June 28, 2007

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: INCORPORATE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS USING ASTRUM (TAC NOS. MD2567 AND MD2568)

Dear Mr. Palmisano:

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. DPR-42 and Amendment No. 169 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 the Technical Specifications (TSs) in response to your application dated July 6, 2006, as supplemented by letters dated September 15 and December 26, 2006.

The amendments incorporate new Large-Break Loss-of-Coolant Accident (LBLOCA) analyses using the realistic LBLOCA methodology in the Nuclear Regulatory Commission approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology using Automated Statistical Treatment of Uncertainty Method (ASTRUM)." The amendments revise TS 5.6.5.b to include the reference to WCAP-16009-P-A.

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. 179 to DPR-42
2. Amendment No. 169 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

June 28, 2007

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: INCORPORATE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS USING ASTRUM (TAC NOS. MD2567 AND MD2568)

Dear Mr. Palmisano:

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. DPR-42 and Amendment No. 169 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 the Technical Specifications (TSs) in response to your application dated July 6, 2006, as supplemented by letters dated September 15 and December 26, 2006.

The amendments incorporate new Large-Break Loss-of-Coolant Accident (LBLOCA) analyses using the realistic LBLOCA methodology in the Nuclear Regulatory Commission approved WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology using Automated Statistical Treatment of Uncertainty Method (ASTRUM)." The amendments revise TS 5.6.5.b to include the reference to WCAP-16009-P-A.

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.179 to DPR-42
2. Amendment No.169 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

PUBLIC	LPL3-1 r/f	RidsNrrDorlLpl3-1	RidsNrrPMMChawla
RidsNrrLATHarris	RidsOGCRp	RidsAcrsAcnwMailCenter	RidsNrrDirsItsb
G. Hill, OIS	RidsRgn3MailCenter	**TS:ML071800155**	RidsNrrDorlDpr

Adams Accession: **ML071230789**, **Pkg:ML071800293** *per Memo dated January 17, 2007

OFFICE	LPL3-1	LPL3-1/PM	LPL3-1/LA	DSS/SPWB	OGC	LPL3-1/(A)BC
NAME	JPoole	MChawla	THarris	JNakoski*	JRund	TTate PM for
DATE	06/28/07	06/28/07	06/28/07	01/17/07	05/21/07	06/28/07

OFFICIAL RECORD COPY

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 179
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 July 6, 2006, as supplemented by letters dated September 15 and December 26, 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. 179, 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 the date of its issuance and shall be implemented with the next fuel cycle (Unit 1 Cycle 25) commencing following the winter 2008 refueling.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Patrick Milano for/

Travis L. Tate, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: June 28, 2007

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 July 6, 2006, as supplemented by letters dated September 15 and December 26, 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. 169, 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 the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Patrick Milano for/

Travis L. Tate, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: June 28, 2007

ATTACHMENT TO LICENSE AMENDMENT NOS. 179 AND 169

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Facility Operating License No. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by a marginal line.

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
5.0-36	5.0-36

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NMC 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, NMC 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, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purpose of volume reduction and decontamination.

C. This amended 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) <u>Maximum Power Level</u>

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No.179, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) <u>Physical Protection</u>

NMC 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 Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NMC 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, NMC to transfer byproduct materials from other job sites owned by Northern States Power Company for the purposes of volume reduction and decontamination.

C. This amended 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

NMC is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.169, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NMC 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 Safeguards Information protected under 10 CFR 73.21, is entitled: “Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program,” Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 169 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

On July 6, 2006, Nuclear Management Company, LLC (NMC, or licensee) submitted a license amendment request (LAR) [Agencywide Documents Access and Management System (ADAMS), Accession No. ML061880026] to apply the Nuclear Regulatory Commission (NRC)-approved Westinghouse best-estimate (BE) large-break loss-of coolant accident (LBLOCA) methodology, as described in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005 (ADAMS Accession Nos. ML050910159/61(NP) and ML050910162/63(P)), to its Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP 1 and 2). The licensee also requested a license amendment to include the ASTRUM LBLOCA methodology in the core operating limits reports (COLRs) for both plants. In response to the staff's request for additional information (RAI), the licensee supplemented its requests in letters dated September 15, 2006 (ADAMS Accession No. ML062610088), and December 26, 2006 (ADAMS Accession No. ML063600313). The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 12, 2006 (71 FR 53718).

2.0 REGULATORY EVALUATION

The BE LBLOCA analyses were performed to demonstrate that the emergency core cooling system (ECCS) design would provide sufficient ECCS flow to transfer the heat from the reactor core following a LBLOCA and at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than the amounts that would compromise cladding ductility and result in excessive hydrogen generation.

The NRC staff reviewed the analyses to assure that the PINGP 1 and 2 reflect suitable redundancy in components and features; and that suitable interconnections, leak detection, isolation, and containment capabilities are available such that the safety functions could be accomplished. The analyses assumed a single-failure for LBLOCAs and considered the availability of only onsite or offsite electric power (i.e., assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available), consistent with the requirements of General Design Criterion 35 (GDC 35) of Appendix A to Part 50 of Title 10 of the *Code of Federal Regulations*

(10 CFR). The NRC staff used the acceptance criteria for ECCS performance provided in Section 50.46 of 10 CFR Part 50 (10 CFR 50.46), in assessing the acceptability of the Westinghouse ASTRUM methodology for PINGP 1 and 2.

In its assessment of the acceptability of the methodology for PINGP 1and 2, the NRC staff also reviewed the limitations and conditions stated in its safety evaluation (SE) supporting general approval of the Westinghouse ASTRUM methodology and the range of parameters described in the ASTRUM topical report.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's demonstration evaluations of the ECCS performance analyses, conducted in accordance with the ASTRUM methodology, for operation of PINGP 1 and 2 at the currently licensed core power of 1650 MWt (the analyses were conducted at the rated power of 1650 MWt plus 2 percent measurement uncertainty or 1683 MWt). These specific analyses were performed to demonstrate the suitability of the ASTRUM methodology for application at PINGP 1 and 2. Upon approval, the specific analyses would be acceptable and specifically applicable to PINGP 1 and 2, when operated with the fuel(s) identified in Table 1 of this safety evaluation. For PINGP 1 and 2, the BE LBLOCA analyses were conducted assuming that the plants use cores containing Vantage$^+$ (Zirlo - clad fuel) assemblies.

In its application, the licensee provided the results for the PINGP 1 and 2 BE LBLOCA analyses that were performed in accordance with the ASTRUM methodology and assuming each operating at rated power of 1650 MWt (plus 2 percent measurement uncertainty or 1683 MWt). The licensee's results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 1
LARGE-BREAK LOCA ANALYSIS RESULTS - PINGP Units 1 and 2

Parameter	Unit 1 ASTRUM Vantage$^+$ Results	Unit 2 ASTRUM Vantage$^+$ Results	10 CFR 50.46 Limits
Limiting Break Size/Location	DEG/PD	DEG/PD	N/A
Cladding Material	Zirlo	Zirlo	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	1546°F	1594°F	2200°F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	0.5%	0.2%	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	> 0.01 %	> 0.01%	1.0% (10 CFR 50.46(b)(3))

DEG/PD is a double-ended guillotine break at the pump discharge.

In its analyses, the licensee also addressed the concern that the Vantage$^+$ fuel cladding may have pre-existing oxidation that must be considered in its LOCA analyses. In the supplemental letter dated September 15, 2006, responding to an NRC staff RAI, the licensee indicated that it considered whether the fuel cladding has both pre-existing oxidation and oxidation resulting from the LOCA (pre- and post-LOCA oxidation both on the inside and the outside cladding surfaces). In the supplemental letter dated December 26, 2006, the licensee noted that both PINGP units will be fueled only with Vantage+ fuel (with Zirlo cladding) and clarified that the optimized fuel assembly (OFA) fuel mentioned in its September 15, 2006, letter, referred to "PINGP Optimized Fuel Assembly (OFA) fuel." The "OFA" is an assembly design (geometric, structural, etc.) that was customized for PINGP 1 and 2 to enhance the performance of the fuel (in this case, 14x14, 0.400" outside rod diameter Vantage+ fuel). The term "Vantage+ fuel" is also commonly used to designate fuel assemblies with Zirlo-clad fuel.

In the September 15, 2006, letter, the licensee indicated that the calculated pre-LOCA cladding oxidation was factored into the licensee's BE LBLOCA analyses for the Zirlo clad fuel, consistent with the Westinghouse ASTRUM methodology. Both plants will be operated as governed by the Westinghouse-recommended program to limit operational duty within fuel duty limits, even during a fuel pin's final cycle in the core, such that the sum of the calculated pre- and post-LOCA oxidation will be sufficiently small so that the total local oxidation will remain less than the 17 percent acceptance criterion of 10 CFR 50.46(b)(2) as noted above.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA, and therefore it does not need to be addressed when determining whether the calculated total core-wide oxidation meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

As discussed previously, NMC had Westinghouse conduct BE LBLOCA analyses for PINGP 1 and 2 operating at about 102 percent of the current licensed power level of 1650 MWt using an NRC-approved Westinghouse methodology (ASTRUM). The NRC staff concluded that the results of these analyses demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 1650 MWt. Meeting these criteria provides reasonable assurance that, at the current licensed power level, the PINGP 1 and 2 cores will be amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of PINGP 1 and 2 to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) is unaffected by this amendment, and will be addressed, if needed, in a future NRC safety evaluation.

In its September 15, 2006, letter, the licensee stated, "NMC and its vendor (Westinghouse) have ongoing processes that assure that the ranges and values of the input parameters for the Prairie Island Nuclear Generating Plant (PINGP) Large Break Loss of Coolant Analyses conservatively bound the ranges and values of the as-operated PINGP Unit 1 parameters. Likewise, NMC and its vendor (Westinghouse) have ongoing processes that assure that the ranges and values of the input parameters for the PINGP, Unit 2 LBLOCA analyses conservatively bound the ranges and values of the as-operated PINGP Unit 2 parameters." The NRC staff finds that this statement, along with the generic acceptance of ASTRUM, provides assurance that ASTRUM and its LBLOCA analyses apply to PINGP 1and 2, respectively, operated at their current licensed power levels.

4.0 PINGP 1&2 TECHNICAL SPECIFICATIONS COLR REFERENCES AND BASES CHANGES

In support of the LAR, the licensee proposed to make changes to TS 5.6.5, “Core Operating Limits Report,” for each plant to reflect use of a new LBLOCA analysis methodology to perform LBLOCA analyses in support of PINGP 1 and 2 operation. The licensee also provided new TS Bases sections to describe the rationales for the new TS. The NRC staff reviewed these TS provisions, assessed them for consistency against NUREG-1431,“Standard Technical Specifications, Westinghouse Plants,” Revision 3, TS 5.6.5 (quoted on Page 14 of 59 in the LAR) as stated below, and found their content acceptable.

TS 5.6.5 COLR:

27. WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)”.

This methodology was found to apply to all Westinghouse and Combustion Engineering pressurized-water reactor (PWR) designs in the NRC generic safety evaluation of the ASTRUM methodology. Therefore, ASTRUM is acceptable for application to PINGP 1 and 2, which are PWRs of Westinghouse design, and for inclusion in PINGP 1 and 2 TS for each plant. The above listed TS 5.6.5 Reference 27 was presented in the licensee’s submittal as a TS addition. This reference does not include the WCAP-16009-P-A revision number (i.e., “0"); nor does it include the date of approval for the methodology. The licensee will list the topical report, including the latest revision number, and date of approval in the COLR for each of the PINGP units consistent with guidance provided in NUREG -1431.

The NRC staff finds that ASTRUM is applicable to PINGP 1 and 2 and that the limitations and conditions of the NRC’s SE approving ASTRUM were satisfied. Thus, the NRC staff concludes that the proposed addition of WCAP-16009-P-A to TS 5.6.5 is acceptable.

5.0 SUMMARY

Based on its review as discussed above, the NRC staff concluded that the Westinghouse ASTRUM methodology, as described in WCAP-16009-P-A, is acceptable for use for PINGP 1 and 2 in demonstrating compliance with the requirements of 10 CFR 50.46(b). The NRC staff’s conclusion is based on the staff’s verification that the PINGP design is among the designs for which ASTRUM application was approved.

The NRC staff’s review of the acceptability of the ASTRUM methodology for PINGP 1 and 2 focused on assuring that the licensee and its vendor have a processes to assure that specific input parameters or bounding values and ranges (where appropriate) are used to conduct the PINGP 1 and 2 LBLOCA analyses, that the analyses will be conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM methodology, and that the results will satisfy the requirements of 10 CFR 50.46(b) for PINGP 1 and 2.

The NRC staff finds the Westinghouse ASTRUM BE LBLOCA analysis methodology acceptable for application to PINGP 1 and 2 and for inclusion in the PINGP 1 and 2 TS 5.6.5 and COLRs. As discussed above, the staff also finds the specific LBLOCA analyses acceptable that were performed with the ASTRUM methodology for PINGP 1 and 2 operating at powers up to their licensed power level of 1650 MWt.

6.0 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.

7.0 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 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 53718). 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.

8.0 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.

Principal Contributor: F. Orr

Date: June 28, 2007

cc:

Jonathan Rogoff, Esquire
Vice President, Counsel & Secretary
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Manager, Regulatory Affairs
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

Manager - Environmental Protection Division
Minnesota Attorney General's Office
445 Minnesota St., Suite 900
St. Paul, MN 55101-2127

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
1719 Wakonade Drive East
Welch, MN 55089-9642

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4351

Administrator
Goodhue County Courthouse
Box 408
Red Wing, MN 55066-0408

Commissioner
Minnesota Department of Commerce
85 7th Place East, Suite 500
St. Paul, MN 55101-2198

Tribal Council
Prairie Island Indian Community
ATTN: Environmental Department
5636 Sturgeon Lake Road
Welch, MN 55089

Nuclear Asset Manager
Xcel Energy, Inc.
414 Nicollet Mall, R.S. 8
Minneapolis, MN 55401

Michael B. Sellman
President and Chief Executive Officer
Nuclear Management Company, LLC
700 First Street
Hudson, MI 54016

Douglas E. Cooper
Senior Vice President and Chief
  Nuclear Officer
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016