UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 30, 2017

Mr. Scott D. Northard
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT REVISING SPENT FUEL POOL CRITICALITY TECHNICAL SPECIFICATION (CAC NOS. MF7121 AND MF7122, EPID L-2015-LLA-0002)

Dear Mr. Northard:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 222 to Renewed Facility Operating License No. DPR-42 and Amendment No. 209 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated November 17, 2015.

The amendments revise applicable TSs regarding spent fuel pool criticality safety analysis for Prairie Island Nuclear Generating Plant, Units 1 and 2.

The NRC staff has determined that its safety evaluation (SE) for the subject amendment contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version. Both versions of the SE are enclosed.

Enclosure 3 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 3, this document is decontrolled.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Kimberly J. Green for

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 222 to DPR-42
2. Amendment No. 209 to DPR-60
3. Proprietary Safety Evaluation
4. NonProprietary Safety Evaluation

cc without Enclosure 3: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
License No. DPR-42

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated November 17, 2015, as supplemented by letters dated May 23, 2016, February 16, 2017, and October 4, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: November 30, 2017

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-60

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated November 17, 2015, as supplemented by letters dated May 23, 2016, February 16, 2017, and October 4, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: November 30, 2017

ATTACHMENT TO LICENSE AMENDMENT NOS. 222 AND 209

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Renewed Facility Operating License Nos. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3
Page 3	Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.7.16-1	3.7.16-1
4.0-7	4.0-7
4.0-10	4.0-10

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Storage Pool Boron Concentration

LCO 3.7.16 The spent fuel storage pool boron concentration shall be ≥ 2500 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel storage pool boron concentration to within limit.	Immediately

Table 4.3.1-3 (page 1 of 1)
For Fuel Not Operated In Units 1 and 2 Cycles 1 - 4
Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a
Function of Decay Time and Enrichment (En)

FUEL CATEGORY	DECAY TIME	COEFFICIENTS			
		A_1	A_2	A_3	A_4
2	0	-1.1640	15.1916	-56.7743	65.2736
3	0	-0.2213	2.6959	-0.9136	-11.0959
	5	-0.2568	2.9933	-2.0421	-9.5730
	10	-0.3012	3.4074	-3.5247	-7.7578
	15	-0.2790	3.1007	-2.4261	-8.9334
	20	-0.2959	3.2578	-3.0233	-8.1560
4	0	1.3659	-14.9709	63.0347	-72.9223
5	0	0.1255	-1.6774	20.7491	-31.8434
	5	-0.0520	0.0723	14.5901	-25.4754
	10	0.1681	-2.2188	21.4991	-31.7286
	15	-0.3431	3.0482	4.0932	-14.6591
	20	-0.2576	2.2345	6.1980	-16.3085
6	0	0.6666	-7.4900	41.2094	-51.6844
	5	0.5686	-6.3968	36.4332	-46.2433
	10	0.3895	-4.5024	29.6132	-39.2399
	15	0.1962	-2.5813	23.2107	-32.9620
	20	0.1192	-1.7984	20.4749	-30.1950

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 of a Fuel Category, the assembly burnup must exceed "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment". The specific minimum burnup required for each fuel assembly is calculated from the following equation for each increment of decay time:

$$Bu = A_1*En^3 + A_2*En^2 + A_3*En + A_4$$

2. Initial enrichment (En) is the nominal U-235 enrichment. Any enrichment between 1.7 and 5.0 weight percent U-235 may be used. If the computed Bu value is negative, zero shall be used.

3. Decay Time is in years. An assembly with a cooling time greater than 20 years must use 20 years. No extrapolation is permitted.

4. If Decay Time value falls between increments of the table, the lower Decay Time value shall be used or a linear interpolation may be performed as follows: Compute the Bu value using the coefficients associated with the Decay Time values that bracket the actual Decay Time. Interpolate between Bu values based on the increment of Decay Time between the actual Decay Time value and the computed Bu results.

5. This table applies to fuel assemblies that were not operated in the Unit 1 or Unit 2 core during operating Cycles 1 through 4.

Notes:

1. In all arrays, an assembly of higher Fuel Category number can replace an assembly designated with a lower Fuel Category number.

2. Category 1 is fuel up to 5.0 weight percent U-235 enrichment and does not credit burnup.

3. Fuel Categories 2 through 6 are determined from Tables 4.3.1-2 or 4.3.1-3.

4. An "R" designates a location that requires insertion of an RCCA in the fuel assembly.

5. An "X" designates a location that requires an empty cell, except that the empty cells in Array F may store assembly structural materials including nozzles, guide tubes, and grids.

6. An empty (water-filled) cell may be substituted for any fuel-containing cell in all storage arrays.

7. Array F shall only interface with Array A, and no other.

8. Except for the center rodded assembly of the 3x3 Array G and the special interface defined between Array A and Array F, each assembly location is part of up to four 2x2 arrays (assembly in the lower right, lower left, upper right, upper left) and each assembly must simultaneously meet the requirements of all those arrays of which it is a part.

9. Category 7 is reserved for the fuel that was consolidated in the spent fuel consolidation demonstration project described in Updated Safety Analysis Report Section 10.2.

Figure 4.3.1-1 (page 3 of 3)
Allowable Storage Arrays

ENCLOSURE 4

NONPROPRIETARY SAFETY EVALUATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR 42

AND AMENDMENT NO. 209 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

~~This document contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390. Proprietary information is identified by underlined text (in red font) enclosed within double brackets as shown here [[example proprietary text]].~~

Enclosure 4

1.0 INTRODUCTION

By application dated November 17, 2015, Agencywide Documents Access and Management System (ADAMS) Accession No. ML15327A244), as supplemented by letters dated May 23, 2016 (ADAMS Accession No. ML16144A804), February 16, 2017 (ADAMS Accession No. ML17047A454), and October 4, 2017 (ADAMS Accession No. ML17279A120), Northern States Power Company, a Minnesota Corporation (NSPM, the licensee) requested changes to the technical specifications (TSs) for Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 5, 2016 (81 FR 19648).

The proposed changes would revise TSs 3.7.16, "Spent Fuel Storage Pool Boron Concentration," and 4.3.1, "Fuel Storage Criticality," to allow spent fuel pool (SFP) storage of fresh and spent nuclear fuel containing a boron-based neutron absorber in the form of zirconium diboride integral fuel burnable absorber (IFBA).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c) states that TSs will include items in the following categories:

10 CFR 50.36(c)(2)(ii), referring to requirements for limiting conditions for operation, states:

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

10 CFR 50.36(c)(4) states:

> *Design features.* 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 this section.

10 CFR 50.36(c)(5) states:

> *Administrative controls.* Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

10 CFR 50.68(b)(1) states:

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

10 CFR 50.68(b)(4) states:

> If no credit for soluble boron is taken, the k-effective [K_{eff}] of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Since approval is being requested for changes to fresh and spent fuel storage requirements pertaining to the SFP racks, 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) are not considered in this evaluation because these requirements address fresh fuel in fresh fuel storage racks.

10 CFR 2.108(a) states:

> The Director, Office of Nuclear Reactor Regulation, Director, Office of New Reactors, or Director, Office of Nuclear Material Safety and Safeguards, as appropriate, may deny an application if an applicant fails to respond to a request for additional information within thirty (30) days from the date of the request, or within such other time as may be specified.

3.0 TECHNICAL EVALUATION

In a letter dated August 29, 2013, the NRC staff issued Amendment No. 209 to Renewed Facility Operating License No. DPR-42 and Amendment No. 196 to Renewed Facility Operating License No. DPR-60 for the PINGP, Units 1 and 2, respectively (ADAMS Accession No. ML13241A383). Those amendments revised TS 3.7.17, "Spent Fuel Pool Storage," and TS 4.3.1, "Fuel Storage Criticality," to provide new SFP loading restrictions that meet subcriticality for all postulated conditions at the time of submittal. Those TS changes corrected non-conservatisms in the SFP criticality safety analysis-of-record. The amendments also changed the evaluation methodology used for the SFP criticality safety analysis (CSA).

The license amendment request (LAR) submitted on November 17, 2015, proposed changes to the evaluation methodology applicable to fuel operated after Cycle 4 including a new fuel design to be used in future cycles. The LAR included a proprietary and nonproprietary version of WCAP-17400, Supplement 1, Revision 1, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis – Supplemental Analysis for the Storage of IFBA Bearing Fuel" (WCAP-17400, S1, R1), as enclosures. In its October 4, 2017 letter, the licensee provided WCAP-17400, Supplement 1, Revision 2 (WCAP-17400, S1, R2), which included methodology updates consistent with the licensee's response to a request for additional information (RAI). WCAP-17400, S1, R2, Section S6, "Comparisons with 'Fuel Not Operated in Cycles 1-4' Burnup Limits," summarizes several methodology changes. The NRC staff considered those changes and identified a few more for completeness relative to the previously approved SFP CSA as follows:

- Presence of both IFBA and gadolinium (Gd) as burnable absorbers (BAs) in the same assembly during depletion[1]
- Change in the uncertainty evaluation[2]
- Update of fuel assembly misload analysis
- Change in input of cycle average soluble boron during depletion from 900 parts per million (ppm) to 1000 ppm
- Use of IFBA bearing fuel specific input for axial burnup profile and moderator temperature profile input
- Modeling of annular blanketed fuel as annular fuel instead of solid for both depletion and criticality calculations

The proposed changes in methodology require revisions to TS 3.7.16, "Spent Fuel Pool Boron Concentration," and TS 4.3.1. In TS 3.7.16, the amount of soluble boron required to be maintained in the SFP would increase from greater than or equal to 1800 ppm to greater than or equal to 2500 ppm. In TS 4.3.1, Table 4.3.1-3, "For Fuel Not Operated in Units 1 and 2 Cycles 1-4, Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (BU) as a Function of Decay Time and Enrichment (En),"[3] would be updated.

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[1] As opposed to only the presence of Gd as a BA within the fuel during depletion.

[2] Changes affect treatment of fission product and actinide worth, eccentric positioning, fuel rod pitch, and fuel assembly spacer grid growth.

[3] This table is only for fuel not operated in Units 1 and 2 Cycles 1-4. For fuel that was operated in Units 1 and 2 Cycles 1-4, TS 4.3.1, Table 4.3.1-2, "For Fuel Operated in Units 1 and 2 Cycles 1-4, Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Decay Time and Enrichment (En)," which is unaffected in the updated analysis, applies.

The licensee also submitted in Enclosure 6 to its November 17, 2015, LAR an evaluation of how Westinghouse's primary neutronic codes (i.e., a lattice code and a nodal core simulator code), which support the SFP CSA, remain valid for modeling fuel assemblies containing both IFBA and Gd absorber rods at PINGP. The nature of this evaluation implies that these neutronic codes are valid for determining the depleted isotopic compositions of the new IFBA and Gd fuel assembly designs for SFP CSA purposes (e.g., in terms of axial burnup profile determination). Enclosure 6 evaluated benchmarks in consideration of the new fuel assemblies with a mixture of IFBA and Gd fuel rods and demonstrated that the results were comparable to those from NRC-approved use of these codes for fuel depletion and core physics calculation purposes. This evaluation is discussed further in the depletion code validation subsection below.

The NRC staff reviewed the LAR, including WCAP-17400, S1, R1, identified the need for additional information, and sent an RAI in an e-mail dated April 12, 2016 (ADAMS Accession No. ML16105A006, nonpublic - proprietary). The licensee responded to the NRC staff's RAI in a letter dated May 23, 2016 (ADAMS Accession No. ML16144A804). The NRC staff also identified two additional items that required additional information and, therefore, sent another RAI by e-mail on October 31, 2016 (ADAMS Accession No. ML17158B491, nonpublic - proprietary). In its October 31, 2016, RAI, the NRC staff stated that it expected a licensee response by January 24, 2017. The RAI relates to eccentric positioning of fuel in the SFP and the reactivity effects of spacer grid growth. The staff's discussion of this RAI is presented in Section 3.1.1.1 of this safety evaluation (SE). By letter dated February 16, 2017 (ADAMS Accession No. ML17047A454), the license provided a response to the NRC staff's October 31, 2016, RAI, stating that "NSPM is working with the NRC's Licensing PM [project manager] to schedule a public meeting for further discussion of the RAIs." The NRC staff conducted the requested public meeting on March 9, 2017 (a summary of the meeting is at ADAMS Accession No. ML17093A601). By letter dated October 4, 2017 (ADAMS Accession No. ML17279A120), the licensee provided another response to the NRC staff's October 31, 2016, RAI, and modified its SFP CSA methodology to use more realistic assumptions as documented in WCAP-17400, S1, R2.

3.1 NRC Staff Evaluation

The NRC staff's review was performed consistent with Section 9.1.1 of NUREG-0800 (ADAMS Accession No. ML070570006).

On August 19, 1998, the NRC staff issued an internal memorandum containing guidance for reviewing CSAs of fuel storage at light-water reactor power plants. This memorandum is known colloquially as the "Kopp Letter," after the author, Laurence Kopp (ADAMS Accession No. ML072710248). While the Kopp Letter does not specify a methodology, it does provide some guidance on the more salient aspects of a fuel storage CSA, including computer code validation. The guidance is germane to boiling-water reactors and pressurized-water reactors, and to borated and unborated conditions. The NRC staff used the Kopp Letter for the review of the current application.

On September 29, 2011, the NRC staff issued Interim Staff Guidance (ISG) DSS-ISG-2010-01 (ADAMS Accession No. ML110620086). The purpose of the ISG is to provide updated review guidance to the NRC staff to address the increased complexity of recent SFP CSAs and operations. The staff also used ISG DSS-ISG-2010-01 for the review of the current application.

On August 25, 2016, draft guidance document Nuclear Energy Institute (NEI) 12-16, Revision 2 – DRAFT A, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants" (ADAMS Accession No. ML16155A120) became publicly available. NEI 12-16 incorporates guidance previously documented in both the Kopp Letter and DSS-ISG-2010-01, and contains additional guidance based on recent operating experience and research. Although the final revision to NEI 12-16 is currently in the process of being endorsed in an NRC regulatory guide, the guidance is relevant to both the development and review of SFP CSAs, therefore, the NRC staff also considered it for the review of the current application.

3.1.1 Spent Fuel Pool Criticality Safety Analysis Review

The analysis in WCAP-17400, S1, R2, is an update to analyses provided in previously approved License Amendment No. 209, therefore, detailed technical discussion is only provided for those areas of the analysis that have changed. If an area of the analysis is stated to be unchanged, this is also indicated for the corresponding area, but not discussed in detail.

A main objective of a SFP CSA is to determine limiting fuel mechanical and operational characteristics that will maximize the reactivity of fuel stored in the SFP so that the minimum subcritical margin can be determined in accordance with 10 CFR 50.68. This can be achieved by maximizing spectral hardening in combination with minimizing uranium (U)-235 depletion. Specifically, spectral hardening creates fissile plutonium isotopes, which increase the reactivity of the fuel assembly when stored in the SFP. Several mechanisms cause spectral hardening such as lower water moderator densities, which decreases neutron thermalization, ultimately decreasing thermal U-235 fission and leading to increased plutonium creation. The presence of BAs inserted into guide tubes displaces water, which again reduces neutron moderation, and allows for more fissionable U-238 to be converted to fissile plutonium. By their nature, BAs are efficient at absorbing thermal neutrons, preventing them from reaching the fuel, and reducing the consumption of fissile isotopes. Since BAs preferentially absorb thermal neutrons, rather than fast neutrons, this facilitates conversion of U-238 to fissile plutonium. Higher fuel temperatures also have a similar effect as higher temperatures increase Doppler broadening in the U-238 fuel resonance region of the absorption cross-section, increasing the probability of capturing neutrons, and converting U-238 to fissile plutonium. These and other considerations, such as the reactivity effects of manufacturing tolerances, are reviewed by the NRC staff to ensure that the analysis assumptions are appropriate.

The computer codes used – both the depletion code and criticality code – are also verified and validated in order to establish a high degree of confidence in the numerical results. The validation should be based on actual experimental results that take into account all relevant characteristics of the application of interest. For the depletion code validation, there are acceptable methods for determining code biases and bias uncertainties due to fuel depletion (e.g., those based on NUREG/CR-7108 (ADAMS Accession No. ML12116A124)). However, engineering judgment, as discussed in the Kopp Letter, is still relied upon. For the criticality code validation, there are extensive sets of laboratory critical experiments (LCEs), but these experiments are not generally comprehensive for most applications. To treat validation gaps (e.g., due to lack of fission products in LCEs), additional uncertainties are typically incorporated into minimum subcritical margin assessments.

3.1.1.1 Depletion Analysis

Selection of Bounding Fuel Assembly and Depletion Conditions

Bounding Fuel Assembly Design

The 422V+ fuel design was identified to be the bounding fuel assembly design based on the NRC staff's review of WCAP-17400, Revision 0. The licensee states that the 422V+ fuel design is the only design considered in WCAP-17400 S1, R2, because there are no plans to transition to another fuel type in the foreseeable future.

The current fuel design uses Gd bearing fuel rods without IFBA and conservatively models the top and bottom axial blanket regions as fully enriched solid pellets while the WCAP-17400, S1, R2, analysis evaluates the addition of IFBA bearing fuel rods and takes credit for the top and bottom axial blanket regions being fully enriched annular pellets. Crediting the annular blankets is an appropriate assumption for a limited amount of post-Cycle 4 fuel since this limited amount of post-Cycle 4 fuel actually contains annular pellets in the blanket regions as part of the fuel design at PINGP. However, the licensee assumes [[

]], explaining that this is an acceptable modeling simplification because [[

]]. The NRC staff finds this to be an acceptable modeling simplification for the PINGP SFP CSA based on NRC staff confirmatory analyses which show that [[

]]. While the licensee implies that the conservatism of [[ ]] would outweigh the potential non-conservatism of [[ ]], this had not been demonstrated by the licensee as part of WCAP-17400, S1, R2. The NRC staff's confirmatory analysis findings clarify the basis for the acceptability of modeling [[ ]]. That is, modeling [[ ]] does not significantly increase reactivity in the PINGP SFP CSA in WCAP-17400, S1, R2.[4]

The licensee chose to evaluate the new IFBA and Gd bearing 422V+ fuel assembly design, and associated depletion characteristics, separate from the current design without IFBA. The analysis uses a single set of bounding burnup and enrichment limits that would meet either the assumptions of the previously approved WCAP-17400, Revision 0 methodology or the assumptions of the new WCAP-17400, S1, R2, methodology for IFBA bearing fuel. By evaluating the current post-Cycle 4 fuel separately from the newly proposed IFBA bearing fuel to be used in the future, separate fuel-specific depletion parameters can be used rather than a single bounding set of depletion parameters, which removes excess conservatism while maintaining appropriate fuel-specific assumptions. The NRC staff finds this acceptable because a single set of burnup and enrichment limits, for each storage configuration, is determined by

[4] Models similar to those used in Appendix A, "Comparison of Burnable Absorber Modeling Assumptions in PINGP Depletion Calculations," were used to verify the appropriateness of [[ ]].

comparing the post-Cycle 4 bounding fuel design modeled in WCAP-17400, Revision 0, to the post-Cycle 4 bounding fuel design modeled in WCAP-17400, S1, R2 and using the bounding of the two limits.

To determine whether the WCAP-17400, Revision 0, or the WCAP-17400, S1, R2, methodology produces bounding burnup and enrichment limits, the WCAP-17400, Revision 0, bias and uncertainty analysis was updated to be consistent with the new WCAP-17400, S1, R2, bias and uncertainty analysis, which included new accounting practices and a new bias based on NRC staff RAIs. The impact of the WCAP-17400, Revision 0, bias and uncertainty reanalysis resulted in slightly reduced burnup and enrichment limits in some cases as reflected in WCAP-17400, R1, R2, Section S6.2, "Comparison of Calculated Burnup Requirements and Final Combined Burnup Requirements." The methodology updates to the bias and uncertainty analysis is discussed in more detail in the corresponding section of this SE.

WCAP-17400, R1, R2, Table S6-18, "Comparison of Input Core Operational Parameters During Depletion," summarizes the differences between the previously approved SFP CSA and the new one with IFBA added to the fuel. The values analyzed limit the applicability of the depletion analysis performed for the PINGP SFP CSA. Any change in the depletion parameters outside the bounds of Table S6-18 and any change in the depletion parameters outside the bounds of those described in applicable RAI responses are outside the scope of approval of this SE.

Burnable Absorber Usage

The net effect of modeling Gd in the SFP versus not modeling it was previously demonstrated to reduce reactivity in Enclosure 1 "Spent Fuel Criticality Analysis, Response to Requests for Additional Information (RAI)" to a letter dated May 16, 2012 (ADAMS Accession No. ML12139A198), as well as in Enclosure 3 to a letter dated September 4, 2012 (ADAMS Accession No. ML12249A069). The May 16 and September 4, 2012, letters were provided by the licensee in response to NRC staff RAIs.

It was shown that [[

]], the modeling practices of (1) and (2) were found to be a conservative practice. The NRC staff, therefore, finds these modeling practices acceptable for PINGP. See confirmatory analysis in Appendix A, which supports this conclusion.

The IFBA material specifications are given in WCAP-17400, S1, R2, Table S3-4, "IFBA Specifications," where both actual planned values and values used in the analysis are given. The analyzed values for the IFBA B-10 loading, thickness, and length either equal actual planned values or exceed them, which is appropriate because either the correct design value or a conservative design value was assumed as an analysis input. WCAP-17400, S1, R2, Table S3-2, "Design Basis Fuel Assembly Design Specifications," provides the maximum number of IFBA rods used in the analysis. The values analyzed limit the applicability of the depletion analysis performed for the PINGP SFP CSA, and any change in plant operation in excess of the

values analyzed, after increasing design values by the amount necessary to bound manufacturing tolerances, are outside the scope of approval of this SE.

Previous studies performed by the licensee and referenced in the NRC staff's August 29, 2013, SE regarding SFP criticality changes related to PINGP have shown that it is bounding to not model the Gd burnable absorber when performing the depletion analysis as part of the PINGP SFP CSA. The licensee now proposes the addition of IFBA, with boron-10 (B-10) as the neutron absorbing material, into its modern (i.e., post-Cycle 4) Gd-bearing fuel assemblies. The NRC staff finds that this decision will not invalidate the no-Gd assumption because the licensee appropriately models the IFBA in both the new depletion analysis and the SFP CSA.

Soluble Boron Modeling

The [[ ]] soluble boron concentration increased [[

]] The NRC staff finds the changes acceptable because the changes are based on representative design data incorporating the effects of the proposed fuel design.

Fuel and Moderator Temperature

The methodology used for fuel temperature modeling during depletion is unchanged from that approved by the NRC staff in WCAP-17400, Revision 0. However, the NRC staff noted that [[

]]. In the RAI 3 response, the licensee confirmed that fuel temperature effects based on the new fuel design have been appropriately incorporated into the depletion analysis. The NRC staff also noted that the [[

]] is unchanged. Since the introduction of a new fuel design could affect [[ ]], the NRC staff asked the licensee why the [[

]] was unchanged in WCAP-17400, S1, R1, in RAI 4. In the RAI 4 response, the licensee explained how use of [[

]][5] Since the previously approved fuel temperature modeling methodology was used with appropriate inputs, the NRC staff finds the fuel temperature modeling to be acceptable.

The methodology used for moderator temperature modeling during depletion is unchanged from that approved by the NRC staff in WCAP-17400, Revision 0. However, the moderator temperatures assumed in the new analysis have changed based on only considering moderator temperature profiles related to the new proposed IFBA bearing fuel design. Specifically, the moderator temperature profiles considered in this analysis are based on the design transition cycles to IFBA bearing fuel. The staff finds the changes acceptable because the changes are based on representative design data incorporating the effects of the proposed fuel design.

[5]The current core design rated thermal power is 1683 MWt with a maximum instantaneous peak assembly average relative power of 1.576.

Assembly Power and Operating History Effects

The NRC staff has confirmed that this area of the methodology remains unchanged from WCAP-17400, Revision 0. Regarding assembly power modeling, there is no increase to the design basis maximum assembly power which is the relevant input to the previously approved depletion analysis methodology.

Regarding operating history effects, these were appropriately treated independent of the fuel design in the previously approved methodology; therefore, changing the fuel design has no impact on how operating history effects are accounted for in the new CSA.

Based on the above, the NRC staff finds that maintaining the previous modeling of assembly power and accounting of operating history effects is acceptable.

Rodded Operation

The NRC staff confirmed that this area of the methodology remains unchanged from WCAP-17400, Revision 0. Similar to the impact to operating history effects, changing the fuel design has no impact on how rodded operation is accounted for in the new CSA; therefore, the staff finds that maintaining the previous treatment for rodded operation, explained in detail in the NRC staff's WCAP-17400, Revision 0, SE is acceptable.

Axial Burnup Profiles

The methodology used for axial burnup profile selection is unchanged from that approved by the NRC staff in WCAP-17400, Revision 0. WCAP-1700, S1, R2, uses a [[

]] The NRC staff finds that analyzing profiles from [[

]] is adequate to obtain [[

]] profiles that are representative of future reactor operation.

<u>Depletion Code Validation</u>

The methodology used for depletion code validation is unchanged from that approved by the NRC staff in WCAP-17400, Revision 0. The resultant depletion code uncertainty for the various configurations has changed. This is due to the change in depleted fuel isotopics associated with the new fuel design and its operation.

The Kopp Letter, which is referenced by DSS-ISG-2010-01, states the following:

> A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

This 5 percent depletion worth uncertainty allowance, referred to as the Kopp 5 percent, has generally been found to be acceptable when licensees use nuclide concentrations from depletion calculations with depletion codes that have been previously approved by the NRC for licensing applications (e.g., as part of the nuclear design evaluation). If a licensee were to use a depletion code that had not been previously approved by the NRC for licensing applications, additional verification would be necessary to justify the application of the Kopp 5 percent. Since PARAGON – the lattice code used – has been previously approved for use in performing depletion calculations supporting the PINGP SFP CSA, the NRC staff finds the use of PARAGON to perform depletion calculations supporting the SFP CSA and the application of the Kopp 5 percent to be acceptable.

More recent studies sponsored by NEI also suggests that the Kopp 5 percent is an acceptable engineering judgment in lieu of formal depletion code validation when using the Electric Power Research Institute (EPRI) depletion code validation methodology (ADAMS Accession No. ML17013A132).[6]

Recent work by Westinghouse using EPRI's depletion code validation methodology applied to the depletion and criticality codes used in WCAP-17400, S1, R2, shows that, for the case with the most similar fuel to the design basis fuel to be stored at PINGP (i.e., Case 6, "104 IFBA depletion"), a conservative burnup-dependent bias ranging between 165-1310 percent millirho (pcm) was determined indicating that no additional depletion code bias correction is needed (see "EPRI Depletion Benchmark Calculations Using PARAGON," Annals of Nuclear Energy 81: 1–5. http://www.sciencedirect.com/science/article/pii/S0306454915001437).[7] This study also provides the NRC staff with additional assurance that the representation of depleted fuel in the PINGP SFP storage environment is conservative.

As a result of the above discussion, the NRC staff finds the continued use of the Kopp 5 percent in the PINGP SFP CSA to be acceptable.

Modeling of Fuel Assemblies Containing both IFBA and Gadolinium Absorber Rods with Westinghouse Core Design Code Systems

The NRC staff has reviewed Westinghouse's high level validation studies supporting use of Westinghouse lattice and nodal core simulator codes as part of the PINGP SFP CSA methodology.[8] Specifically, the lattice code is used to produce spent fuel isotopics used in criticality calculations, and the nodal core simulator is used to produce the population of plant-specific axial burnup profiles. The plant-specific axial burnup profiles are analyzed further to develop a set of bounding plant-specific axial burnup profiles used by the lattice code to produce spent fuel isotopics used in criticality calculations.

Proprietary Enclosure 6 to the LAR discusses validation studies performed indicating that the Westinghouse core design codes used in the PINGP SFP CSA can adequately model fuel assemblies containing both IFBA and Gd. The core design codes consist of a lattice code and a nodal core simulator code. Westinghouse explains that the new fuel assembly design with both

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[6] This methodology is currently under NRC review. Approval is pending based on submittal of final report revisions incorporating methodology revisions discussed in RAI responses.

[7] The specific depletion and criticality codes used in the WCAP-17400 S1, R2, SFP CSA are PARAGON with a ENDF/B-VI.3-based 70-group cross-section library for depletion calculations and SCALE 5.1 with a 44-group ENDF/B-V cross-section library.

[8] Provided in Enclosure 6 to the November 17, 2015 LAR.

IFBA and Gd in an assembly will be predicted as accurately by the nodal core simulator as for current cores with either IFBA or Gd rods in separate assemblies. Westinghouse explains that this primarily depends on the accuracy of cross-section data generated by the lattice code as input into the nodal core simulator code.

Validation of the lattice code was provided by:

1. Comparison to results from MCNP[9] (LANL 2003) which implements higher order methods (e.g., cross-section and geometry representation):

    a. Assumptions made in the [[ ]] were validated by comparing pin powers from assemblies containing IFBA and Gd separately

    b. Accuracy of the neutron flux solver was confirmed by comparing pin powers and reactivity from the specific fuel assembly designs planned for use at PINGP that contain IFBA and Gd together

2. Comparison to relevant[10] critical experiments which were part of the NRC-approved lattice code topical report

Validation of both the lattice code and nodal core simulator code as a code system was provided by comparison to plant data; here, pin powers are compared for assemblies containing IFBA and Gd separately. Westinghouse explains that:

> The mixture of IFBA and Gd will not introduce any challenging methodology in modeling the presence of these two absorbers in the same assembly since the tracking and depletion of the boron and gadolinium isotopes will be the same. If the cross-section data generated by the lattice code is accurate, the ANC prediction will be as accurate as for current cores with either IFBA or gadolinium rods in separate assemblies.

Westinghouse concluded that the results of the validation studies:

> Show that the fuel assemblies with the mixture of IFBA and gadolinium burnable absorber rods behave like the assemblies without burnable absorbers or with a single type of them (either IFBA or Gd). Therefore, the new assemblies are expected to be predicted by [Westinghouse lattice and nodal core simulator codes] with the same accuracy as the ones that are currently in use in operating plants.

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[9] MCNP is a general-purpose Monte Carlo code developed by the Department of Energy's Los Alamos National Laboratory that can be used for, among other things, neutron transport calculations. The code and its accompanying nuclear data sets have been extensively validated by LANL for various neutron transport calculations.

[10] As stated in Enclosure 6, "the relevant B&W [(Babcock/Wilcox)] cases are cores 15 and 17 described in Table 3-6 of Reference 2, all these experiments have both Gd rods (with 4 w/o [Gd] and 1.94 w/o U-235 enrichments) and B_4C control rods in their configurations."

Westinghouse substantiated this conclusion by:

1. Providing the maximum calculated reactivity difference between the lattice code and MCNP,

2. Confirming that there was no change in accuracy as a function of burnup, and

3. Providing the maximum calculated difference in pin power distribution between the lattice code and MCNP.

Consequently, the NRC staff finds that the use of Westinghouse lattice and nodal core simulator codes used as part of the PINGP SFP CSA methodology is acceptable based on:

1. Westinghouse highlighted validation studies performed in the NRC-approved lattice code topical report,

2. Excellent agreement shown between the lattice code and MCNP predicted pin powers providing confidence in the calculation of neutron flux for the new PINGP-specific fuel assembly design, and

3. Strong indication that the Westinghouse lattice and nodal code simulator codes will predict assembly powers for the proposed PINGP fuel and core design containing both IFBA and Gd within a fuel assembly as accurately as with past fuel and core designs that have had exclusively either IFBA or Gd within a fuel assembly.

The NRC staff approval of the Westinghouse lattice and nodal core simulator code for use in modeling assemblies containing adjacent IFBA- and Gd-bearing fuel rods, is specific to the PINGP SFP CSA methodology, and does not constitute generic approval. Furthermore, the staff approval of these codes applies only to the SFP CSA methodology.

Criticality Analysis

Several areas of the criticality analysis were updated to account for the effects of IFBA bearing fuel in WCAP-17400, S1, R2. One modeling assumption that affects several calculations includes the B-10 atom fraction change from 0.199 to 0.194 in calculations involving soluble boron. Another is [[ ]] as stated in Section S3.7, "KENO Modeling Assumptions." The first assumption is acceptable because it removes the need for an approximation by explicitly assuming the correct B-10 atom fraction as an analysis input. The second assumption is acceptable because it reduces neutron absorption in the SFP CSA for instances where it would actually be higher, which is conservative.

The following subsections describe the various aspects of the revised SFP CSA methodology.

Criticality Code Validation

The NRC staff confirmed that this area of the methodology remains unchanged from WCAP-17400, Revision 0. This area of the methodology is unchanged since no new materials were introduced and no fuel or rack design changes were made to the criticality models. Consequently, the criticality code validation results remain valid.

Normal Conditions

The NRC staff confirmed that this area of the methodology remains unchanged from WCAP-17400, Revision 0; however, the design basis allowable storage arrays were reanalyzed to incorporate the effects of IFBA bearing fuel. Fuel Category 1, fresh fuel up to 5 wt% U-235, is unaffected by the introduction of IFBA bearing fuel because [[
]] as stated in WCAP-17400, S1, R2, Section S3.7.

The only normal conditions affected by introduction of IFBA bearing fuel, outside of the calculational updates to the design basis allowable storage arrays described above, are specific cases described in WCAP-17400, S1, R2, Section S4.4.3, "Type 3 Normal Conditions," covering the limitations associated with storage of a consolidated fuel rod storage canister, damage or compacted fuel structural components, failed fuel baskets, movable in-core detectors, and filter assemblies. Specifically, the IFBA bearing fuel impacts the storage of components falling under Fuel Category 7 which are to be stored in Array F, which are only allowed to interface with Array A. Although Array F was not reanalyzed with IFBA bearing fuel, Array A was reanalyzed with IFBA bearing fuel, therefore, the interface must be checked consistent with the previously approved interface verification methodology described in WCAP-17400, Revision 0.

The Array A and Array F interface condition was reanalyzed by [[
]] This interface verification methodology is unchanged from what the NRC staff previously approved in WCAP-17400, Revision 0, and the regulatory k-effective limit is not exceeded, therefore, storage of Array A containing IFBA bearing fuel adjacent to Array F without IFBA bearing fuel is acceptable.

The other portion of the WCAP-17400, Revision 0, Type 3, normal condition methodology that has the potential to be affected is [[ ]] methodology. However, since this methodology [[
]], the analysis conclusions remain unchanged and are therefore acceptable.

Accident Conditions

The NRC staff has confirmed that this area of the methodology remains unchanged from WCAP-17400, Revision 0, except for the addition of a new and limiting accident analysis – that is, a multiple assembly misload into the storage racks as described in WCAP-17400, S1, R2, Section S4.5.1.2. The analysis was reviewed and the [[ ]] was found to be the bounding misload accident. [[

]]

In LAR, Section 2.4, the licensee states that analysis of the multiple misload analysis extends beyond the double contingency principle (DCP). While the NRC staff agrees that application of

the DCP cannot screen out consideration of the multiple misload event, the staff disagrees with the characterization that evaluation of the multiple misload event is inherently conservative. Since the likelihood of the event has not been quantified in the PINGP SFP CSA evaluation and has not been shown to be a highly unlikely event, analysis of the event cannot be credited as a conservatism. If the event can be demonstrated to be highly unlikely, then analysis of the event would be justifiably conservative.

Since the licensee has identified and analyzed the most limiting credible accident, which defines the minimum soluble boron requirement identified in TS 3.7.16, the NRC staff finds the accident conditions analysis to be acceptable.

Bias and Uncertainty Analysis

Determination of Total Bias and Uncertainty

Most aspects of the bias and uncertainty analysis methodology remain unchanged in WCAP-17400, S1, R2. This subsection reviews those aspects that have changed in detail relative to WCAP-17400, Revision 0.

In WCAP-17400, Revision 0, the lack of validation for minor actinides and fission products was treated as an uncertainty. In WCAP-17400, R1, R2, this uncertainty was recharacterized as a [[ ]] bias component based on the most recent information available concerning minor actinide and fission product validation using NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality (k-effective) Predictions" (ADAMS Accession No. ML12116A124). Based on the approach described in WCAP-17400, S1, R2, the NRC staff finds that the bias determination has been applied consistent with the guidance in NUREG/CR-7109, [[ ]]. Furthermore, since the latest best-practice is being used to treat the lack of validation for minor actinides and fission products, which is more conservative than what was previously assumed in the PINGP SFP CSA, the staff finds this acceptable.

The NRC staff confirmed that the manufacturing tolerance uncertainties described in WCAP-17400, S1, R2, Section 4.1.2.1.1, "Manufacturing Tolerance," remains unchanged from WCAP-17400, Revision 0, except for [[ ]].

The [[ ]] are related to RAIs 6 and 7, respectively, as documented in an October 31, 2016, e-mail to the licensee. These issues were identified based on guidance in NEI 12-16. Although the final revision to NEI 12-16 is currently in the process of being endorsed (ADAMS Accession No. ML17192A220), the NRC staff had found the guidance related to the concerns identified in RAIs 6 and 7 acceptable without exception.

RAI 6 is related to accounting for the reactivity effect of eccentric positioning [[ ]] in SFP CSAs. The NRC staff does not view this as an uncertainty because the positioning of each assembly is generally uncontrolled – that is, assembly placement into a SFP storage cell by fuel handling operators cannot guarantee assembly placement into a fixed location (e.g., perfectly centered in the cell) for every assembly placement. Therefore, to assume a centrally located position in the CSA, one would have to have assurance that an operator could not inadvertently bias assembly placement within a SFP

storage cell. Furthermore, a relatively small number of assemblies as compared to the full inventory (9 or 16 out of a thousand), would need to inadvertently end up in the worst-case configuration to achieve close to the maximum k-effective increase. Due to the relatively large number of assemblies available for randomly achieving the worst-case, the NRC staff considered that it is more than likely that the worst-case configuration, or one close to it, could exist, assuming that the assemblies are placed randomly. In RAI 6, the NRC staff requested that the licensee justify the accounting of the fuel assembly eccentric positioning reactivity effect in light of the current NRC and industry understanding of this phenomenon to ensure that the 10 CFR 50.68(b)(4) requirements are met.

By letter dated October 4, 2017, the licensee provided a response to the NRC staff's October 31, 2016, RAI in Enclosure 1 (proprietary) and Enclosure 2 (nonproprietary), and updated aspects of the PINGP SFP CSA to credit more realistic assumptions specific to the new IFBA bearing fuel design.

In the response to RAI 6, the licensee discusses accounting for the reactivity impact of eccentric positioning as a bias rather than an uncertainty. For [[
]], the eccentric positioning model consisted of [[

]]. Based on the overall consistency with the NEI 12-16 guidance on fuel assembly eccentric positioning modeling and the low likelihood of the modeled worst-case eccentric positioning arrangement, as described by the licensee, the NRC staff finds the fuel assembly eccentric positioning bias analysis to be acceptable.

RAI 7 is related to the potential for fuel assembly spacer grid growth during irradiation and its impact on SFP CSAs because spacer grids have been shown to expand over the course of their utilization in the reactor.[11]

The NRC staff sensitivity studies showed that the effect of uniform pitch changes of 0.5 percent and 1 percent for some fuel cladding materials under SFP storage conditions can result in relatively large reactivity effects of approximately 500 pcm and 1000 pcm, respectively. Consequently, staff RAI 7 requested that the licensee explain how fuel assembly spacer grid growth affects the PINGP SFP CSA documented in WCAP-17400, S1, R1, and the ability to meet 10 CFR 50.68(b)(4) requirements.

In its response to RAI 7, the licensee discusses the inclusion of a burnup dependent bias to incorporate the impact of fuel assembly spacer grid growth on the PINGP SFP CSA. The response to RAI 7 states that a burnup dependent bias has been developed in order to account for the impact of grid growth in the PINGP SFP CSA using [[

[11]See Figure 4 of "*An Investigation on Irradiation-Induced Grid Width Growth in Advanced Fuels,*" edited by the Korean Nuclear Society (KNS) from proceedings of the KNS Autumn meeting.

]]. The analysis conservatively assumes that [[

]]. Based on its review, the NRC staff finds that the licensee realistically accounted for the depletion effects of grid growth using data applicable to PINGP, therefore, the modeling of grid growth during depletion is acceptable.

To account for the reactivity effects of grid growth in the PINGP SFP CSA, [[ ]] for Storage Arrays A, B, D, and G. Equation 1 in the RAI 7 response was [[

]]. The NRC staff finds it acceptable to use the [[ ]] to conservatively quantify the grid growth bias estimate for Array A and also apply it to Arrays B and G since Array B is of lower storage density than Array A, and Array G is of the same storage density as Array A, but with lower grid growth reactivity sensitivity due to the presence of a required control rod within the array.

Equation 2 in the RAI 7 response was developed based on [[

]]. The NRC staff finds the Array D grid growth bias acceptable because the [[ ]] was used to conservatively quantify grid growth effects specific to Array D.

The grid growth bias for Array E [[

]]. In this case, the analysis [[

]]. The NRC staff finds this acceptable because the [[ ]] was used to conservatively quantify grid growth effects [[ ]], and the [[ ]] was used to conservatively quantify grid growth effects [[

]].

Due to the addition of the grid growth bias, the analysis [[

]]. The NRC staff finds the [[ ]] to be acceptable because it has been conservatively quantified [[ ]] and applied appropriately in the PINGP SFP CSA. For [[ ]] evaluations, the [[

]]. The NRC staff finds the [[ ]] to be acceptable for [[

]].

Based on the above discussion related to RAI 7, the NRC staff finds that the licensee has (1) appropriately and conservatively accounted for (1) the effects of fuel assembly spacer grid growth and (2) the fuel rod pitch tolerance consistent with the guidance provided in NEI 12-16.

The PINGP SFP CSA updates described in the responses to RAIs 6 and 7 resulted in an increase in reactivity throughout the SFP CSA. Consequently, to offset these increases, and maintain comparable burnup and enrichment limits in the proposed TS, the PINGP SFP CSA methodology was modified to remove excess conservatism as previously discussed in the sub-section above titled, "Selection of Bounding Fuel Assembly and Depletion Conditions." Specifically, conservatism was removed by: (1) using moderator temperatures and axial burnup profiles specific to the new IFBA bearing fuel design rather than also including profiles that were used in the WCAP-17400, Revision 0, SFP CSA and (2) changing the fuel assembly axial blanket region modeling for the new IFBA bearing fuel design.

Burnup Limits for Storage Arrays

In WCAP-17400, Section S3.7.1, "Array Descriptions," the licensee proposes six normal storage configurations for use throughout the PINGP SFPs defined by combinations of one or two fuel categories. Lower number fuel categories correspond to more reactive configurations and higher number fuel categories correspond to less reactive configurations. Since no burnup is required for Category 1 fuel, no burnup limits are presented for corresponding storage array configurations. WCAP-17400 S1, R2, also states that Category 7 fuel cannot be used to store IFBA bearing fuel at PINGP since the necessary analysis was not performed.

The burnup and enrichment limits defined by the burnup and enrichment loading curves (BULCs) used by PINGP are provided, with example requirements at the analyzed initial fuel enrichments and as a function of decay time, in WCAP-17400, S1, R2, Section S5.1, and are the same BULCs implemented in the revised TS, Table 4.3.1-3. The notes with revised TS, Figure 4.3.1, that provide specific guidelines for SFP fuel assembly storage were reviewed by the NRC staff. The NRC staff noted a lack of specificity in revised TS, Figure 4.3.1, since TS, Table 4.3.1-3, will now apply to both older non-IFBA fuel and new IFBA-bearing fuel. Since the WCAP-17400, S1, R2, criticality analysis did not analyze Category 7 fuel with IFBA-bearing fuel, as noted above, proposed TS, Figure 4.3.1, notes were updated to limit fuel Category 7 to "fuel that was consolidated in the spent fuel consolidation demonstration project described in updated safety analysis report (USAR), Section 10.2," as explained in the RAI 2 response provided by the licensee in a letter dated May 23, 2016. Consequently, the NRC staff finds the updates to TS 4.3.1 to be appropriate and consistent with the PINGP SFP CSA reviewed in WCAP-17400, S1, R2, and are, therefore, acceptable.

WCAP-17400, S1, R1, Section S5.1, "Burnup Limits & Restrictions on Storage Arrays," contains tables with various decay-time dependent fitting coefficients to be used with an equation relating the initial fuel enrichment to the minimum burnup for fuel assembly loading into the various

storage arrays – i.e., these tables define, by curve fit, the various BULCs. The NRC staff performed a confirmatory analysis, based on the response to RAI 1 provided by the licensee in a letter dated May 23, 2016, to confirm that all BULCs either pass through the explicit burnup/enrichment points or exceed them based on a curve defined by linear interpolation between points. The following table shows item (1) the maximum difference in burnup (with units of gigawatt-days per metric ton uranium -- GWd/MTU) found between the Section S5.1 curve fits and the actual burnup/enrichment points analyzed in WCAP-17400, S1, R1, which were used to form a piecewise polynomial spline function of degree 1, and item (2) the fraction of points along each curve that were found to be nonconservative. The first column ('cat') corresponds to the fuel category defined and analyzed in WCAP-17400, S1, R1, the second column ('maxDiffs') of the table corresponds to item (1) described in the previous sentence, and the third column ('fracNonCon') corresponds to item (2) described in the previous sentence.

cat	maxDiffs	fracNonCon
2	-0.0001705134	0.000
3	0.0046096733	0.031
4	0.0043327724	0.000
5	0.0012372688	0.011
6	0.0275693894	0.074

The 'fracNonCon' column indicates that a significant portion of the BULCS defined by fitting coefficients are non-conservative (e.g., for approximately 7 percent of the burnup range of Category 6 fuel) because the curve fit produces a lower burnup than required for storage. However, the maximum difference is approximately 0.03 GWd/MTU, which is worth on the order of 10 pcm -- an insignificant non-conservatism because the magnitude of the reactivity effect is so small that it is considered to be negligible.

Because the BULC generation method was demonstrated to be appropriate with respect to all calculated burnup and enrichment limits documented in WCAP-17400, S1, R1, as confirmed by the NRC staff's confirmatory analysis, the NRC staff finds that the BULCs provided in WCAP-17400, S1, R2, generated using the same method, are acceptable.

Based on its review detailed above, the NRC staff confirmed that it is acceptable to use a single set of conservative BULCs which bounds both IFBA bearing fuel and fuel not operated in Cycles 1-4 from WCAP-17400, Revision 0, which did not consider the effects of IFBA.

3.2 NRC Staff Evaluation Conclusion

The NRC staff finds the SFP CSA described in WCAP-17400, S1, R2, supporting fuel assembly storage in the PINGP SFP to be acceptable. Furthermore, WCAP-17400, S1, R2, is consistent with PINGP's revised TS 3.7.16 and TS 4.3.1; therefore, the proposed TS revisions are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments on November 9, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding June 6, 2017 (82 FR 26133). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

Principal Contributor: Amrit Patel, NRR

Date of issuance: November 30, 2017

Appendix A

APPENDIX A

COMPARISON OF BURNABLE ABSORBER MODELING ASSUMPTIONS IN PINGP DEPLETION CALCULATIONS

A.1 OVERVIEW

WCAP-17400, Revision 0, assumes no gadolium (Gd) present in fuel assemblies during depletion; this assumption was found to be bounding in terms of maximizing spent fuel reactivity compared to modeling the Gd that is actually present in these fuel assemblies.

In WCAP-17400, S1, R2, integral fuel burnable absorber (IFBA) was added as a fuel assembly burnable absorber (BA), in separate fuel pins, in addition to Gd. The U.S. Nuclear Regulatory Commission (NRC) performed three separate fuel assembly depletion calculations to confirm the licensee's claim that only modeling IFBA, and continuing to not model Gd, is a conservative practice for the proposed Prairie Island Nuclear Generating Plant (PINGP) fuel design. The three calculations assume:

1. No Gd and no IFBA (WCAP-17400, Revision 0, conservative analysis assumption approved in 2013 license amendment request

2. Gd and IFBA (Similar fuel design to that proposed in WCAP-17400, S1, R2)

3. Only IFBA (WCAP-17400, S1, R2, analysis assumption)

Fuel assembly average burnup up to 50 megawatt days per metric ton of uranium (GWd/MTU) was considered to be representative of the minimum allowable assembly average burnup (BU) for Category 6 fuel, which contains the highest burnup fuel of all categories. The confirmatory analysis uses the T-DEPL sequence in SCALE 6.2b2 with the standard ENDF/B-VII 238 fine neutron energy group library to perform two-dimensional (2D) material average flux depletion calculations (versus pin-by-pin depletion) to understand the relative k-effective sensitivity versus burnup for the three assumptions.

A.2 ANALYSIS ASSUMPTIONS

A.2.1 Fuel Assembly Model

- Fuel assembly characteristics specified in WCAP-17400, S1, R1 and in confirmatory analyses:
  - 14x14 422V+ fuel assembly design; 16 guide tubes (GTs) and 1 instrument tube (IT)
  - 5 weight percent enriched uranium oxide (UO_2)
  - [[ ]] of the UO_2 theoretical density (TD)

- Fuel assembly dimensions specified in WCAP-17400, S1, R2:
  - 0.3659 inch (in) fuel pellet diameter
  - 0.422 in clad OD (outside diameter)
  - 0.3734 in clad ID (inside diameter)
  - 0.422 in IT OD

  - 0.3740 in IT ID
  - 0.5260 in GT OD
  - 0.4920 in GT ID
  - 0.556 in fuel rod pitch

- Fuel assembly dimensions used in confirmatory analyses:
  - SCALE 6.2b2 Westinghouse 14x14 depletion calculation template values
  - Only IT and GT dimensions different than WCAP-17400, S1, R2, values; not expected to affect confirmatory analysis conclusions

A.2.2 Burnable Absorber Modeling

- [[ ]] IFBA fuel rods; [[ ]]
  - Concentration: [[ ]] mg/in; [[ ]]
  - Thickness: [[ ]] mils; [[ ]]
  - Length: [[ ]] in; not considered in 2D depletion calculation
- [[ ]] Gd fuel rods at a concentration of [[ ]] Gd; [[ ]] Gd rods modeled; 1 Gd ring
- Burnable absorber map assumed since not provided by Westinghouse:
  - 1 represents fuel rod without BA
  - 2 represents GT
  - 3 represents IT
  - 4 represents fuel rod with only IFBA as BA
  - 5 represents fuel rod with only Gd as BA

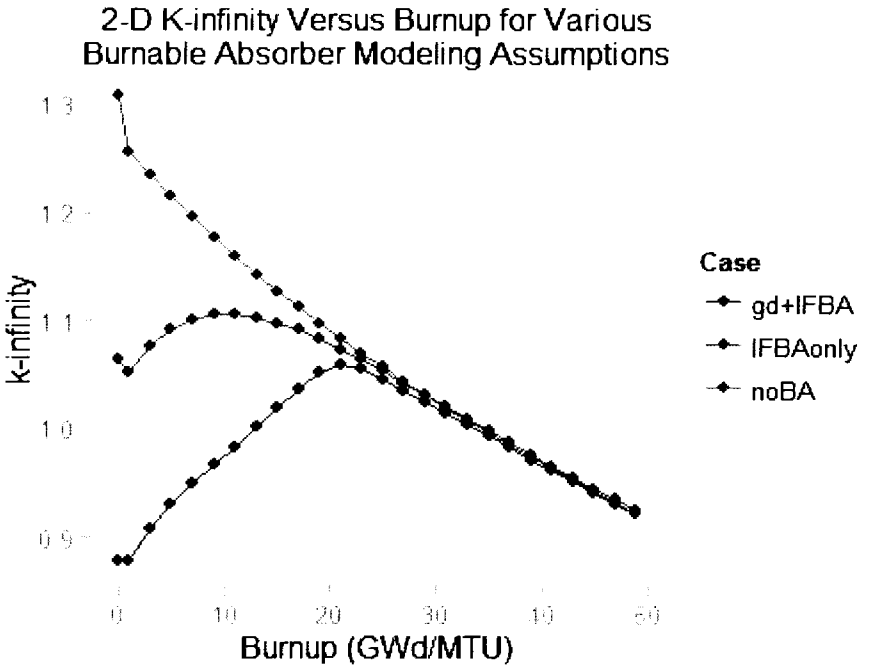
[[

]]

A.3 Results and Conclusion

Comparison of k-effective versus burnup for the three cases confirms the WCAP-17400, S1, R2, conclusion that continuing to not model Gd, while modeling IFBA, will result in a conservative set of spent fuel isotopics to be used in the PINGP SFP CSA. In the figure below, it is seen that modeling no burnable absorber will result in a conservative set of isotopics up to a burnup of approximately 30 GWd/MTU. After this burnup, it is more conservative to model only IFBA with no Gd. However, either of these cases bounds the actual case where both IFBA and Gd are present for all burnup. Consequently, the IFBA only burnable absorber assumption is acceptable to use for the proposed PINGP fuel design which now includes IFBA.

While it is likely that there will be other permutations of Gd and IFBA usage than the one studied here, the NRC staff has reasonable assurance that the burnable absorber modeling assumptions specific to PINGP will remain conservative regardless of the IFBA and Gd loadings used within the range of burnable absorber parameter values specified in WCAP-17400, S1, R2, and Enclosure 6 to the 2015 LAR. Consequently, the staff finds the burnable absorber depletion calculation modeling assumptions affecting the PINGP SFP criticality calculations that credit burnup to be acceptable.


2-D K-infinity Versus Burnup for Various
Burnable Absorber Modeling Assumptions
k-infinity
1.3
1.2
1.1
1.0
0.9
0
10
20
30
40
50
Burnup (GWd/MTU)
Case
gd+IFBA
IFBAonly
noBA

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT REQUEST RELATED TO SPENT FUEL POOL CRITICALITY TECHNICAL SPECIFICATION CHANGES (CAC NOS. MF7121 AND MF7122, EPID L-2015-LLA-0002) DATED NOVEMBER 30, 2017

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