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

August 29, 2013

James E. Lynch
Site Vice President
Northern States Power Company - Minnesota
Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089-9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SPENT FUEL POOL CRITICALITY CHANGES (TAC NOS. ME6984 AND ME6985)

Dear Mr. Lynch:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed 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 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 August 19, 2011, supplemented by letters dated May 16, 2012, September 4, 2012, February 8, 2013, and July 17, 2013.

The amendments revise TS 3.7.17, "Spent Fuel Pool Storage," and TS 4.3.1, "Fuel Storage Criticality" to provide new spent fuel pool (SFP) loading restrictions that meet subcriticality for all postulated conditions. The TS changes correct non-conservatisms in the SFP criticality analysis-of-record. The amendments also change the evaluation methodology used for the SFP criticality analysis.

NOTICE: Enclosure 4 to this letter contains Proprietary Information. Upon separation from Enclosure 4, this letter is DECONTROLLED

The NRC has determined that the related Safety Evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a redacted publicly available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 209 to DPR-42
2. Amendment No. 196 to DPR-60
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

cc w/encls 1, 2, and 3: Distribution via 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. 209
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 August 19, 2011, as supplemented by letters dated May 16, 2012, September 4, 2012, February 8, 2013, and July 17, 2013, 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 hereby 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. 209, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

This license amendment is effective as of the date of its issuance and shall be implemented within 120 days. In conjunction with implementation of the amendment, procedures will be revised to require an assessment of a fuel assembly's exposure to rodded power operation in the core prior to moving that fuel assembly into the spent fuel pool (SFP) storage racks. If an assembly experiences more than 100 megawatt day per metric ton uranium (MWd/MTU) of core average full-power rodded operation exposure, this exposure experienced while rodded will not be credited for determining the coefficients used to categorize fuel assemblies as described in WCAP-17400-P.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 29, 2013

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. 196
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 August 19, 2011, as supplemented by letters dated May 16, 2012, September 4, 2012, February 8, 2013, and July 17, 2013, 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 hereby 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. 196, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

This license amendment is effective as of the date of its issuance and shall be implemented within 120 days. In conjunction with implementation of the amendment, procedures will be revised to require an assessment of a fuel assembly's exposure to rodded power operation in the core prior to moving that fuel assembly into the spent fuel pool (SFP) storage racks. If an assembly experiences more than 100 megawatt day per metric ton uranium (MWd/MTU) of core average full-power rodded operation exposure, this exposure experienced while rodded will not be credited for determining the coefficients used to categorize fuel assemblies as described in WCAP-17400-P.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 29, 2013

ATTACHMENT TO LICENSE AMENDMENT NOS. 209 AND 196

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 revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
DPR-42, License Page 3	DPR-42, License Page 3
DPR-60, License Page 3	DPR-60, License 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
3.7.17-1	3.7.17-1
3.7.17-2	3.7.17-2
3.7.17-3	---
4.0-2	4.0-2
4.0-3	4.0-3
4.0-5	4.0-5
4.0-6	4.0-6
4.0-7	4.0-7
4.0-8	4.0-8
---	4.0-9
---	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. 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) 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. 196, 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

3.7.17 Spent Fuel Pool Storage

LCO 3.7.17 Each fuel assembly, fuel insert, or hardware stored in the spent fuel pool shall satisfy the loading restrictions of Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	Initiate action to move the noncomplying fuel assembly to an acceptable location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the fuel assembly, fuel insert, or other hardware placed in the spent fuel storage racks is stored in accordance with Specification 4.3.1.1.	Prior to storing or moving the fuel assembly, fuel insert, or other hardware
SR 3.7.17.2	Verify spent fuel pool inventory.	Within 7 days after completion of a spent fuel pool fuel handling campaign

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in USAR Section 10.2;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 400 ppm, which includes an allowance for uncertainties as described in USAR Section 10.2;

d. A nominal 9.5 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies, fuel inserts, and hardware loaded in accordance with Figure 4.3.1-1.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in USAR Section 10.2;

c. $k_{eff} \leq 0.98$ if accidentally filled with a low density moderator which resulted in optimum low density moderation conditions; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 727′ 4″ (Mean Sea Level).

Table 4.3.1-1 (page 1 of 1)
Fuel Categories Ranked by Reactivity

FUEL CATEGORY	RELATIVE REACTIVITY
1	High
2	↓
3	
4	
5	
6	Low
7	Consolidated Fuel

Notes:

1. Fuel category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category 2 is less reactive than Category 1, etc. The more reactive fuel categories require additional measures to be placed on fuel placement in the Spent Fuel Pool (SFP) racks, e.g., more use of water-filled cells or Rod Control Cluster Assemblies (RCCAs).

2. Any higher-numbered fuel category (except Category 7) may be used in an array specifying a lower-numbered fuel category.

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

4. Category 7 is consolidated fuel stored in Consolidated Rod Storage Canisters.

5. Categories 2 through 6 are determined from Tables 4.3.1-2 and 4.3.1-3.

Table 4.3.1-2 (page 1 of 1)
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)

FUEL CATEGORY	DECAY TIME	COEFFICIENTS			
		A_1	A_2	A_3	A_4
3	0	0.000	-0.722	14.272	-31.167
	20	0.000	-1.944	20.494	-39.085
5	0	0.673	-8.242	44.607	-56.428
	20	1.784	-16.297	60.035	-64.713
6	0	1.097	-10.246	47.457	-56.456
	20	1.820	-15.656	56.856	-60.351

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 3.4 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 operated in the core for any period of time during Unit 1 or Unit 2 Cycles 1 through 4.

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	-0.669	9.018	-32.080	33.507
3	0	-0.120	1.300	5.006	-18.765
	5	-0.167	1.766	3.085	-16.141
	10	-0.218	2.249	1.405	-14.163
	15	-0.281	2.949	-1.267	-10.873
	20	-0.401	4.237	-5.881	-5.513
4	0	1.355	-14.866	62.715	-72.624
5	0	0.569	-6.563	37.088	-47.854
	5	0.302	-3.795	27.410	-37.964
	10	0.151	-2.248	21.874	-32.204
	15	-0.198	1.133	11.031	-21.713
	20	-0.427	3.424	3.614	-14.522
6	0	0.567	-6.205	35.936	-45.944
	5	0.923	-9.720	45.538	-53.858
	10	0.728	-7.992	40.264	-48.929
	15	0.343	-4.016	27.236	-36.380
	20	0.283	-3.391	24.925	-33.963

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.

Any fresh fuel, irradiated fuel, or non-fuel material shall meet the following restrictions prior to placement in the Spent Fuel Pool storage racks when any fuel is in the spent fuel pool:

A. Any array of storage cells containing fuel shall comply with the storage patterns in Figure 4.3.1-1 and the requirements of Tables 4.3.1-1, 4.3.1-2, and 4.3.1-3 as applicable. The category number of fuel assemblies selected for a 2x2 or 3x3 array (category determined using Table 4.3.1-2 or 4.3.1-3) shall be equal to or greater than the category number shown in the respective figure.

B. Any storage array location designated for a fuel assembly may be replaced with a failed fuel basket (fuel rod storage canister or failed fuel pin basket), incore detectors, or other non-fissile hardware.

C. Fuel assembly inserts designed for use in the reactor core may be inserted in a stored assembly (in the Spent Fuel Pool) without affecting the fuel category.

Figure 4.3.1-1 (page 1 of 3)
Spent Fuel Pool Loading Restrictions

DEFINITION	ILLUSTRATION	
Array A Category 6 assembly in every cell.	6	6
	6	6
Array B Category 3 assembly in 3-of-4 cells, with empty cell in the fourth cell.	3	3
	3	X
Array C Checkerboard pattern of diagonally-opposed Category 1 assemblies with empty cells.	1	X
	X	1
Array D Checkerboard pattern of two face-adjacent Category 5 assemblies with an empty cell and Category 1 assembly. Allows for transition from Array C and other arrays.	5	5
	1	X
Array E Checkerboard pattern of two diagonally-opposed Category 2 assemblies with an empty cell and Category 4 assembly.	4	2
	2	X
Array F Checkerboard pattern of diagonally-opposed Category 7 consolidated rod storage canisters and empty cells, which may be filled with assembly nozzles, guide tubes, and grids.	7	X
	X	7
Array G 3-by-3 pattern of Category 5 assemblies with an RCCA loaded in the center assembly.	5	5	5
	5	5R	5
	5	5	5

Figure 4.3.1-1 (page 2 of 3)
Allowable Storage Arrays

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.

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

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 196 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

1.0 INTRODUCTION

By application dated August 19, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112360231), supplemented by letters dated May 16, 2012, September 4, 2012, February 8, 2013, and July 17, 2013 (ADAMS Accession Nos. ML12139A198, ML12249A069, ML13039A306, and ML13199A416, respectively), Northern States Power Company, a Minnesota Corporation (the licensee), doing business as Xcel Energy, requested changes to the Technical Specifications (TSs) for Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

The proposed changes correct non-conservatisms in the PINGP spent fuel pool (SFP) criticality analysis-of-record, which the licensee has addressed in the PINGP Corrective Action Program. The licensee has maintained operability of the SFP through interim administrative controls on SFP loading patterns that are more restrictive than the current TS requirements.

Specifically, the proposed changes would revise 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. The amendments would also revise the evaluation methodology used for the SFP criticality analysis.

The licensee proposed seven different storage configurations and seven different fuel categories. Each fuel category, except for the category for the fresh 5 weight percent ^{235}U fuel has a burnup versus initial enrichment requirement that must be met for safe storage. The storage configurations must also comply with acceptable interface requirements. In its application, the licensee submitted Westinghouse Report, WCAP-17400-P Revision 0, documenting PINGP's SFP criticality analysis. The proposed changes to TS 3.7.17, "Spent Fuel Pool Storage" and TS 4.3.1, "Fuel Storage Criticality" impose the storage requirements reflecting the new SFP criticality analysis.

The supplemental letters dated May 16, 2012, September 4, 2012, February 8, 2013, and July 17, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 14, 2012 (77 FR 8291).

Portions of the licensee's August 19, 2011, May 16, 2012, and September 4, 2012, submittals contain proprietary information and are therefore, withheld from public disclosure.

2.0 REGULATORY EVALUATION

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), 10 CFR 50.36(c)(3), and 10 CFR 50.36(c)(4).

Paragraph 50.68(b)(1) of 10 CFR requires, "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."

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

Paragraph 50.68(b)(3) of 10 CFR requires, "If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

Paragraph 50.68(b)(4) of 10 CFR requires, "If no credit for soluble boron is taken, 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 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."

Paragraph 10 CFR 50.68(c) stipulates that, "While a spent fuel transportation package approved under Part 71 of this chapter or spent fuel storage cask approved under Part 72 of this chapter is in the spent fuel pool: (1) The requirements in § 50.68(b) do not apply to the fuel located within that package or cask; and (2) The requirements in Part 71 or 72 of this chapter, as applicable, and the requirements of the Certificate of Compliance for that package or cask, apply to the fuel within that package or cask."

The U.S. Atomic Energy Commission (AEC) issued its Safety Evaluation (SE) for PINGP before the revised General Design Criteria (GDCs) were published in 1971. A PINGP GDC requires that, "Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls."

On September 29, 2011, the NRC staff issued the 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 nuclear criticality analyses and operations. The NRC staff used ISG DSS-ISG-2010-01 for the review of the current application.

As guidance for reviewing criticality analyses of fuel storage at light-water reactor power plants, the NRC staff issued an internal memorandum on August 19, 1998 (ADAMS Accession No. ML003728001). This memorandum is known as the "Kopp Letter," after the author, Laurence Kopp. The Kopp Letter provides guidance on salient aspects of a criticality analysis. The guidance is germane to boiling-water reactors and pressurized-water reactors, and to borated and unborated conditions.

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 NRC staff used the following regulatory guidance in reviewing the human performance aspects of the license amendment request:

NUREG-1764, "Guidance for the Review of Changes to Human Actions," provides guidance to determine the appropriate level of human factors engineering (HFE) review of human actions credited for safety, based on their risk importance.

NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2, was used by the NRC staff to review licensees' HFE programs to verify that these programs incorporate HFE practices and guidance accepted by the staff.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" Chapter 18.0, "Human Factors Engineering," Revision 1, provides guidance to the NRC staff in the review of the HFE aspects of modifications affecting risk-important human actions.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes

The licensee proposed seven different storage configurations and seven different fuel categories. Each fuel category, except for the category for the fresh 5 weight percent ^{235}U fuel, has a burnup versus initial enrichment requirement that must be met for safe storage. The storage configurations must also comply with the interface requirements. The proposed changes to TS 3.7.17, "Spent Fuel Pool Storage" and TS 4.3.1, "Fuel Storage Criticality" impose the storage requirements reflecting the new SFP criticality analysis. Specific changes have also been proposed to improve the TS structure and expand the scope to include the placement of fuel inserts and other hardware that may affect criticality.

3.1.1 Changes to TS 3.7.17, "Spent Fuel Pool Storage"

The proposed change will remove TS Figure 3.7.17-1, which has been used to discriminate between restricted and non-restricted locations in the SFP. The licensee proposed to refer personnel to TS 4.3.1.1 for all SFP loading restrictions, thereby eliminating the concept of "restricted" and "unrestricted" loading in the SFP (i.e., all loading will be "restricted"). In addition, the proposed TS 3.7.17 will expand the LCO and Surveillance Requirement (SR) 3.7.17.1 to address the storage of fuel inserts and hardware.

3.1.2 Changes to TS 4.3.1, "Fuel Storage Criticality"

Regarding the proposed change to TS 4.3.1.1.c, the value of SFP boron concentration that ensures that conditions in the SFP maintain a neutron multiplication factor (keff) less than or equal to 0.95 will be revised. Additionally, the proposed change will modify sub-item (e) to apply the new loading requirements of TS Figure 4.3.1-1 and expand the scope to include fuel inserts and hardware. The proposed change will replace current loading restrictions in TS Figures 4.3.1-1 through 4.3.1-4 with new loading restrictions embodied by TS Tables 4.3.1-1 through 4.3.1-3 and TS Figure 4.3.1-1.

The proposed change will also delete TS 4.3.1.3 in its entirety with no replacement. Currently, this TS has imposed a 10 CFR Part 72 spent fuel cask loading restriction that is more appropriately addressed in PINGP Independent Spent Fuel Storage Installation TSs (Special Nuclear Material license SNM-2506). The proposed Figure 4.3.1-1 makes provisions for spent fuel pool contents that were not previously included in the TS. These contents include consolidated rod storage canisters, failed fuel baskets, and fuel assembly inserts.

3.2 Reactor Systems

3.2.1 Method of Review

With its application, the licensee submitted Westinghouse Report, WCAP-17400-P Revision 0, documenting PINGP's criticality analysis. The NRC staff's review was performed consistent with Section 9.1.1 of NUREG-0800.

On September 29, 2011, the NRC staff issued the 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 nuclear criticality analyses and operations. The NRC staff used ISG DSS-ISG-2010-1 for the review of the current application.

On August 19, 1998, the NRC staff issued an internal memorandum containing guidance for reviewing criticality analyses of fuel storage at light-water reactor power plants. This memorandum is known colloquially as the "Kopp Letter" (ADAMS Accession No. ML003728001), after the author, Laurence Kopp. While the Kopp Letter does not specify a methodology, it does provide some guidance on the more salient aspects of a nuclear criticality safety (NCS) analysis, including computer code validation. The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), and to borated and unborated conditions. The NRC staff used the Kopp Letter for the review of the current application.

3.2.2 SFP NCS Analysis Review

3.2.2.1 Computational Methods

For the criticality calculation, the licensee used SCALE Version 5.1, with the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross section library. SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design, developed and maintained by Oak Ridge National Laboratory under contract with NRC and the U.S. Department of Energy (DOE) to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs. For the depletion calculation to determine the spent fuel isotopics, the licensee used the two-dimensional PARAGON code with an Evaluated Nuclear Data File, Version 6 (ENDF/B-VI) neutron cross section library. PARAGON has been approved by the NRC for depletion analysis (ADAMS Accession No. ML042250311). These computer codes and the nuclear data sets with them have been used in many NCS analyses, and are industry standards. Therefore, the NRC staff considers their use in the current application to be acceptable.

3.2.2.2 Criticality Code Validation

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation. The ISG DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

NUREG/CR-6698 states, in part, that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698 as guidance for review of the code validation methodology presented in the application. The basic elements of validation are outlined in NUREG/CR-6698, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

SCALE used in both the code benchmark analysis and the fuel storage analysis, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-III, and KENO V.a (KENO). The licensee performed the validation of the CSAS25 sequence by comparing KENO calculated k-effective values with several different sets of critical configurations. A total of [[ ]][1] critical configurations were included. The licensee determined and applied separate sets of bias and bias uncertainty based on the specific storage conditions. The sources of critical configurations [[

]]

[[

]] Therefore, the NRC staff considers [[ ]] to be an appropriate source of information for the critical experiment models. Critical experiments [[ ]] contain important features such as soluble boron and poisoned fuel rods. [[

]] The NRC staff has reviewed the experiments used in the validation of the criticality code for the PINGP SFP and considers them appropriate for that use.

Fission product k-effective validation was identified by the applicant as a validation gap. The analysis uses [[ ]] of the fission product worth as an uncertainty to cover the fission product validation gap. The value is based on the licensee's engineering judgment of fission product nuclear data. The NRC staff concludes that the approach used to determine the uncertainty and the derived values is appropriate [[

]].

The licensee identified the applicable operating conditions for the validation (e.g., fissile isotope, enrichment of fissile isotope, fuel chemical form, types of neutron absorbers, moderators and reflectors, range of moderator to fissile isotope, and physical configurations). [[

]] The licensee performed a trend analysis and identified an additional bias for those parameters with a statistically significant trend.

Based on the above discussion, the NRC staff concludes that the information supporting the code validation is acceptable.

[1] The use of double box parentheses identifies the enclosed information as proprietary.

3.2.2.3 Fuel Assembly Selection

Section 3.1 of WCAP-17400-P provides information on fuel assembly selection. The licensee analyzed current, future, and all legacy fuel assembly designs used or expected to be used at PINGP to establish the limiting fuel assembly design. Westinghouse supplied the STANDARD (STD) fuel assembly for Cycles 1 through 4 for both units. Exxon Nuclear Fuel (ENC) supplied three different fuel assembly designs, ENC STANDARD, ENC HIGH BURNUP, and ENC TOPROD for cycles 5 through 10. Beginning with Cycle 11, Westinghouse has supplied the Optimized Fuel Assembly (OFA), HIGH BURNUP OFA, 400 VANTAGE+ (400V+), and 422 VANTAGE+ (422V+) fuel assembly designs.

The licensee performed a fuel assembly reactivity comparison and concluded that the 422V+ fuel assembly design, assuming a bounding theoretical density (TD), is an acceptable reference fuel assembly design for subsequent criticality analysis. The licensee depleted the OFA, STD, and 422V+ fuel assembly designs covering the applicable burnup range. The licensee used the resulting isotopic information in KENO to determine the reactivity of each fuel assembly design. A comparison showed that the reactivity of STD and 422V+ was comparable throughout the burnup range. The NRC staff noted that the comparison was performed using only a single array configuration, [[

]].

The active fuel of the STD fuel assembly is 0.75 inches longer and uses Zircaloy-4 as the cladding material. The 422V+ fuel assembly uses the ZIRLO™ material. The licensee conservatively modeled the less neutron-absorbing cladding material to maximize reactivity.

The reactivity of OFA was [[

]].

The ENC fuel is significantly less reactive compared to the 422V+ fuel. The licensee determined that the ENC fuel assembly designs are about 800 to 2000 pcm less reactive compared to the 422V+ depending on the burnup. The three ENC fuel types have a stack density of [[ ]] TD as compared to the [[ ]] TD assumed for 422V+.

Based on the noted design considerations, the NRC staff considers acceptable the selection of 422V+ as the reference assembly design.

3.2.2.4 Depletion Analysis

Section 3.3 of WCAP-17400-P provides information on the depletion analysis. To take credit for the reduction in reactivity due to fuel burnup, the spent fuel composition should be based on an

appropriate depletion analysis with proper assumptions regarding depletion uncertainty, depletion parameters, and axial burnup and temperature profiles.

3.2.2.4.1 Depletion Uncertainty

The licensee used the two-dimensional PARAGON code to calculate the isotopic composition of the spent fuel as a function of fuel burnup, initial feed enrichment, and decay time. The NRC staff has approved PARAGON for PWR depletion calculations as a part of its approval of "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (ADAMS Accession No. ML042250311). The uncertainty in the k_{eff} introduced by the depletion isotopic uncertainty was addressed by applying [[
]] as an uncertainty component in the determination of the maximum k_{eff}. The NRC staff concludes that this uncertainty treatment is acceptable because it is consistent with ISG DSS-ISG-2010-01.

3.2.2.4.2 Depletion Parameters

The ISG DSS-ISG-2010-01 provides guidance that depletion simulations should be performed with parameters that maximize the reactivity of the depleted fuel assembly. The ISG DSS-ISG-2010-01 references NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (Reference 5) which discusses the treatment of depletion parameters. For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize plutonium production.

The NUREG/CR-6665 also recommends using a conservative cycle average boron concentration for the depletion analysis. The licensee used a boron concentration that bounds past and anticipated future cycle average boron concentrations for both units.

The NUREG/CR-6665 does not have a specific recommendation for specific power and operating history. The NUREG/CR-6665 estimated this effect to be about 0.002 Δk_{eff} using operating histories it considered. Based on the difficulty of reproducing a bounding or even a representative power operating history, NUREG/CR-6665 merely recommends using a constant power level and retaining sufficient margin to cover the potential effect of a more limiting power history. The licensee used a [[ ]] for the depletion calculations and applied a [[ ]] uncertainty on the operating history.

To date, PINGP has used two different types of burnable poisons: Burnable Poison Rod Assemblies (BPRAs), and gadolinia integrated fuel rods. The licensee confirmed that no other reactivity control device has been used at PINGP. The licensee stated that the BPRA bearing assemblies were only used in the first four cycles of each unit and these assemblies were enriched to a maximum of 3.4 weight percent ^{235}U. For these assemblies, the licensee determined the isotopic number densities assuming conservative conditions and a bounding BPRA loading to maximize the reactivity of the discharged fuel. The licensee calculated a separate set of burnup versus enrichment curves for these assemblies used in the first four cycles.

The licensee stated that the current and future approach for power distribution control at PINGP is to use gadolinia which is included as an integral part of the fuel matrix. The NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," showed that throughout burnup, the reactivity for fuel assemblies containing gadolinia remain lower than the reactivity for fuel assemblies without gadolinia due to the residual poison that will not completely

burn out of an assembly. Based on this result, the licensee concluded that not crediting gadolinia is conservative and modeled the fuel accordingly. The NRC staff noted that the referenced NUREG-6760 analysis was based on a cask analysis with poison plates and the corresponding conclusions may or may not be applicable to the PINGP spent fuel pool analysis which assumed un-poisoned racks. The licensee provided site-specific analysis that, given the assumptions made in that analysis, demonstrates that the PINGP method of addressing the effects of gadolinia in its fuel is acceptably conservative.

The licensee considered the impact of rodded operation as it affects the reactivity of the discharged assembly. The licensee elected to treat the assemblies that experienced rodded operation in two parts. For fuel assemblies from Cycles 1 through 4, the licensee concluded that depletion with burnable poison fully inserted in the BPRAs would bound depletion with partially inserted RCCAs. The NRC staff finds this reasonable since duration of rodded operation would be bounded by the depletion with BPRA over the life of the fuel. For fuel assemblies from Cycle 5 to the present and into the future, the licensee claimed that up to 1 GWd/MTU of rodded operation would be bounded by the design basis analysis used to develop the burnup requirements. In response to NRC staff RAIs, the licensee provided quantitative information to support the proposed design basis loading curves.

Rodded Operation

In its February 8, 2013, letter, the licensee proposed the following commitment:

> In conjunction with implementation of the proposed TS, procedures will be revised to require an assessment of a fuel assembly's exposure to rodded power operation in the core prior to moving that fuel assembly into the spent fuel pool (SFP) storage racks. If an assembly experiences more than 100 megawatt day per metric ton uranium (MWd/MTU) of core average full-power rodded operation exposure in the cycle immediately prior to discharge to the spent fuel pool, this exposure experienced while rodded will not be credited for determining the coefficients used to categorize fuel assemblies as described in WCAP-17400-P. In addition, if an assembly experiences more than 1 gigawatt day per metric ton uranium (GWd/MTU) of core average rodded operation lifetime exposure, the assembly shall be either treated as Fuel Category 1 or evaluated to determine which Fuel Category is appropriate for safe storage of the assembly.

In its July 17, 2013, letter, the licensee proposed the following revisions to the commitment:

- Striking the phrase "in the cycle immediately prior to discharge to the spent fuel pool."
- Striking the last sentence of the commitment.

With these changes the commitment now reads as follows:

> In conjunction with implementation of the proposed TS, procedures will be revised to require an assessment of a fuel assembly's exposure to rodded power operation in the core prior to moving that fuel assembly into the spent fuel pool (SFP) storage racks. If an assembly experiences more than 100 megawatt days per metric ton uranium (MWd/MTU) of core average full-power rodded operation exposure, this exposure experienced while rodded will not be credited for determining the coefficients used to categorize fuel assemblies as described in WCAP-17400-P.

Rodded operation will affect the reactivity of discharged spent nuclear fuel (SNF) in several ways. The thermal neutron absorption in the control rods will harden the neutron spectrum resulting in increased production of fissile plutonium isotopes. It will also skew the axial burnup profile. These effects would raise the SNF's net reactivity. However, the fuel assembly would also likely see reduced moderator and fuel temperatures, effects that would reduce the SNF's net reactivity. The licensee has rod insertion limits that allow them several options to operate at power with rods inserted. The NRC staff did not believe the licensee's analysis fully addressed the variability that would be required to estimate the reactivity of SNF that had experienced significant amounts of rodded operation. The current version of the commitment does not allow for significant amounts of rodded operation to be considered when determining whether a fuel assembly meets the storage requirements. Therefore, the NRC concludes that this commitment acceptably accounts for the variability of rodded operation.

A requirement is included in Section 3, "Implementation Requirements," of the NRC license amendments associated with this safety evaluation, in order to ensure that the proposed procedure changes in the above commitment are incorporated coincident with the licensee's implementation of the amendments.

3.2.2.4.3 Axial Burnup and Temperature Profiles

At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," has shown that, at assembly burnups above about 10 to 20 GWd/MTU, this results in an underprediction of k-effective. Generally the underprediction becomes larger as burnup increases. This is what is known as the "end effect."

Proper selection of the axial burnup profile is necessary to ensure k-effective is not underpredicted due to the end effect.

The NUREG/CR-6801 provides recommendations for selecting an appropriate axial burnup profile. The NCS analysis documented in WCAP-17400-P did not use the axial burnup profiles from NUREG/CR-6801. A description of how the axial burnup profiles were derived is provided in Section 3.3.3 of WCAP-17400-P. The licensee used [[ ]] axial burnup profiles from past and current core designs to derive five burnup profiles corresponding to [[

]].

When the analysis was initially performed, it included the anticipation of a future extended power uprate (EPU) license amendment request (LAR) and several aspects of the analysis were intended to bound that EPU. However, in its February 8, 2013, letter (ADAMS Accession No. ML13039A306), the licensee indicated that it no longer intended to submit an EPU LAR and the issue regarding the EPU's effect on post irradiated fuel has not been resolved. Additionally, the licensee may at some time in the future reconsider its decision and submit a power uprate LAR. Therefore, any effects of a future power increase are not fully covered by this license amendment and would need to be addressed at that time. From the database of profiles, the method identifies [[

]]. The NRC staff requested additional information regarding the derivation of the axial burnup profiles. Based on the licensee's response, the NRC staff determined that the method selects the axial burnup profile corresponding to a spent fuel assembly with the most underburned top region of the fuel. Therefore, the method is acceptably conservative.

In its response to an NRC staff RAI, the licensee also showed that the use of annular pellets in the axial blankets [[ ]].

Based on the above discussion, the NRC staff concludes that the licensee selected an appropriate burnup profile.

3.2.3 Criticality Analysis

3.2.3.1 Normal Conditions

The PINGP Units 1 and 2 share a common spent fuel storage pool that employs one modular storage rack design throughout the pool. The PINGP utilizes the high density racks that were designed with Boraflex neutron absorber inserts. The criticality analysis does not credit any remaining Boraflex neutron absorber material. The model assumes that the remaining Boraflex material [[ ]]. The criticality analysis is based on [[ ]]. During normal operation, at least 1800 ppm of soluble boron is present in the SFP and the moderator temperature is less than or equal to 150 °F. The licensee proposes seven normal storage configurations for use throughout the PINGP spent fuel pool.

The manufacturing tolerances of the storage racks and fuel assemblies contribute to the reactivity. Consistent with the Kopp Letter, the determination of the maximum k-effective should consider either: (1) a worst-case combination with mechanical and material conditions set to maximize k-effective, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the racks.

The licensee's analysis evaluated the following uncertainty components: [[ ]]. The NRC staff found that this assumption was acceptable in section 3.2.2.3 above.

To determine the reactivity uncertainty associated with a specific manufacturing tolerance, the licensee used KENO to calculate the k-effective for the reference condition and the k-effective for the perturbed case. The reference condition is the condition with nominal dimensions and properties. All tolerance perturbations were applied in the direction that increases reactivity relative to the nominal condition. [[ ]] Based on the considerations discussed above, the NRC staff finds the licensee's treatment of manufacturing tolerances acceptable because it is consistent with the guidance in the Kopp Letter.

According to the Kopp Letter, the criticality analysis should account for the temperature corresponding to the highest reactivity. The licensee performed the base criticality analysis at [[ ]]. To address the full range of allowable operating temperatures, the licensee performed additional KENO calculations [[ ]]. The maximum positive difference was applied as a bias to the maximum k-effective.

The NRC staff concludes that this is consistent with the Kopp Letter, and, therefore, acceptable.

3.2.3.2 Abnormal Conditions

Section 4.5 of WCAP-17400-P presents the abnormal conditions considered in the analysis. The licensee considered the following abnormal conditions:

- misloaded assembly
- inadvertent removal of an RCCA
- spent fuel temperature outside of operating range, and
- dropped and misplaced fuel assembly

The licensee determined that the limiting abnormal condition was the misloaded assembly. The licensee created models for the misloaded assembly by placing a fresh 5.0 weight percent fuel assembly into a water-filled cell for the storage configurations requiring water filled cells (Arrays B, C, D, and E). The licensee's analysis determined that the limiting Array [[ ]] would require 910 parts per million (ppm) of soluble boron to comply with the regulatory k-effective limit of 0.95.

The licensee did not explicitly calculate the k-effective for the inadvertent removal of an RCCA from Array G claiming that the misload event is bounding. Removal of an RCCA from Array G results in an array uniformly loaded with Category 5 fuel assembly. The misload event results in an array with two fresh assemblies (Category 1) face adjacent and two Category 5 fuel assemblies face adjacent in a 4x4 array. Therefore, the NRC staff agrees with the licensee that the limiting misload event would bound the condition following an inadvertent RCCA withdrawal.

To cover the SFP heat up conditions, the licensee analyzed the pool with a moderator density of 0.96 grams per cubic centimeter, which corresponds to the lowest water density at boiling point at atmospheric conditions. In addition, the licensee analyzed the pool with a moderator density of 0.75 grams per cubic centimeter. The k-effective results showed that both cases were bounded by the misload event.

The licensee stated that the dropped fuel assembly is non-limiting because of the separation between a dropped assembly lying across the top of the fuel assembly top nozzle and the active fuel region, would be less reactive than a misloaded fuel assembly.

The licensee stated that the misplacement of a fuel assembly alongside the racks is less reactive than the misloaded fuel assembly because the misplaced fuel assembly will be bordered on two sides by water, and the leakage present in that scenario would make for a less reactive arrangement than a misloaded fuel assembly, which would have fuel on all sides.

The accidents considered are reasonable considering the guidance in the Kopp Letter and the margin available between the amount of soluble boron required to maintain k-effective less than or equal to 0.95 and the soluble boron concentration required by the technical specifications.

Based on the above discussion, the NRC staff finds the licensee's evaluation of the accident conditions acceptable.

3.2.4 Staff's Evaluation of Elimination of TS 4.3.1.3 from TSs

TS 4.3.1.3 was added to the PINGP TSs by Amendment Nos. 99 and 92 for Unit 1 and Unit 2, respectively, on July 9, 1992 (ADAMS Accession No. ML022210492). This pre-dates the promulgation of 10 CFR 50.68, which was originally issued in November 1998. The NRC revised 10 CFR 50.68 in November 2006 to add paragraph 10 CFR 50.68(c). Paragraph 10 CFR 50.68(c) was added to address the issue identified in NRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations."

With the issuance of the November 2006 revision to 10 CFR 50.68, there was no longer a need for licensees to perform separate nuclear criticality safety analyses for a cask when it was in the spent fuel pool. Since the November 2006 revision to 10 CFR 50.68, the nuclear criticality safety requirements for a cask are governed by the applicable Part 71 and/or Part 72 license. Therefore, the inclusion of cask loading nuclear criticality safety requirements in the Part 50 license is duplication and creates the possibility of an inadvertent violation if the Part 50 license is not revised when the Part 71 and/or Part 72 license is revised. Therefore, the NRC staff finds it acceptable to delete TS 4.3.1.3 in its entirety.

3.2.5 Summary

The licensee submitted an LAR to address the non-conservatisms in the spent fuel pool nuclear criticality safety analysis of record and the associated TSs. The LAR is supported by Westinghouse Report, WCAP-17400-P, Revision 0, which documents the criticality analysis for PINGP spent fuel storage. The proposed changes to TS 3.7.17, "Spent Fuel Pool Storage" and TS 4.3.1, "Fuel Storage Criticality" impose the storage requirements reflecting the new SFP criticality analysis.

The NRC staff reviewed the analysis to ensure that the assumptions and analytical technique used are adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirements will be met.

Based on the discussions in Section 3.2 of this safety evaluation, the NRC staff finds reasonable assurance that PINGP will comply with the applicable regulatory requirements. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

3.3 Human Performance and Health Physics

3.3.1 Potential Effects on Human Performance

In its application, the licensee stated that "... the proposed amendments involve no physical modifications to the SFP storage racks or to any other system, structure, or component. No change to the minimum SFP boron concentration limit is required. The only physical effect associated with this proposed amendment will be the reconfiguration of fuel in the SFP storage racks." With respect to the possibility of increased risk due to human errors, the NRC staff reviewed the potential impact of the proposed changes to TS loading restrictions on

opportunities for error, and the current or proposed barriers in place to prevent or mitigate human errors.

3.3.1.1 Changes to Design Basis Operator Actions

The NRC staff has determined that the proposed changes do not impact the operator actions or their timing as credited in the design basis accidents analyzed in Section 14.5.1 "Fuel Handling," of the PINGP Updated Safety Analysis Report (USAR) and that it is appropriate to continue to credit these actions.

3.3.1.2 Changes to Fuel Characterization

Regarding the actions involved in characterizing fuel, the licensee stated that the only change is the use of Core Operating Cycle instead of gadolinium content to distinguish which TS table applies to a specific irradiated fuel assembly. Because both of these parameters are obtained from procurement or operating records, the actions associated with these two parameters (Core Operating Cycle and gadolinium content) are essentially the same. From the human factors standpoint, the risk associated with identifying the wrong Core Operating Cycle is equivalent to the current risk of mistakenly identifying a non-gadolinium assembly as a gadolinium assembly. Therefore, the NRC staff concludes that there is no appreciable increase in the probability of human error due to this proposed change. The following barriers to prevent characterization errors are currently used and will continue to be used after the proposed changes are implemented:

- Training of nuclear engineers and operators
- Qualification of nuclear engineers and operators
- Documentation of fuel assembly characteristics (availability for use in the characterization process)

3.3.1.3 Changes to Fuel Categorization

A fuel assembly's category determines which storage arrays are acceptable for which assemblies. The process for categorization is being changed to make it more consistent. Previously, fuel assemblies were first screened as "restricted" or "non-restricted". Now, this pre-screening is not necessary – all assemblies are considered "restricted" and therefore, go through a thorough categorization process based on reactivity. New categories have been created to cover everything from the highest reactivity (Category 1) to lowest (Category 6) and an additional Category 7 for Consolidated Rod Storage Canisters. Once the "ShuffleWorks" software in use at PINGP has been updated with the definitions of the new categories, "ShuffleWorks" will categorize a fuel assembly based on its burn-up requirements. From this point, the process will remain the same as current practices. Therefore, the NRC staff concludes that there is no appreciable increase in the probability of human error due to this proposed change.

The following barriers to prevent categorization errors are currently used and will continue to be used after the proposed changes are implemented:

- Experience of nuclear engineers and operators
- Qualification of nuclear engineers and operators
- Operating experience with "ShuffleWorks" software
- Independent verification of "ShuffleWorks" output (Fuel Transfer Logs (FTLs))

- Independent categorization of affected assemblies and preparation of FTLs by qualified individuals

3.3.1.4 Configuration Control

The licensee stated that the process of moving fuel is currently controlled by procedure and that the overall process will not change. That is, fuel movers will continue the current practice of following FTLs when moving and storing fuel assemblies. The nuclear engineers who develop the FTLs will be given one-time training in the proposed, new TS criticality requirements and storage configuration requirements.

The only substantive change in configuration control in the SFP is that cells that are identified as empty cells will be controlled like any other cell. Affected personnel will be notified that empty cells in the SFP must remain empty and a note will be included in the TS to prevent the previously acceptable practice of storing inconsequential, non-fuel hardware in empty cells. There are multiple barriers in place to prevent mis-positioning of an assembly. None of these will change or be negatively impacted by the proposed change to the TS. Therefore, the NRC staff concludes that there is no appreciable increase in the probability of human error due to this proposed change.

The following barriers to prevent fuel movement errors are currently used and will continue to be used after the proposed changes are implemented:

- Approved procedures
- Independent verification of fuel categorizations
- Independent verification of FTLs
- Use of a Fuel Handling Supervisor or Fuel Accountability Engineer to verify "to" and "from" locations
- Movement by one step at a time
- Use of a step verifier
- Pre-job briefs
- Use of three-way communication
- Post-campaign inventory

3.3.2 Summary

Based on the discussions in Section 3.3 of this safety evaluation, the NRC staff concludes that the proposed TS changes do not affect manual actions credited in PINGP USAR, Chapter 14, Section 14.5.1, "Fuel Handling." The staff further concludes that the proposed TS changes are acceptable from the human performance perspective based on the licensee's statements that appropriate training will be provided, procedures and other administrative controls will continue to be applied, qualified nuclear engineers will be used to categorize fuel assemblies and to identify proper storage locations, and fuel movement equipment, including controls, displays, and alarms, will not be affected.

3.4 Licensing Basis Change for SFP Criticality

In its August 19, 2011, application, the licensee requested NRC approval to change its regulatory basis for SFP criticality analysis from 10 CFR 70.24 to 10 CFR 50.68(b), which would allow for elimination of criticality accident monitoring requirements, while maintaining the subcriticality criteria defined by 10 CFR 50.68(b). The regulations at 10 CFR 50.68(a) state that

the licensee shall comply with either 10 CFR 70.24 or 10 CFR 50.68(b). Therefore, the licensee may comply with 10 CFR 50.68(a) by complying with 10 CFR 50.68(b) in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24. The licensee does not require prior NRC approval to comply with 10 CFR 50.68(a) in this manner.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements 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 amendments involve 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (77 FR 8291). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: August 29, 2013

The NRC has determined that the related Safety Evaluation (SE) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Proprietary information is indicated by text enclosed within double brackets. Accordingly, the NRC staff has also prepared a redacted publicly available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 209 to DPR-42
2. Amendment No. 196 to DPR-60
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

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