February 5, 2006

Mr. Thomas J. Palmisano
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: "SPENT FUEL POOL STORAGE" (TAC NOS. MC5811 AND MC5812)

Dear Mr. Palmisano:

The Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-42 and Amendment No. 162 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to theTechnical Specifications (TSs) in response to your application dated February 1, 2005, supplemented by letters dated February 22, September 16, December 2, 2005, and January 5, 2006.

The amendments revise the spent fuel pool criticality analysis methodology and technical specifications governing the storage of irradiated fuel in the spent fuel pool. The staff concludes that subcritical conditions would be maintained in the spent fuel pool under the revised technical specification storage requirements.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Mahesh L. Chawla, 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. 172 to DPR-42
2. Amendment No. 162 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

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OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	SPWB/BC	ITSB/BC	OGC	NRR/LPL3-1/BC (A)
NAME	MChawla	THarris	JNakoski	TBoyce	AHodgdon	TKobetz
DATE	2/7/06	2/7/06	1/27/06	1/30/06	2/1/06	2/5/06

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-42

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated February 1, 2005, supplemented by letters dated February 22, September 16, December 2, 2005, and January 5, 2006, 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 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. 172, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy J. Kobetz, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 5, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

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	3.7.17-3
3.7.17-4	----------
4.0-2	4.0-2
4.0-4	4.0-4
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	----------
4.0-11	----------
4.0-12	----------
4.0-13	----------
4.0-14	----------
4.0-15	----------
4.0-16	----------

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-60

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated February 1, 2005, supplemented by letters dated February 22, September 16, December 2, 2005, and January 5, 2006, 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 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. 162, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy J. Kobetz, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 5, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

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	3.7.17-3
3.7.17-4	----------
4.0-2	4.0-2
4.0-4	4.0-4
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	----------
4.0-11	----------
4.0-12	----------
4.0-13	----------
4.0-14	----------
4.0-15	----------
4.0-16	----------

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 162 TO FACILITY OPERATION LICENSE NO. DPR-60

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated February 1, 2005 (Reference 1), as supplemented by letters dated February 22, September 16, December 2, 2005, and January 5, 2006 (Refs. 2, 3, and 4), Nuclear Management Company (NMC) requested approval of a license amendment for Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP). NMC requested this amendment to revise its criticality analyses and technical specifications (TS) governing the storage of irradiated fuel in the spent fuel pool (SFP).

The PINGP TSs currently permit the licensee to store 1386 fuel assemblies in the SFP, not including those assemblies that can be returned to the reactor. Additionally, the TSs permit the licensee to install four additional temporary storage racks, with a combined capacity of 196 assemblies, in the cask lay down area of the SFP. NMC's proposed changes to the TSs would result in less restrictive storage requirements for irradiated fuel assemblies that continue to satisfy the Nuclear Regulatory Commission's (NRC's) regulatory safety criteria for fuel storage.

The NRC staff reviewed the amendment request to ensure that NRC regulations were satisfied and that public health and safety was maintained. The NRC staff determined that the amendment request does satisfies NRC regulations regarding the safe storage of irradiated fuel in the SFP. Additionally, the NRC staff determined that the licensee's criticality analysis methodology was performed in accordance with NRC guidance documents. Therefore, the NRC staff finds that the proposed criticality analysis and TS changes are acceptable.

Section 2.0 of this safety evaluation describes the regulatory requirements associated with the proposed changes. The NRC staff's technical evaluation is provided in Section 3.0 of this safety evaluation.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements and Review Documents

Title 10 of the *Code of Federal Regulations*, Part 50 Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," (Ref. 5) provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, "Prevention of criticality in fuel storage and handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes. The staff reviewed the amendment request to ensure that the licensee complied with GDC 62. During its review, the NRC staff determined that PINGP was licensed prior to the issuance of the GDC in Appendix A of Part 50. As such, PINGP was designed and constructed to comply with the Atomic Energy Commission General Design Criteria (AEC GDC) as proposed on July 10, 1967. These criteria are described in PINGP's Updated Safety Analysis Report. Specifically, PINGP complies with AEC GDC 66, which states:

> 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.

AEC GDC 66 is nearly identical to GDC 62 in wording and intent. A licensee that satisfies either of these GDCs meets the NRC's intent of minimizing the potential for a criticality event during new and spent fuel storage.

The current licensing basis for the PINGP spent fuel pools is WCAP-14416-NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," (Ref. 6). In a letter dated June 12, 1997 (Ref. 7), the NRC issued Amendments 129 and 121 for PINGP, Units 1 and 2, respectively, approving this licensing basis. In that amendment, NMC took advantage of relaxed criticality control requirements that ensured the effective multiplication factor (k_{eff}) of the spent fuel pool was less than 1.0 if flooded with unborated water, and less than or equal to 0.95 if flooded with borated water. These requirements provide reasonable assurance that a inadvertent criticality event is precluded in the spent fuel pool.

In addition to the licensee's use of the WCAP-14416 methodology, PINGP was originally licensed to and continues to comply with 10 CFR 70.24, "Criticality accident requirements," (Ref. 8). In accordance with the requirements of 10 CFR 70.24, the licensee maintains a criticality monitoring system capable of detecting a criticality event and initiating appropriate alarms to alert station personnel to evacuate. The licensee's Amendment No. 12 to its Final Safety Analysis Report (Ref. 9), definitively states that the license complies with 10 CFR 70.24.

On November 2, 2000, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 00-015, "Axial Burnup Shape Reactivity Bias," (Ref. 10). NSAL 00-015 described non-conservatism(s) in the three-dimensional axial burnup biases used in WCAP-14416-NP-A. In response to the identified non-conservatism(s), on July 27, 2001, the NRC staff sent a letter to Westinghouse (Ref. 11) withdrawing approval for WCAP-14416-NP-A. In its letter, the NRC staff concluded that WCAP-14416-NP-A may no longer be relied upon as an "approved methodology" by the NRC staff or licensees. Additionally, the staff stated that for future licensing actions, licensees would need to submit plant-specific criticality analyses for spent fuel pool configurations that

included technically supported margins. Therefore, for the purposes of its February 1, 2005, license amendment request (LAR), NMC did not rely upon the WCAP-14416-NP-A methodology, but instead submitted plant-specific criticality analyses in accordance with the staff's guidance documents.

The NRC staff provided guidance and defined acceptable methodologies for performing SFP criticality analyses in three documents:

1. NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4 (Ref. 12);

2. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," (Ref. 13); and

3. Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (Ref. 14).

The NRC staff used the guidance contained in these documents to assist in its review of the licensee's amendment request.

2.2 Description of Proposed TS Changes

In Exhibit B of Ref. 1, PINGP provided marked-up TSs and corresponding bases pages. The staff reviewed each of these changes against the acceptance criteria described in Section 2.1 of this report and found them acceptable. The basis for the staff's acceptance and a description of the review it performed is located in Section 3.0 of this report. The following is the descriptive list of proposed changes as provided by the licensee in Exhibit B of Ref. 1:

1. TS Limiting Condition for Operation (LCO) 3.7.17, "Spent Fuel Pool Storage": NMC's LAR proposed to revise the current "All-Cell" spent fuel assembly storage figures to provide a single criticality analysis figure that bounds all fuel assembly types currently stored in the PINGP SFPs.

2. TS 4.3, "Fuel Storage": NMC's LAR revised the "3x3 Array" spent fuel assembly storage figures in this TS to account for its new bounding criticality analysis. Spent fuel assemblies stored in the "3x3 Array" configuration will be classified based on whether they are unshimmed or shimmed with gadolinium.

In addition to the proposed TS changes described in Exhibit B of Ref. 1, NMC provided the revised corresponding bases pages. In Ref. 2, NMC provided additional revisions to the TS pages in response to NRC questions regarding the storage of fuel assemblies containing gadolinium. The NRC staff reviewed these revisions in support of the proposed TS changes.

3.0 TECHNICAL EVALUATION

In determining the acceptability of PINGP's amendment request, the staff reviewed three aspects of the licensee's analyses: 1) the computer codes employed; 2) the methodology used to calculate the maximum k_{eff}; and 3) the storage configuration and limitations proposed. For

each part of the review, the staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC acceptance criteria were developed and could be maintained in the PINGP spent fuel pools during cask loading, unloading, and handling operations.

3.1 Computer Codes

The licensee performed the analysis of the reactivity effects for the SFP with SCALE version 4.3 (Ref. 15) with the 44-group ENDF/B-V neutron cross section library. SCALE version 4.3 includes the KENO V.a code, a three-dimensional Monte Carlo criticality code. The KENO V.a code was benchmarked against criticality experiments under conditions that reflect the variables for fuel storage in the SFP. The critical benchmark experiments included 19 Babcock and Wilcox experiments (Ref. 16) carried out in support of the Close Proximity Storage of Power Reactor Fuel and 11 experiments from the Pacific Northwest Laboratory (PNL) experimental program (Ref. 17). The NRC has previously accepted the use of this data for benchmarking the KENO V.a code under storage conditions similar to those proposed in the NMC amendment request (Refs. 14 and 18). The experimental data are sufficiently diverse to establish that the method bias and uncertainty will apply to spent fuel under the proposed storage conditions in the PINGP SFP. The licensee determined the KENO V.a code calculation (methodology) bias is 0.0259 with a 95/95 bias uncertainty of +/- 0.00288 using the 44-group ENDF/B-V neutron cross section library. Additionally, the licensee determined the reactivity effect (delta-k) for each manufacturing tolerance of the fuel assemblies and storage lattice based on a full-scale model representation of the spent fuel storage racks.

In addition to using the KENO V.a code to perform the criticality analyses, the licensee employed the DIT (Discrete Integral Transport) code to perform the fuel depletion analyses that were used to develop the revised TS figures. DIT performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly to determine the isotopic composition of the spent fuel as a function of fuel burnup and initial feed enrichment. The DIT code and its cross section set have been used in the design of reload cores and extensively benchmarked against operating reactor history and test data. In accordance with NRC guidance documents, the licensee applied a 5 percent burnup decrement to ensure that the results obtained for the depletion analysis were conservative.

The NRC staff reviewed the licensee's application of the codes to determine whether each could reasonably calculate, based on conservative assumptions and inputs, the appropriate parameters necessary to support the maximum k_{eff} analyses. In Ref. 10, the staff stated that KENO V.a was an acceptable computer code for the analysis of fuel assemblies stored in the SFP. Additionally, in Ref. 14, the NRC staff stated that the Babcock and Wilcox series of criticality experiments, as described in Ref. 16, provided an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control and close-packed arrays of fuel. Additionally, the NRC has previously accepted the use of the PNL experimental database employed by the licensee for performing benchmarking calculations of the codes used in the analysis (Refs. 18, 19, and 20). Therefore, the staff concludes that the licensee's use of the KENO V.a code for calculation of the nominal k_{eff} was appropriate since it was benchmarked against experimental data that adequately reflects the proposed fuel assembly and storage conditions proposed in the NMC amendment request. Additionally, the NRC has previously approved the use of the DIT code for performing depletion analyses in support of burnup credit

for spent fuel storage (Refs. 18, 19, and 20); therefore, the staff finds that the licensee's use of the DIT code was acceptable for performing the fuel depletion analyses.

3.2 Methodology

In accordance with the guidance contained in Refs. 12, 13, and 14, the licensee performed criticality analyses of the SFP storage racks under limiting storage conditions. The licensee employed a methodology that combines a worst-case analysis based on the bounding fuel and SFP conditions, with a sensitivity study using 95/95 analysis techniques. The major components in this analysis were a calculated (nominal) k_{eff} based on the limiting fuel assembly and storage configuration, spent fuel pool temperature and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

In performing its criticality analysis, the licensee first calculated a k_{eff} based on nominal SFP loading conditions using the KENO V.a code. NMC calculated this nominal k_{eff} for each fuel assembly design, fuel enrichment, and storage configuration that is considered in the scope of the SFP storage at PINGP. The licensee performed sensitivity calculations of each of the assemblies stored in the PINGP SFP to determine the most reactive fuel assembly design under normal and accident conditions. NMC determined that the Westinghouse 14 x14 Standard fuel assembly design was the most reactive fuel assembly in the "All-Cell" storage configuration and that the Westinghouse 14x14 optimized fuel assembly represented the most reactive fuel assembly when placed at the center of the "3x3 Array" storage configuration and surrounded with Westinghouse 14x14 Standard assemblies. NMC then applied the most limiting fuel designs in its normal and accident analyses to ensure that bounding k_{eff} values were determined.

In addition to determining the bounding fuel design, the licensee calculated the effects of the bounding SFP temperatures. The licensee used the minimum and maximum permissible design basis SFP temperatures and corresponding water densities to determine which resulted in the most limiting nominal k_{eff}. In Ref. 2, NMC provided the results of sensitivity calculations performed over the range of design basis SFP temperatures. The results showed that optimum moderation occurred under the full density conditions assumed in the analysis. Therefore, with respect to optimum moderation, the licensee ensured that its criticality analyses are bounding under normal storage conditions. This is consistent with NRC acceptance criteria and guidance documents.

To the calculated k_{eff}, the licensee added the methodology bias. As stated in the description of the KENO V.a code, the licensee determined the methodology bias from the critical benchmark experiments.

Additionally, to determine the maximum k_{eff}, the licensee performed a statistical combination of the reactivity effects for code and methodology uncertainties, manufacturing tolerances, and burnup uncertainties. The code and methodology uncertainties account for the mean calculational variance and uncertainty in the benchmarking of the KENO V.a code. The licensee determined this uncertainty to a 95/95 threshold which is consistent NRC acceptance criteria and guidance documents.

In addition to including the code uncertainty, the licensee performed analyses to determine appropriate and conservative fuel and storage cask mechanical tolerances as well as including

a tolerance for eccentric positioning of the fuel assemblies in the storage cells. For each tolerance, the licensee calculated a delta-k between the nominal condition and the most limiting tolerance condition. For the fuel rod manufacturing tolerances, the dominating contributors are the fuel enrichment, theoretical density, cladding thickness, and pellet diameter. For each of these contributors, the licensee included bounding and conservative effects that result in maximizing the delta-k. Likewise, for the storage cask fabrication tolerances, the licensee included conservative and bounding tolerances on key parameters, such as the cell pitch and cell inner dimension that result in maximizing the delta-k. By using the most limiting tolerance conditions, the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. In addition to manufacturing tolerances, NMC analyzed eccentric positioning of fuel assemblies in the storage rack lattice cells. NMC determined that eccentric positioning of the assemblies such that the center-to-center pitch was at its minimum, resulted in a minor increase in the k_{eff}. NMC appropriately included the bounding delta-k from eccentric positioning in its tolerance calculations.

Finally, in lieu of performing detailed burnup uncertainty analyses, the licensee applied a 5 percent burnup reactivity decrement in accordance with NRC guidance documents (Ref. 14). To calculate a burnup reactivity decrement, the licensee analyzed the difference in reactivity (Δk) between fresh and irradiated fuel assemblies under identical storage configurations. The licensee then applied a 5 percent penalty on the maximum Δk to ensure a conservative calculation of burnup credit. For example, a licensee may determine that a fresh fuel assembly array provides a k_{eff} of 1.1 while the same fuel assembly burned to 40,000 megawatt-days per metric ton uranium has a k_{eff} of 0.9. The Δk between the fresh and irradiated fuel assemblies is 0.2 and the burnup reactivity decrement is then calculated to be 0.01 (5 percent of 0.2). The licensee would then reduce the burnup credit taken from 0.2 to 0.19 (i.e., 0.2 minus 0.01). In Ref. 4, the licensee provided the results of its 5 percent burnup reactivity decrement analysis. The licensee's results demonstrate that the 5 percent reactivity decrement is conservative relative to a 5 percent burnup uncertainty analysis as presented in the original LAR (Ref. 1). The licensee's original burnup uncertainty analysis was based on an unapproved methodology and did not contain sufficient information to demonstrate NRC regulatory and safety criteria were satisfied. The inclusion of a burnup reactivity decrement in the criticality analyses, in conjunction with the use of the other conservative assumptions and inputs, assures that NRC regulatory requirements are satisfied.

The licensee's proposed TS changes place considerable emphasis in the criticality analyses on a burnup credit; therefore, the accurate determination of the burnup profile is essential to ensure the acceptance criteria for k_{eff} are satisfied. As previously stated, the licensee employed the DIT code for determining the appropriate burnup credit. For a given spent fuel assembly, the fuel burnup is a function of axial position. In performing the depletion analysis, the licensee analyzed the burnup in discrete axial zones to ensure that the axial "end-effect" was adequately captured. NMC divided the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Additionally, NMC made the top axial zone sufficiently small (typically less than 8 inches) to capture the steep burnup gradient with axial position in this region of the fuel assembly. The axial burnup profile employed in this analysis, was based on the most limiting axial burnup shape from a Department of Energy (DOE) topical report (Ref. 21). The NRC staff has approved similar axial profile models based on the DOE database in other license amendments (Refs. 18 and 20). In order to generate the isotopic concentrations for

each segment of the axial profile, appropriate fuel and moderator temperatures and soluble boron concentrations that both reflect historical operating conditions at PINGP as well as represent appropriately conservative values intended to maximize the residual reactivity of the spent fuel assemblies must be used in the depletion analysis. In Ref. 2, NMC provided additional information that demonstrated the values chosen for these parameters satisfied these criteria. The data provided by NMC demonstrated that the values used in the burnup credit analysis represented conservative assumptions for the fuel and moderator temperatures and soluble boron concentrations relative to the historical operating conditions at PINGP. Therefore, the NRC staff finds that the methodology employed and the assumptions used to perform the burnup credit analysis are acceptable.

In addition to a burnup credit, NMC proposed to credit the presence of gadolinium in some assemblies. In Refs. 1 and 2, the licensee provided a detailed description of its analysis methodology for the treatment of fuel assemblies containing gadolinium. The licensee described its analysis assumptions and provided a technical basis for why each contributed to a conservative prediction of a spent fuel assembly's residual reactivity. The NRC staff reviewed the licensee's analysis, including its methodologies and assumptions, and concludes that they are both conservative and acceptable.

Once the reactivity effects for each of the tolerances and uncertainties were determined, the licensee statistically combined these results in accordance with the guidance contained in Ref. 14. The NRC staff reviewed the licensee's methodology for calculating each of the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values. The NRC staff finds the licensee's methods for calculating the maximum k_{eff} conservative and acceptable.

3.3 Proposed Storage Configuration

The primary purpose of the licensee's amendment request was to gain the NRC staff's approval for revised storage requirements within the spent fuel storage racks in the PINGP spent fuel pool. The licensee's revised TSs would permit unrestricted storage of spent fuel assemblies in the "All-Cell" configuration provided each assembly satisfied minimum burnup requirements as a function of initial enrichment. Additionally, PINGP proposed TSs to permit the storage of fuel assemblies in a "3x3 Array" with credit for the presence of gadolinium. The "3x3 Array" storage configuration permits the storage of fresh, unirradiated fuel assemblies in the center storage location. The storage of assemblies in both the "All-Cell" and "3x3 Array" configurations is controlled based on revised TS figures restricting the burnup of assemblies based on their initial enrichment. Assemblies with burnups greater than those on the figures, may be stored in accordance with the revised TS storage configurations.

In determining the acceptable burnup versus enrichment curves, NMC used the codes and methodologies described in Sections 3.1 and 3.2, respectively, of this report. In developing the burnup versus enrichment curves, the licensee included a cooling time credit. This cooling time credit is based on the decay of residual fissile materials in the spent fuel assemblies and the buildup of neutron absorbing elements. In Ref. 2, the licensee provided a discussion of how it will conservatively apply the cooling time credit to spent fuel assemblies.
The methodology employed by the licensee for determining the cooling time credit has been approved by the NRC staff in previous license amendments and, therefore, is acceptable.

TS Figures 3.7.17-1 and 4.3.1-3 provide the fuel assembly burnup limit requirements for storage of spent fuel assemblies in the “All Cell” and “3x3 Array” configurations, respectively. These figures depict the limiting burnup as a function of initial fresh fuel enrichment required to store spent fuel assemblies in the associated configurations at PINGP SFP. For the “All Cell” configuration, an assembly with a burnup greater than the limits in Figure 3.7.17-1 may be may be stored in the “All Cell” region of the SFP without restrictions on its location. For the “3x3 Array” configuration, an assembly with a burnup greater than the limits on Figure 4.3-1.3 may be stored in “burned” assembly location as shown in TS Figure 4.3.1-1. This allows the licensee to store fresh fuel assemblies in the “3x3 Array” configuration in locations adjacent to sufficiently irradiated fuel assemblies. In developing these burnup versus enrichment curves, NMC performed KENO V.a analyses, as described previously, based on limiting storage conditions. To ensure that the NRC acceptance criteria were satisfied, NMC set its target value of k_{eff} at its self-imposed limit of 0.995 minus the magnitude of the limiting analytical biases and uncertainties. The sum of the biases and uncertainties was conservatively calculated to be 0.02901 for the “All Cell” storage configuration. For the “3x3 Array” storage configuration, the licensee calculated the sum of the biases and uncertainties to be 0.02977 for gadolinium shimmed spent fuel assemblies and 0.2800 for unshimmed fuel assemblies. The licensee developed a third order polynomial of limiting assembly burnup as a function of initial enrichment from these data. This polynomial will be used to determine the acceptability of assemblies for storage in the SFP.

In addition to analyzing the proposed loading configurations, the licensee performed detailed accident analyses. The accidents analyzed included the following: 1) a fresh fuel assembly dropped on top of the storage racks; 2) a fresh fuel assembly misloaded into an incorrect storage location; 3) a fresh fuel assembly misloaded between storage racks (in the gap between storage racks); 4) intramodule gap reduction due to seismic event; and 5) SFP water temperature greater than 150 EF. NMC determined that the bounding accident was the misloading of a fresh fuel assembly in the gap of water between storage modules and next to another fresh fuel assembly. NMC determined that the reactivity increase (Δk_{eff}) associated with this accident was 0.05914. Since the NRC staff does not require a licensee to assume two independent accidents occurring simultaneously, NMC calculated the amount of soluble boron required to mitigate the consequences of this accident. NMC determined that 263 ppm of soluble boron would be required to compensate for the reactivity increase caused by the worst-case misloading event and maintain the k_{eff} less than 0.95. Since the current TS LCO 3.7.16 requires the SFP boron concentration to be greater than or equal to 1800 ppm, the staff agrees that sufficient soluble boron will be available to preclude an inadvertent criticality event for this and all less severe accidents.

4.0 SUMMARY

The NRC staff reviewed the effects of the proposed changes against the current PINGP licensing basis and NRC guidance documents. In its review of the criticality analyses supporting the proposed changes, the NRC staff found that NMC employed realistically conservative assumptions, inputs, and methodologies in every step of the analysis. Based on the results of the criticality analyses, the staff found that the licensee’s amendment request

provided reasonable assurance that under both normal and accident conditions, the licensee would be able to operate the plant safely and comply with NRC regulations. Therefore, the NRC staff finds the licensee's amendment request acceptable.

The NRC staff 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) such activities will be conducted in compliance with the commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (70 FR 12748). 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.

7.0 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) 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.

8.0 REFERENCES

1. Letter from J.M. Solymossy (NMC) to U.S. Nuclear Regulatory Commission, "License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, 'Spent Fuel Storage' and 4.3, 'Fuel Storage'," dated February 1, 2005, ADAMS Accession No. ML050330138.

2. Letter from T.J. Palmisano (NMC) to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, 'Spent Fuel Storage' and

4.3, 'Fuel Storage' (TAC Nos. MC5811 and MC5812)," dated September 16, 2005, ADAMS Accession No. ML052620079.

3. Letter from T.J. Palmisano (NMC) to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, 'Spent Fuel Storage' and 4.3, 'Fuel Storage' (TAC Nos. MC5811 and MC5812)," dated December 2, 2005, ADAMS Accession No. ML053390121.

4. Letter from T.J. Palmisano (NMC) to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, 'Spent Fuel Storage' and 4.3, 'Fuel Storage' (TAC Nos. MC5811 and MC5812)," dated January 5, 2005, ADAMS Accession No. ML060060056.

5. Title 10 Code of Federal Regulations, Part 50 Appendix A, General Design Criterion 62, "Prevention of criticality in fuel storage and handling."

6. WCAP-14416-NP, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Rev. 1, November 1996.

7. Letter from B.A. Wetzel (NRC) to R.O. Anderson (Northern States Power Company), "Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 - Issuance of Amendments Re: Credit for Soluble Boron in Spent Fuel Pool Criticality Analyses (TAC Nos. M93072 and M93073)," dated June 12, 1997.

8. 10 CFR 70.24, "Criticality accident requirements."

9. Letter from E.C. Ward (Northern States Power Company) to P.A. Morris (NRC), "Prairie Island Nuclear Generating Plant E-6197, Docket Nos. 50-282 and 50-306, Amendment No. 12," dated November 19, 1971.

10. NSAL 00-015, "Axial Burnup Shape Reactivity Bias," Westinghouse Electric Company, November 2, 2000.

11. Letter from S. Dembek (NRC) to H.A. Sepp (Westinghouse), "Non-Conservatisms in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology," dated July 27, 2001, ADAMS Accession No. ML012080337.

12. NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4, April 1996.

13. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981.

14. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998.

15. “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” NUREG/CR-0200; distributed by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1998.

16. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

17. S.R. Bierman and E.D. Clayton, “Criticality Experiments with Subcritical Clusters of 2.35 Wt% 235U Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6,” NUREG/CR-1547, PNL-3314, July 1980.

18. Letter from R.B. Ennis (NRC) to J.A. Price (Dominion Nuclear Connecticut, Inc.), “Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Spent Fuel Pool Requirements (TAC No. MB3386),” dated April 1, 2003, ADAMS Accession No. ML030910485.

19. Letter from G.S. Vissing (NRC) to R.C. Mecredy (Rochester Gas and Electric Corporation), “R.E. Ginna Nuclear Power Plant - Amendment Re: Revision to the Storage Configuration Requirements Within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron (TAC No. MA8443),” dated December 7, 2000, ADAMS Accession No. ML003761578.

20. Letter from B. Benney (NRC) to G.M. Rueger (Pacific Gas and Electric Company), “Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis (TAC Nos. MB2982 and MB2984), “dated September 25, 2002, ADAMS Accession No. ML022610080.

21. DOE Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages,” DOE/RW-0472 Rev. 2, September 1998.

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