation of cross-sections is by benchmarking critical experiments that are designed to closely represent the configurations of the desired criticality application. The validation of fission products, however, is more difficult because few critical experiments are available. Due to the limited availability of fission product benchmark data, a factor of uncertainty was considered in the criticality safety analysis.

NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k_{eff}) Predictions" (Reference 13) presents findings that show for minor actinide and fission product nuclides for which adequate critical experiment data are not available, calculations of k_{eff} uncertainty due to nuclear data uncertainties can be used to establish a bounding bias value which was approximately 1.5 percent of the worth of the minor actinides and fission products.
[

]a,c

5.2.3.1.13 Eccentric Fuel Assembly Positioning Bias

The fuel assemblies are assumed to be nominally located in the center of the storage rack cell; however, it is recognized that an assembly could in fact be located eccentrically within its storage cell. Reference 14 indicates that assembly eccentric positioning should be considered in racks without absorber panels. Racks in this analysis contain two absorber panels between each storage location, however, they are not credited in this analysis, so an eccentric positioning bias is determined.

To quantify the reactivity effects of eccentrically located fuel within a fuel storage cell, [

][a,c]

a,c

Figure 5-4 [ ][a,c]

5.2.3.1.14 SFP Temperature Bias

The Farley Units 1 & 2 SFPs do not have a nominal temperature; instead it operates within an allowable range. [

]a,c

5.2.3.1.15 Borated and Unborated Biases and Uncertainties

Technical Specifications require each SFP to have k_{eff} to be < 0.95 under borated conditions accounting for all applicable biases and uncertainties. [

]a,c

5.2.3.2 Storage Array Biases & Uncertainties Results

Tables 5-4 through 5-10 give the calculated biases and uncertainties for Array A, B, and C, and D for STD and OFA fuel as well as the total sum of biases and uncertainties and administrative margin for determination of the listed Target k_{eff} values. Note that for Array B, the initial enrichments shown at the top of Table 5-6 for STD fuel and Table 5-7 for OFA fuel correspond to Fuel Category 4. [

]a,c Fuel Category 2 assemblies are fresh fuel assemblies up to 5 wt%, with fresh IFBA requirements discussed in Section 6.1.

Table 5-4 Biases & Uncertainties for Array A with STD Fuel

a,c

□

Table 5-5 Biases & Uncertainties for Array A with OFA Fuel

a,c

a,c

Table 5-6 Biases & Uncertainties for Array B with STD Fuel

Table 5-7 Biases & Uncertainties for Array B for OFA Fuel

a,c

a,c

Table 5-8 Biases & Uncertainties for Array C with STD Fuel

Table 5-9 Biases & Uncertainties for Array C with OFA Fuel

a,c

a,c

Table 5-10 Biases & Uncertainties for Array D with STD Fuel

5.3 INTERFACE CONDITIONS

Interfaces are the locations where there is a change in either the storage racks or the storage requirements of the fuel in question. In this analysis, only intra-region interfaces are evaluated since all racks are of the same design and no pool region interfaces are present. In addition to the intra-region interfaces, Array D assemblies are required to have a row of empty storage cells (or the pool wall) face adjacent to all sides of the storage array.

The only interface conditions that need to be addressed in this analysis are those between different fuel storage arrays. [

]^{a,c}

a,c

Figure 5-5 [ ]^{a,c}

Additionally, Array D contains 11 fuel storage locations generically evaluated for storage within the Farley SFPs. One storage cell along the four storage cell wide side of the outside of the Array must remain empty i.e. water-filled. The storage array must have at least one row of empty cells between it and any other array (Array A, B, C, or D). A row of empty cells is not needed on any section of the configuration face adjacent to the SFP wall.

5.4 NORMAL CONDITIONS

This section discusses normal conditions within the SFPs which are in addition to the steady-state storage of fresh and spent fuel assemblies. During normal operation, the SFPs have a soluble boron concentration of greater than 2000 ppm and a moderator temperature ≤ 185ºF. Beyond the storage of fuel assemblies, there are five major types of normal conditions covered in this analysis. These five conditions are explained in subsections 5.4.1 through 5.4.5.

5.4.1 Type 1 Normal Conditions

Type 1 conditions involve the placement of components in the guide tubes and/or instrument tube of intact fuel assemblies while normally stored in the storage racks. This also includes removal and reinsertion of these components into the fuel when stored in the rack positions using specifically designed tooling. Examples include control rods, neutron sources, guide tube probing, fuel assembly guide tube length measurement, and ultrasonic test equipment being placed in a testing location on top of a spent fuel storage rack.

The Type 1 normal conditions typically include the insertion of components into fuel assemblies for storage in the SFPs (e.g., depleted Pyrex). The SFPs as single systems are over moderated. A single fuel assembly however, is significantly undermoderated, and reducing the interstitial hydrogen to uranium ratio lowers the system k_{eff} as seen by the fact that all rod pitch uncertainty cases show that a reduction in rod pitch reduces reactivity. Additionally, calculations have been performed which show that [

]^{a,c} Any components designed to be inserted into an assembly may be stored in a fuel assembly in the SFPs.

5.4.2 Type 2 Normal Conditions

Type 2 conditions involve evolutions or transitional fuel assembly actions where the fuel assembly is removed from its normal storage rack location for a specific procedure and reinserted after the completion of the procedure. Examples of Type 2 conditions include fuel assembly visual inspection, reconstitution, cleaning and sipping. During the Type 2 assembly evolutions only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies. Descriptions of each of these items are provided, along with the evaluation of the impact on this criticality safety analysis.

One cell pitch of separation is defined as one rack pitch distance away from all sides of the assembly (including both face adjacent and corner adjacent cells). Outside of a storage cell both pertain to fuel which has been removed to a location outside of the storage rack as well as to the areas of an assembly exposed above the rack due to partial insertion.

Fuel assembly cleaning is defined as placing cleaning equipment adjacent to a single assembly and either jetting water from or into a nozzle. The cleaning equipment will displace water adjacent to the assembly and can use demineralized (unborated) water to clean assemblies. The demineralized water used in this process is not confined to a particular volume but would be readily dispersed into the bulk water of the SFP. In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies. The large delta between the Technical Specification required boron concentration and the boron concentration credited in this analysis and the relatively small volume of demineralized water used for this operation guarantees that the addition of unborated water does not constitute a significant dilution event.

Fuel assembly inspection is defined as placing non-destructive examination equipment against at least one face of an assembly. Periscopes and underwater cameras can be placed against all four faces of the assembly simultaneously and will displace water. In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies.

Fuel assembly reconstitution involves rod movement from and/or to an assembly. In most cases, damaged rods will be replaced with SS rods, but natural uranium rods may also be used. If a rod is replaced with either SS or a rod made of natural uranium the reactivity of the available fissile material of the single assembly will be decreased while capable moderation remains the same resulting in a reduction in reactivity. If the fuel rod in question is either replaced by a fuel rod from another assembly or the fuel rod is to be removed without replacement, adjustments must be made to the burnup storage requirement of the assembly being reconstituted.

[

Fuel assembly sipping is defined as placing one fuel assembly in the sipping equipment. The fuel assembly is separated from all other stored fuel by at least one assembly pitch via the equipment design. While the sipping equipment can be placed within one assembly pitch adjacent to a storage rack loaded with fuel, the fuel assembly loaded into the sipping equipment must be more than one assembly pitch removed from the fuel located in the storage racks. During this operation, demineralized water may be introduced to the sipping container, exposing the assembly(s) to an unborated environment.

Fuel assembly cleaning, inspection, reconstitution, and sipping are bounded by this criticality analysis.
[

]a,c

5.4.3 Type 3 Normal Conditions

Type 3 conditions involve insertion of components that are not intact fuel assemblies, into the fuel storage rack cells. For Farley Units 1 & 2 SFPs, these include a loose pellet canister as well as a failed fuel rod storage canister. Additionally, any components that do not contain fissile materials can replace a fuel assembly of any fuel category in one of the approved storage configurations described in Section 5.2.1.

The fuel rod storage canister at Farley Units 1 & 2 is a rectangular lattice of storage tubes for failed fuel rods arranged in an 8 x 8 pattern. Not all rows contain 8 tubes as can be seen from the modeled schematic in Figure 5-6. Table 3-5 contains pertinent design information.

a,c

[

]

Figure 5-6 Schematic View of Modeled Failed Fuel Rod Storage Canister

[

]a,c

[

]^{a,c}

The loose pellet transport canisters (LPTCs) are SS canisters designed to store up to 5000 loose fuel pellets. Design details are given in Table 3-6. Modeling of the LPTCs [

]^{a,c}

a,c

Figure 5-7 [ ]^{a,c}

[

]^{a,c}

5.4.4 Type 4 Normal Conditions

Type 4 normal conditions include temporary installation of non-fissile components on the rack periphery facing the pool wall. Analyses of the storage arrays contained within this criticality analysis assume an infinite array of storage cells. This assumption bounds the installation of any non-fissile components on the periphery of racks.

5.4.5 Type 5 Normal Conditions

Type 5 conditions involve miscellaneous conditions that do not fit into the first four normal condition types. Examples include usage of fuel handling tools for their intended purpose, miscellaneous debris under the storage racks, and damaged storage cells.

A damaged storage cell is defined as a cell where the cell liner is out of tolerance or the entry channel has been damaged. These cells should not be used to store fuel assemblies, but they may be used to store items that need to be stored as a fuel assembly (i.e., non-fissile material or a fuel rod canister, etc.).

Insertion of handling tools into the top of fuel assemblies or other components occurs frequently in the SFP environment. The insertion of handling tools into the top of an assembly is bounded by the storage of inserts in fuel assemblies and therefore, from a criticality perspective, all fuel handling tools are acceptable for their intended purpose.

Performance of Foreign Object Search and Retrieval (FOSAR) from fuel assemblies and/or storage cells must meet the following guidelines.

1. If a FOSAR is done on a storage cell, any fuel assembly residing in the storage cell must be removed before the action takes place.

2. For FOSAR done on a fuel assembly, if the operations do not occur in the active fuel region and do not require tooling to reside in the active fuel region, the FOSAR does not impact criticality and the assembly can remain in its storage cell.

3. If the FOSAR requires tooling to be present in the active fuel region, then the fuel assembly must be separated from other fuel assemblies by at least one assembly pitch.

The Farley SFPs have pump/filtration systems which sit on top of the fuel racks. These systems displace water above the assembly, which is conservative in unborated conditions. Borated models for accident analysis use unborated water above and below the active fuel before a reflective boundary is applied. In addition, the separation from the fuel rods due to the top nozzle and the fuel rod end plug is sufficient to prevent significant neutron interaction. Therefore, there is no restriction on the location of the filtration systems.

5.5 SOLUBLE BORON CREDIT

Section 2.1 contains k_{eff} requirements under both the assumption that the pool is flooded with pure water and that the pool contains soluble boron. This section outlines the calculations that were performed to demonstrate the soluble boron concentration necessary to meet the soluble boron requirements in Section 2.1. In reporting the soluble boron requirements, the atomic percent (at%) of ^{10}B in boron is conservatively assumed to be 19.4 to bound the potential variation in the isotopic concentration of boron within the SFPs.

5.5.1 Soluble Boron Requirements for Normal Conditions

Soluble boron credit for Normal Operating Conditions is evaluated for Farley Units 1 & 2. Additional pertinent details for modeling each storage array conservatively for the Normal Operating Conditions soluble boron determination are as follows. While additional models were evaluated with all fresh fuel, cases with the highest burnup are conservative as expected due to the reduction in boron worth with burnup due to spectral hardening. As a result, the following models are conservative for each storage array:

- Array A: Fuel assemblies were modeled as fresh 5 wt% ^{235}U fuel.
- Array B: Fuel is modeled as three 5 wt% initial enrichment fuel at 48 GWd/MTU and one fresh 5 wt% ^{235}U fuel assembly.
- Array C: Fuel is modeled as 5 wt% initial enrichment at 34 GWd/MTU.
- Array D: Fuel is modeled as 3 wt% ^{235}U initial enrichment at 4 GWd/MTU

To determine the maximum soluble boron concentration for normal conditions to meet a 95/95 k_{eff} of < 0.95 including biases and uncertainties, where 95/95 k_{eff} is defined as

$$95/95\ k_{eff} = \text{KENO}\ k_{eff} + 2\sigma_{k_{eff}} + \text{B\&U} + \text{Adm. margin}, \qquad \text{Equation 5-8}$$

where:

KENO k_{eff} = The simulated accident condition k_{eff}
$\sigma_{k_{eff}}$ = The simulated accident condition k_{eff} Monte Carlo simulation standard deviation
B&U = The total bias and uncertainty term[1]
Adm. Margin = Administrative margin.

The minimum soluble boron concentration to maintain k_{eff} < 0.95 for the limiting normal condition including biases, uncertainties, and administrative margin is 320 ppm, conservatively rounded up from value determined from linear interpolation (plus rounding) of Array B is 270 ppm. Results are given in Table 5-11 for STD/RFA fuel and in Table 5-12 for OFA fuel for all storage array.

[1] Biases and uncertainties are taken from the nominal storage condition for all storage arrays.

a,c

Table 5-11 [ ]a,c

a,c

Table 5-12 [ ]a,c

5.5.2 Soluble Boron Requirements for Accident Conditions

In addition to maintaining k_{eff} not to exceed 0.95 during normal operations, soluble boron is used to offset the potential reactivity insertion events in the SFPs. The following accidents are considered in this analysis:

- Assembly misload
- SFP temperature greater than normal operating range (> 185°F)
- Dropped & misplaced fresh fuel assembly
- Seismic event

5.5.2.1 Assembly Misload

This section addresses the potential for an assembly or assemblies to be placed in a storage cell location, which is not allowed by the burnup requirements in Section 6.1, in addition to an assembly misloaded between the SFP storage rack and concrete wall. This analysis addresses both the misload of a single assembly into an unacceptable storage location and multiple assemblies being misloaded in series into unacceptable storage locations.

5.5.2.1.1 Single Assembly Misload

[

]a,c

5.5.2.1.2 Multiple Assembly Misload

A multiple assembly misload is a postulated accident where assemblies are misloaded in series due to a common cause. [

]a,c

Table 5-13 [ ]a,c

a,c

5.5.2.2 Spent Fuel Temperature Outside the Normal Operating Range

The J.M. Farley Units 1 and 2 SFPs are to be operated at less than 185°F. However, under accident conditions this temperature could be higher. [

]^{a,c}

5.5.2.3 Dropped & Misplaced Fresh Assembly

During placement of the fuel assemblies in the racks, it is possible to drop the fuel assembly from the fuel handling machine. The dropped assembly could land horizontally on top of the other fuel assemblies in the rack. [

]^{a,c}

5.5.2.4 Seismic Event

In the event of an earthquake or similar seismic event, the SFPs storage racks can shift position. This can cause the rack modules to slide together eliminating the space between modules and between modules and the SFP wall. [

]^{a,c}

5.5.3 Soluble Boron Requirements

Table 5-14 data indicates that [

]a,c

Table 5-14 [ ]a,c

a,c

5.6 RODDED OPERATION

While standard operation is performed unrodded, it is allowable to operate at hot full power with rods inserted to the power dependent insertion limits. Operating with control rods inserted into the core impacts the assemblies in the rodded locations. The insertion of a control rod into an assembly during operation has several effects.

The reactivity of an assembly experiencing rodded operation can increase relative to an assembly which does not experience rodded operation. The loss of moderator as water displaced in the GTs when a control rod is inserted into the assembly will harden the neutron spectrum increasing plutonium production. The control rod will also preferentially absorb thermal neutrons, further hardening the neutron spectrum. In addition to the spectral hardening, the control rod will lower the power in the area of the assembly where it is inserted. This will lower the burnup accumulated in the top of the assembly, increasing the end effect. These effects can all increase the reactivity of an assembly, making it possible for an assembly operated with rods inserted to be more reactive than an assembly of the same assembly average burnup which experienced unrodded operation.

While these items can increase reactivity, there are competing effects which reduce assembly reactivity due to rod insertion. When a control rod is inserted into an assembly, the power in that assembly will be reduced. This will reduce both the fuel and moderator temperatures. The reduction in fuel temperature will decrease Doppler broadening leading to less neutron capture by ^{238}U, thus lowering plutonium production. The reduction in moderator temperature will increase moderator density, increasing neutron moderation and therefore softening the neutron spectrum.

In addition to impacting the neutron spectrum, rodded operation can also affect the axial burnup profile of assemblies. Operation with a control rod inserted in an assembly will shift power down, under-depleting the top of the assembly while the control rod is present. Once the control rod has been withdrawn from the assembly, power preferentially moves to the under-depleted top of the assembly, and over time the axial burnup profile developed will return to a profile typical of unrodded operation. Therefore, time-in-life before final discharge of an assembly is an important factor in the impact of rodded operation on assembly reactivity.

NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit" (Reference 15) defines a significant amount of control rod insertion as more than 20 cm into the core.

Farley Units 1 and 2 have not operated at full power with control rods inserted a significant length into the core. Therefore, there is no significant burnup accrued during depletion with rods inserted in the active fuel height, and no need to account for these effects in burnup limits contained within this analysis. Any assemblies incurring significant rodded operation going forward must not credit the rodded burnup.

While typical operation for Farley Units 1 & 2 is performed unrodded, there is potential to operate at reduced power levels with rods inserted. Short term reduced power operation may be the result of plant equipment issues or economic considerations and has occurred at Farley Units 1 & 2. Any impact from short term operation at reduced power levels with rods inserted will be negligible.

6 ANALYSIS RESULTS & CONCLUSION

This section documents the final storage results of the Farley Units 1 & 2 Spent Fuel Pool criticality safety analysis. Included in this section are the burnup requirements for the fuel storage arrays documented in this analysis. This section also contains the Area of Applicability of this analysis. The Area of Applicability (AoA) of the criticality code validation suite is discussed in Reference 19.

6.1 BURNUP AND IFBA REQUIREMENTS FOR STORAGE ARRAYS

Assembly storage is controlled through the storage arrays defined in Section 5.2.1. An array can only be populated by assemblies of the fuel category defined in the array definition or a lower reactivity fuel category (see Table 5-3). Fuel Category 1 does not require burnup or fresh IFBA for storage. Fuel Category 2 assembly storage requirements require that they either must have not been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the IFBA requirements coefficients or have at least 10.0 GWd/MTU of exposure covering the peak reactivity of IFBA bearing assemblies with 5 wt% ^{235}U enrichment. (No IFBA requirements are needed beyond 10.0 GWd/MTU). Fresh IFBA requirements coefficients as well as sample IFBA requirements are given in Table 6-1 and Table 6-2 for STD/RFA fuel and in Table 6-3 and Table 6-4 for OFA fuel. Fuel categories D, 3, and 4 are defined by assembly average burnup, initial enrichment[1], and decay time with burnup requirement coefficients and sample evaluated burnup limits given in Table 6-5 through Table 6-14.

This analysis has provided burnup requirements at discrete decay times, measured in years. However, it is acceptable to interpolate between these decay times to determine burnup requirements at alternate decay times. Using linear interpolation between two already analyzed decay times will give a conservative burnup requirement for the decay time in question. Linear interpolation based on actual decay time should be performed between calculated values of minimum burnup associated with tabulated decay times greater and less than the actual decay time. No extrapolation beyond 20 years is permitted. This is acceptable because isotopic decay is an exponential function which means assembly reactivity will decay faster than the calculations using linear interpolation would predict.

1 Initial enrichment is the maximum nominal enrichment of the fuel, prior to reduction in ^{235}U content due to fuel depletion. If the fuel assembly contains axial regions of different ^{235}U enrichment values, such as axial cutbacks, the maximum initial enrichment value is to be used.

Table 6-1 Fuel Category 2: STD/RFA IFBA Fitting Coefficients

	Fitting Coefficients		
IFBA Thickness	A1	A2	A3
1.00X	5.2750	8.3325	-79.9546
1.25X	3.7476	10.8046	-72.0974
1.50X	1.8593	19.8050	-81.5075

Notes:

1. For a fuel assembly to meet the requirements the assembly must either:
   a. Not have been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the curve fit for the assembly "initial enrichment," or,
   b. Have at least 10.0 GWd/MTU of exposure.
2. The specific minimum IFBA required for each fuel assembly is calculated from the following equation:

$$\# \text{ of IFBA Rods} = A1 * En^2 + A2 * En + A3$$

3. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment greater than 3.2 wt% ^{235}U and less than or equal to 5 wt% ^{235}U may be used. The number of IFBA rods required must be rounded up to the next whole number. Below 3.2 wt% ^{235}U, IFBA is not required.

Table 6-2 Fuel Category 2: Example STD/RFA IFBA Requirements (# of IFBA Rods)

IFBA Thickness	Average Initial Enrichment, wt% ^{235}U				
	3.2	**3.8**	**4.2**	**4.6**	**5**
1.00X	1	28	49	70	94
1.25X	1	24	40	57	76
1.50X	1	21	35	49	64

Note:
1. The values provided in this table are provided as an example. The requirements must be calculated using the coefficients in Table 6-1.

Table 6-3 Fuel Category 2: OFA IFBA Fitting Coefficients

	Fitting Coefficients		
IFBA Thickness	A1	A2	A3
1.00X	6.2658	0.8890	-65.4949
1.25X	3.9144	9.3963	-68.9414
1.50X	1.5898	21.8436	-84.9630

Notes:

1. For a fuel assembly to meet the requirements the assembly must either:
   a. Not have been operated in the reactor and the IFBA loading must exceed the "minimum IFBA" (# rods per assembly) given by the curve fit for the assembly "initial enrichment," or,
   b. Have at least 10.0 GWd/MTU of exposure.
2. The specific minimum IFBA required for each fuel assembly is calculated from the following equation:

$$\# \text{ of IFBA Rods} = A1 * En^2 + A2 * En + A3$$

3. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any enrichment greater than 3.2 wt% ^{235}U and less than or equal to 5 wt% ^{235}U may be used. The number of IFBA rods required must be rounded up to the next whole number. Below 3.2 wt% ^{235}U, IFBA is not required.

Table 6-4 Fuel Category 2: Example OFA IFBA Requirements (# of IFBA Rods)

IFBA Thickness	Average Initial Enrichment, wt% ^{235}U				
	3.2	**3.8**	**4.2**	**4.6**	**5**
1.00X	2	29	49	72	96
1.25X	2	24	40	58	76
1.50X	2	21	35	50	64

Note:
1. The values provided in this table are provided as an example. The requirements must be calculated using the coefficients in Table 6-3.

Table 6-5 Fuel Category D: STD/RFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0	0	9.6344	-24.5678
5	0	0	9.4528	-24.1047
10	0	0	9.3343	-23.8025
15	0	0	9.2508	-23.5896
20	0	0	9.1965	-23.4510

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \quad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.55 wt% ^{235}U and 3 wt% ^{235}U may be used. Below 2.55 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-6 Fuel Category D: Example STD/RFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Average Initial Enrichment, wt% ^{235}U	
	2.55	**3**
0	0	4.336
5	0	4.254
10	0	4.201
15	0	4.163
20	0	4.139

Note:

1. This table is included as an example, the burnup requirements will be calculated using the coefficients provided.

Table 6-7 Fuel Category 3: STD/RFA Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.3997	-4.4670	28.2780	-44.1204
5	0.3637	-4.1462	26.6011	-41.6405
10	0.1856	-2.3309	20.2704	-34.6503
15	0.0892	-1.3905	17.0683	-31.1550
20	0.0388	-0.9253	15.5082	-29.4500

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \quad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.15 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 2.15 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-8 Fuel Category 3: Example STD/RFA Burnup Requirements (GWd/MTU)				
Decay Time (yr.)	**Radial Average Initial Enrichment, wt% ^{235}U**			
	2.15	**3**	**4**	**5**
0	0	11.303	23.101	35.558
5	0	10.667	21.702	33.173
10	0	10.194	21.016	31.630
15	0	9.944	20.579	30.574
20	0	9.795	20.262	29.809

Note:

1. This table is included as an example, the burnup requirements will be calculated using the coefficients provided.

Table 6-9 Fuel Category 3: OFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.1692	-1.8852	18.5219	-32.7830
5	0.0191	-0.4154	13.4482	-27.1777
10	-0.0705	0.4300	10.5987	-24.0722
15	-0.1420	1.1146	8.2825	-21.5440
20	-0.1959	1.6375	6.5093	-19.6130

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \quad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 2.15 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 2.15 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-10 Fuel Category 3: Example OFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	2.15	3	4	5
0	0	10.385	21.971	33.847
5	0	9.944	21.192	32.066
10	0	9.691	20.691	30.859
15	0	9.501	20.332	29.984
20	0	9.364	20.087	29.384

Note:

1. This table is included as an example, the burnup requirements will be calculated using the coefficients provided.

Table 6-11 Fuel Category 4: STD/RFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	-0.6112	4.6655	6.7127	-21.8911
5	-0.3326	2.0713	12.8468	-26.1880
10	-0.1305	0.0505	18.3242	-30.7080
15	0.1360	-2.6856	26.5239	-38.3300
20	0.2321	-3.7177	29.5977	-41.1200

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \quad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 1.7 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 1.7 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-12 Fuel Category 4: Example STD/RFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.7	**3**	**4**	**5**
0	0	23.735	40.491	51.910
5	0	22.014	37.054	48.254
10	0	21.196	35.045	45.863
15	0	20.744	33.500	44.150
20	0	20.481	32.642	42.939

Note:

1. This table is included as an example, the burnup requirements will be calculated using the coefficients provided.

Table 6-13 Fuel Category 4: OFA Burnup Requirement Coefficients

Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.3726	-4.8740	33.7329	-45.9288
5	0.6544	-7.8532	42.6520	-54.1346
10	0.8557	-9.9883	49.1073	-60.1446
15	0.9692	-11.1551	52.5353	-63.2522
20	1.1873	-13.2641	58.6586	-68.8379

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." If the computed minimum burnup value is negative, zero shall be used. The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4 \qquad [GWd/MTU]$$

2. Initial enrichment, En, is the maximum ^{235}U enrichment. Any enrichment between 1.75 wt% ^{235}U and 5 wt% ^{235}U may be used. Below 1.75 wt% ^{235}U, burnup credit is not required.
3. An assembly with a decay time greater than 20 years must use the 20-year (or less decay time) limits.

Table 6-14 Fuel Category 4: Example OFA Burnup Requirements (GWd/MTU)

Decay Time (yr.)	Radial Average Initial Enrichment, wt% ^{235}U			
	1.75	**3**	**4**	**5**
0	0	21.465	34.866	47.461
5	0	20.812	32.704	44.596
10	0	20.387	31.237	42.647
15	0	20.127	30.437	41.697
20	0	19.819	29.559	41.266

Note:

1. This table is included as an example, the burnup requirements will be calculated using the coefficients provided.

6.2 ANALYSIS AREA OF APPLICABILITY

This section details the area of applicability of the analysis concerning assembly characteristics and associated fuel management, including a summary of the data which needs to be confirmed to assure that the results presented here remain valid. Additionally, restrictions are given for other normal SFP conditions. Farley Units 1 & 2 have operated with the STD and OFA fuel designs. [

]a,c

Table 6-15 Criticality [ ]a,c

a,c

a,c

a,c

Additional restrictions for fuel storage are given here. One assembly pitch is defined as one cell in any direction, including both face adjacent and corner adjacent cells.

- Fuel assembly evolutions (fuel cleaning, inspection, reconstitution, and sipping) must occur with at least one assembly pitch of water between the assembly in question and other assemblies. It is also acceptable to perform these actions above the top of the storage racks.

- Fuel assemblies stored with one or more rods missing, leaving a water hole, need to be stored as fresh fuel.

- Fuel assemblies which have had fuel rods replaced with SS, natural uranium, or zirconium alloy rods may be stored as normal (by initial enrichment and burnup).

- Reconstituted fuel which contains fuel rods from other fuel assemblies will be controlled as follows:

    1. The fuel assembly enrichment will be assumed to be the higher of the inserted rod or reconstituted fuel assembly's initial enrichment; and
    2. The fuel assembly burnup will be assumed to be the lower of the reconstituted rod or reconstituted fuel assembly's burnup.

- In all cases, only one fuel assembly will be manipulated at a time and all manipulations will occur outside the storage cell and not within one assembly pitch of other assemblies.

- An inspection can occur within the storage racks without restriction if it does not involve unborated water and nothing occurs within the assembly envelope or below the top of the active fuel.

- Any storage cells considered damaged (outside of their allowable tolerances) cannot be used to store fuel assemblies without further evaluation. These damaged cells may be used to store non-fuel assembly components such as failed fuel baskets in a storage array.

6.3 SOLUBLE BORON CREDIT

Soluble boron is credited in the Farley Units 1 & 2 SFPs to keep $k_{eff} < 0.95$ under all normal and credible accident scenarios. Under normal conditions, this requires less than 320 ppm of soluble boron. Under accident conditions the most limiting accident is the multiple misload accident requiring 1710 ppm of soluble boron.

7 REFERENCES

1. Westinghouse Document WCAP-16045-P-A, Rev. 0, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

2. "SCALE Code System, ORNL/TM-2005/39, Version 6.2.3, March 2018.

3. EPRI Report 3002010613, "Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty—Revision 1," Adams Number ML18088B397, Electric Power Research Institute, 2017.

4. EPRI Report 3002010614, "Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation—Revision 1," Adams Number ML18088B395, Electric Power Research Institute, 2018.

5. V. Kucukboyaci, "EPRI Depletion Benchmark Calculations Using PARAGON," ANS NCSD, October 2013.

6. NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," U.S. Nuclear Regulatory Commission, February 2000.

7. Westinghouse Document WCAP-9522, "FIGHTH – A Simplified Calculation of Effective Temperatures in PWR Fuel Rods for Use in Nuclear Design," May 1979.

8. K.Wood, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," DSS-ISG-2010-001, Accession Number ML102220567, Nuclear Regulatory Commission, Rockville, MD, August 2010.

9. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," Oak Ridge National Laboratory, March 2003.

10. NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers on PWR Burnup Credit," Oak Ridge National Laboratory, March 2002.

11. NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for the PWR Burnup Credit," Oak Ridge National Laboratory, March 2002.

12. L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," Nuclear Regulatory Commission, Rockville, MD, August 1998.

13. NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k_{eff}) Predictions," Oak Ridge National Laboratory, April 2012.

14. NEI-12-16, Revision 3, “Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants,” March 2018.

15. C.E. Sanders, et al., “Parametric Study of the Effect of Control Rods for PWR Burnup Credit,” NUREG/CR-6759, Oak Ridge National Laboratory, Oak Ridge, TN, February 2002.

16. NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data,” U.S. Nuclear Regulatory Commission, September 2008.

17. ML19189A111, “Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report 3002010613, ‘Benchmarks for Qualifying Fuel Reactivity Depletion Uncertainty—Revision 1’ and Topical Report 3002010614, ‘Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation—Revision 1,’” U.S. Nuclear Regulatory Commission, 2019.

18. ML20104C140, “Joseph M. Farley Nuclear Plant - Units 1 and 2 Response to Request for Additional Information Regarding License Amendment Request (LAR) to Update the Spent Fuel Pool Criticality Safety Analysis and Revise Technical Specification (TS) 3.7.15 “Spent Fuel Assembly Storage” and TS 4.3 “Fuel Storage,” U.S. Nuclear Regulatory Commission, 2020.

19. Westinghouse Document Appendix A of WCAP-18414-NP, Revision 0, “J. M. Farley Units 1 & 2 Spent Fuel Pool Criticality Safety Analysis,” September 2019.

Attachment 5

Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-22-039, accompanying Affidavit, and Proprietary Information Notice

Commonwealth of Pennsylvania:
County of Butler

(1) I, Zachary Harper, Manager, Licensing Engineering, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of WCAP-18414-P, R3 "J. M. Farley Units 1 and 2 Spent Fuel Pool Criticality Safety Analysis" (Proprietary Version) be withheld from public disclosure under 10 CFR 2.390.

(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 10 CFR 2.390, 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 and is not customarily disclosed to the public.

   (ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

   (iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, 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.

(5) Westinghouse has policies in place to identify proprietary information. 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.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding 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 (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 8/8/2022

Signed electronically by
Zachary Harper