August 20, 2004

Mr. Joseph M. Solymossy
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, UNIT 1 - ISSUANCE OF AMENDMENT RE: POST-MODIFICATION TESTING OF PRESSURE CONTAINMENT BOUNDARY (TAC NO. MC0605)

Dear Mr. Solymossy:

The Commission has issued the enclosed Amendment No. 165 to Facility Operating License No. DPR-42 for the Prairie Island Nuclear Generating Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSS) in response to your application dated August 27, 2003, as supplemented December 16, 2003, March 22, 2004, and July 19, 2004.

The amendment revises TS 5.5.14 to allow the licensee to perform post-modification testing of the containment pressure boundary following steam generator replacement in accordance with the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section XI, instead of 10 CFR Part 50, Appendix J, Option B. The steam generator replacement is scheduled for fall 2004.

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, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-282

Enclosures: 1. Amendment No. 165 to DPR-42
2. Safety Evaluation

cc w/encl: See next page

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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. 165
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 August 27, 2003, as supplemented December 16, 2003, March 22, 2004, and July 19, 2004, 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. 165, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 20, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 165

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
5.0-27	5.0-27
5.0-28	5.0-28
5.0-29	5.0-29

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-42

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

DOCKET NO. 50-282

1.0 INTRODUCTION

By application dated August 27, 2003, as supplemented December 16, 2003, March 22, 2004, and July 19, 2004, the Nuclear Management Company, LLC (the licensee), requested a change to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant, Unit 1. The amendment would revise TS 5.5.14 to allow the licensee to perform post-modification testing of the containment pressure boundary following steam generator replacement in accordance with the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section XI, instead of 10 CFR Part 50, Appendix J, Option B. The steam generator replacement is scheduled for fall 2004. Specifically, the proposed change would revise TS 5.5.14.a, by adding the following:

> "as modified by the following exception:
>
> 1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement."

The March 22 and July 19, 2004, supplemental letters provided clarifying information that was within the scope of the original amendment request and did not change the Nuclear Regulatory Commission (NRC) staff's initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The licensee intends to replace the existing Unit 1 steam generators during the fall 2004 refueling outage. The steam generator replacement would affect only the closed piping inside containment. The containment structure and the containment liner would not be affected. However, the steam generator shell and the inside-containment portions of the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, steam generator water level instrument lines, and the steam generator tubes and tube sheets are part of the primary reactor containment.

TS 5.5.14 requires that a program be established to implement the leakage testing of the containment, as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(o), and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory

Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program.” RG 1.163 endorses Nuclear Energy Institute (NEI) 94-01, Revision 0, for methods acceptable to comply with the requirements of Option B. NEI 94-01, Revision 0, Section 9.2.4, “Containment Repairs and Modifications,” states:

> Repairs and modifications that affect containment integrity require leakage rate testing (Type A testing or local leakage rate testing) prior to returning the containment to operation.

Local leakage rate testing is not practical in this case. As the next Type A test for Unit 1 is not scheduled to occur until 2007, the licensee would have to perform a Type A test in 2004 especially for the steam generator replacement, unless it obtains an exception to the requirement.

The area of the containment boundary that will be affected by the replacement is also part of the pressure boundary of an American Society of Mechanical Engineers (ASME) Class 2 component/piping system and, as such, the replacement of the steam generators is subject to the repair and replacement requirements of ASME Code, Section XI. The acceptance criteria for ASME Code, Section XI, system pressure testing of welded joints is "zero leakage."

The ASME Code, Section XI, pressure test, unlike the Type A test, does not require the leakage rate to be quantified. The acceptance criterion for the proposed test is no visual through-wall leakage; therefore, there is no need to quantify the leakage rate. This acceptance criterion is more conservative than the Appendix J Type A test which allows some leakage. However, the ASME Code, Section XI, pressure testing can be done without removing the insulation over the piping. This allows some uncertainty in the leakage rate.

The ASME Code, Section XI, requires non-destructive examination (NDE) and visual examination of welds and system leakage testing. If any through-wall leakage is detected from the welds, the leakage is required to be repaired before plant service continues.

The licensee proposes to perform an ASME Code, Section XI, pressure test to satisfy the intent of the Appendix J requirements rather than performing a Type A test.

The NRC staff has approved similar requests for several licensees (see, for example, References 1, 2, and 3).

3.0 TECHNICAL EVALUATION

The Unit 1 steam generator replacement will consist of the following operations, as described in the licensee’s application:

- cutting and removing the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, and steam generator water level instrument lines from the steam generators
- cutting and removing the upper assemblies of the steam generators
- cutting the reactor coolant piping and removing the steam generator lower assemblies
- installing the new steam generator lower assemblies and re-welding the reactor coolant piping
- installing the new steam generator upper assemblies on the new lower assemblies

- re-installing and re-welding the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, and steam generator water level instrument lines

The steam generator replacement would only affect the closed piping inside containment. The containment structure and the containment liner would not be affected. The new steam generator assemblies and the old steam generator assemblies will transit through the containment equipment hatch. However, the steam generator shell and the inside-containment portions of the main steam lines, main and auxiliary feedwater lines, steam generator blowdown lines, steam generator water level instrument lines, and the steam generator tubes and tube sheets are part of the primary reactor containment, as discussed in the Prairie Island Updated Safety Analysis Report, Section 5.2.2.1.1 and Table 5.2-1.

The licensee proposes to eliminate the requirement to perform post-modification integrated leakage rate (Type A) testing following steam generator replacement, which is not consistent with the current TS 5.5.14. The licensee has proposed adding the following to TS 5.5.14.a:

> "as modified by the following exception:
>
> 1. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement."

The licensee stated that, because the affected area of the primary containment boundary is also part of the pressure boundary of an ASME Class 2 component/piping system, the steam generator replacement is also subject to the repair and replacement requirements of the ASME Code, Section XI. The ASME Code, Section XI, testing provides an alternative method, which allows testing of only the modified portions of the containment barrier (steam generator shell and associated closed piping and components) instead of the more comprehensive Type A testing, which would be performed on the entire containment barrier. In addition, the Section XI testing would accomplish leakage testing in Mode 3, in contrast to the current requirement to complete testing prior to entering Mode 4 since it requires testing to be performed at approximately normal reactor operating temperature and pressure. The NRC staff has concluded that entering Modes 3 and 4 prior to quantifying the containment leakage rate is acceptable because the likelihood of a containment release is insignificant due to the configuration of the primary and secondary systems in Modes 3 and 4. In order to have a release through the modified closed piping systems, there would have to be a loss-of-coolant accident concurrent with a through-wall leak, with enough pressure in the containment to overcome main steam system pressure.

The ASME Code, Section XI, also requires NDE and visual examination of welds and system leakage testing. NDE of the welds (ultrasonic or radiographic testing) provides assurance that the joints are free of flaws that could result in significant leakage. This NDE provides confidence to pressurize the secondary side of the steam generators and demonstrate leak-tight integrity with the unit in Mode 3 under no-load conditions.

The NRC staff has reviewed the ASME Code, Section XI, requirements and determined that the ASME Code, Section XI, surface examination, volumetric examination, and system pressure testing requirements are more focused than the Type A testing requirements of Appendix J (which are currently required by TS 5.5.14). The objective of the Type A test is to ensure the leak-tight integrity of the containment area affected by the modification. The ASME Code, Section XI, inspection and testing requirements fulfill the intent of the requirements of Appendix J and the provisions of NEI 94-01, Section 9.2.4. In addition, the test pressure for the system pressure test will be at least 20 times that of a Type A test. Therefore, the NRC staff finds the basis for the licensee's proposed exception from performing a post-modification Type A test following steam generator replacement to be acceptable.

Additionally, as a result of the proposed change to TS 5.5.14.a, the licensee proposes repaginating TS Pages 5.0-27, 5.0-28, and 5.0-29. The NRC staff finds the repagination of these TS pages acceptable.

3.1 Summary

Based on the above evaluation, the NRC staff concludes that post-modification testing of the containment pressure boundary may be performed in compliance with Section XI of the ASME Code rather than in compliance with the requirements of 10 CFR Part 50, Appendix J, Option B. Therefore, the NRC staff finds the proposed change to TS 5.5.14 acceptable.

4.0 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.

5.0 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 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 (69 FR 2744). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from David B. Matthews, NRC, to Mr. J. P. O'Hanlon, Virginia Electric and Power Company, "Issuance of Exemptions from the Requirements of 10 CFR Part 50, Appendix J Associated with Type A Testing Requirements - North Anna Power Station, Unit No. 2 (NA-2) (TAC No. M91778)," March 29, 1995 (ADAMS Accession No. ML013520485).

2. Letter from Allen G. Hansen, NRC, to Robert E. Link, Wisconsin Electric Power Company, "Amendment Nos. 169 and 173 to Facility Operating License Nos. DPR-24 and DPR-27 - Point Beach Nuclear Plant, Unit Nos. 1 and 2 (TAC Nos. M95668 and M95669)," October 9, 1996 (ADAMS Accession No. ML021970405).

3. Letter from Donna Skay, NRC, to P. E. Katz, Calvert Cliffs Nuclear Power Plant, Inc., "Calvert Cliffs Nuclear Power Plant, Unit No. 2 - Amendment Re: Exception to Post-Modification Integrated Leakage Rate Testing (TAC No. MB3444)," June 27, 2002 (ADAMS Accession No. ML021490258).

Principal Contributor: J. Pulsipher

Date: August 20, 2004

Prairie Island Nuclear Generating Plant,
  Units 1 and 2

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