August 19, 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, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MC2439 AND MC2440)

Dear Mr. Solymossy:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 164 to Facility Operating License No. DPR-42 and Amendment No. 155 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 licensing basis in response to your application dated March 25, 2004, as supplemented June 2, 2004.

The amendments approve a change to the licensing basis to allow the use of the methods described in Framatome-ANP Topical Report BAW-10169-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," dated October 1989, for calculating the mass and energy release rates resulting from a postulated main steamline break accident for input to containment analyses. These methods utilize the RELAP5/MOD2-B&W code approved by the NRC staff in a safety evaluation report dated March 14, 1995.

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 Nos. 50-282 and 50-306

Enclosures: 1. Amendment No. 164 to DPR-42
2. Amendment No. 155 to DPR-60
3. Safety Evaluation

cc w/encls: 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. 164
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 March 25, 2004, as supplemented June 2, 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, by Amendment No. 164 , Facility Operating License No. DPR-42 is hereby amended to authorize revision to the Updated Safety Analysis Report (USAR), as set forth in the license amendment application dated March 25, 2004, as supplemented June 2, 2004. The licensee shall update the USAR to incorporate the change in the licensing basis which allows the use of the methods described in Framatome-ANP Topical Report BAW-10169-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," dated October 1989, for calculating the mass and energy release rates resulting from a postulated main steamline break accident for input to containment analyses, as set forth in the application dated March 25, 2004, as supplemented June 2, 2004, and the NRC staff's safety evaluation attached to this amendment. The licensee shall submit the revised description authorized by this amendment with the next update of the USAR.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days. The USAR changes shall be implemented in the next periodic update of the USAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: August 19, 2004

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
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 March 25, 2004, as supplemented June 2, 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, by Amendment No. 155, Facility Operating License No. DPR-60 is hereby amended to authorize revision to the Updated Safety Analysis Report (USAR), as set forth in the license amendment application dated March 25, 2004, as supplemented June 2, 2004. The licensee shall update the USAR to incorporate the change in the licensing basis which allows the use of the methods described in Framatome-ANP Topical Report BAW-10169-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," dated October 1989, for calculating the mass and energy release rates resulting from a postulated main steamline break accident for input to containment analyses, as set forth in the application dated March 25, 2004, as supplemented June 2, 2004, and the NRC staff's safety evaluation attached to this amendment. The licensee shall submit the revised description authorized by this amendment with the next update of the USAR.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days. The UFSAR changes shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: August 19, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 155 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 application dated March 25, 2004, the Nuclear Management Company LLC, (the licensee), for the Prairie Island Nuclear Generating Plant (Prairie Island), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.90, requested an amendment to the operating license for Prairie Island Units 1 and 2. The proposed amendment would allow the use of the methods described in Framatome-ANP Topical Report BAW-10169-A, “RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants,” dated October 1989, for calculating the mass and energy release rates as a result of a postulated main steam line break accident for input to containment analyses. These methods utilize the RELAP5/MOD2-B&W code approved by the staff in a safety evaluation report dated March 14, 1995.

A request for additional information was sent to the licensee via e-mail (ADAMS Accession No. ML042010318) on May 13, 2004. A follow-up telephone conference was held between the Nuclear Regulatory Commission (NRC) staff and the licensee on May 19, 2004, to discuss this request. In response to the NRC staff’s questions, the licensee provided additional information in a letter dated June 2, 2004. The supplemental letter 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.

The license amendments were prompted by calculations to support the Prairie Island Unit 1 Replacement Steam Generator Project. The licensee states that the analysis to predict the mass and energy release rates following a postulated main steamline break accident used the methods described in topical report BAW-10169-A. However, the licensee was not able to conclude that prior NRC acceptance of the main steamline break analysis methods described in BAW-10169-A included the calculation of mass and energy release rates for containment analyses. The licensee, therefore, submitted the March 25, 2004, license amendment request, which is the subject of this review.

Prairie Island Units 1 and 2 are two-loop pressurized-water reactors (PWRs) designed by Westinghouse Electric Corporation. The licensed power levels are 1650 MWt for both Unit 1 and Unit 2. The containments for both units are cylindrical steel shells with hemispherical domes.

2.0 REGULATORY EVALUATION

The Prairie Island units were not licensed to the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50. The Prairie Island Updated Safety Analysis Report (USAR) (Section 1.2, "Principal Design Criteria") states that Prairie Island was designed and constructed to comply with the licensee's "understanding of the intent" of the Atomic Energy Commission (AEC) General Design Criteria (GDC) as proposed on July 10, 1967.

However, the AEC safety evaluation report for Prairie Island stated that the AEC staff assessed the plant, as described in the USAR, against Appendix A design criteria and "...are satisfied that the plant design generally conforms to the intent of these criteria."

GDC 16 of Appendix A to 10 CFR Part 50 requires that the reactor containment and associated systems shall establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and shall assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The licensee is not proposing any material changes to the operating license, technical specifications or the technical specification bases. The licensee is proposing a change to the analysis methods used to demonstrate compliance with the intent of GDC 16 in that the analytical methods used to predict the mass and energy release to the containment following a postulated main steam line break accident are revised.

Standard Review Plan (SRP), NUREG-0800, Revision 1, Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," provides guidance for the review of these calculations. Although Prairie Island was also not licensed to the SRP, this guidance was followed in performing this review.

The NRC staff has previously approved the methods described in BAW-10169-A for the reactor core response. Therefore, this review concentrates on ensuring that the same methods comply with the intent of GDC 16 and SRP Section 6.2.1.4, which pertain to the containment analysis.

3.0 TECHNICAL EVALUATION

The RELAP5/MOD2-BAW computer code is the Framatome-ANP version of the RELAP5/MOD2 computer code developed by Idaho National Engineering Laboratory to calculate the best-estimate response to a wide variety of PWR system transients. This code models the primary and secondary systems, the core neutronics, and system controls. The response of a PWR to both loss-of-coolant accidents (LOCAs) and non-LOCAs, such as the main steamline break, can be calculated with RELAP5/MOD2-BAW. The NRC-approved topical report describing this computer code is BAW-10164-A.[1] The NRC-approved methodology for utilizing this computer code for non-LOCA postulated design-basis events is

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[1] BAW-10164-A, "RELAP5/MOD2-B&W An Advanced Computer Program for Light-Water Reactor LOCA and non-LOCA Transient Analysis," Rev. 3, October 1996.

described in BAW-10169. As discussed previously, the licensee has concluded that NRC approval of BAW-10169 did not include the calculation of the mass and energy released inside the containment due to a postulated main steamline break accident as input to containment analyses. NRC approval did include the calculation of the mass and energy release for determining the reactor core response to a postulated main steamline break accident.

Therefore, the NRC staff review of this proposed license amendment is limited to the calculation of the mass and energy released inside the containment due to a postulated main steamline break accident. This calculation is necessary to develop the mass and energy input to the calculation which demonstrates that the containment pressure and temperature due to the postulated main steamline break accident remains below the containment design values. The proposed changes to the Prairie Island USAR Sections 14.3.1, "Calculational Methods and Input Parameters," and 14.5.5.3.1, "Containment Response," are given in Exhibit B to the licensee's March 25, 2004, application.

The CONTEMPT-LT/028 computer code[2] is used to determine the temperature and pressure response of the containment. This computer code was developed for the NRC and has been widely used by the nuclear industry.

In both the reactor core response calculation and the containment pressure calculation, the mass and energy release is overestimated in order to ensure a conservative result. Thus, if the mass and energy calculation for the core response is conservative, the mass and energy calculation for the containment response should also be conservative. However, in order to provide additional assurance that the mass and energy release calculated for the reactor core response is suitable for the containment response, the NRC staff reviewed several input and modeling assumptions that are known to have a significant effect on the containment response. These inputs are:

- power level
- break size
- entrainment of liquid in the steam break flow
- loss of offsite power assumption
- steam generator liquid inventory at the beginning of the accident
- modeling of the steam system
- modeling of the feedwater system
- single failure assumption
- break flow model
- primary-to-secondary heat transfer
- distribution of reactivity due to difference in the faulted loop and the non-faulted loop temperatures

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[2] Don W. Hargroves, "CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure - Temperature Response to a Loss-of-Coolant Accident," NUREG/CR-0255, March 1979.

3.1 Power Level

The licensee's March 25, 2004, application states that:

> The sensitivity studies [to determine the most limiting power level] will cover a range of power levels from 0% power up to and including 102% of licensed full power.

This is important since it is not possible to predict the most conservative power level without sensitivity studies. The maximum power level of 102 percent includes measurement uncertainties in determining the power and is consistent with Regulatory Guide 1.49[3] and the Prairie Island USAR. The NRC staff finds the licensee's approach acceptable.

3.2 Break Size

The licensee's March 25, 2004, application states that:

> The sensitivity studies [to determine the most conservative break size] will cover break sizes from 0.6 ft^2 to the largest possible double ended rupture.

This is important since it is not possible to predict the most conservative break size without sensitivity studies. The licensee states that the minimum break size of 0.6 ft^2 was selected based on Framatome-ANP experience. Transient results showed a trend toward increasing containment pressure response as the break size was lowered to 0.6 ft^2. As a result, additional breaks of 0.4 ft^2 were analyzed. The limiting break size for the containment response was 0.6 ft^2. The NRC staff finds the licensee's approach acceptable.

3.3 Entrainment of Liquid in The Steam Break Flow

The licensee's June 2, 2004, supplemental letter states that liquid swell within the steam generator and the resulting entrainment in the break flow are explicitly calculated for each case. BAW-10169-A provides comparisons of results calculated with RELAP5/MOD2 and USAR values for a Westinghouse plant calculated with previously-approved Westinghouse methods. This comparison shows good agreement. Therefore, the NRC staff finds the licensee's approach acceptable.

3.4 Loss of Offsite Power

The licensee states that loss of offsite power would result in the immediate loss of the reactor coolant pumps and the main feedwater pumps. Loss of the reactor coolant pumps would reduce the energy transferred from the reactor coolant system to the steam generators. Loss of the feedwater pumps would preclude additional feedwater flow to the steam generators. Therefore, loss of offsite power would result in a significant reduction in the mass and energy added to the containment. Offsite power is, therefore, assumed to remain available. The NRC staff finds this conservative and acceptable.

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[3] Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1 December 1973.

3.5 Steam Generator Liquid Inventory at the Beginning of the Accident

The licensee states that the liquid inventory at any given power level is maximized by assuming a maximum initial water level that bounds all power levels. Thus, the water level stays constant and the mass varies with the temperature of the water in the steam generator. The NRC staff finds this conservative and acceptable.

3.6 Modeling of the Steam System

The licensee states that the modeling of the steam system is consistent with the requirements of BAW-10169-A and is generated using Prairie Island specific data. For those breaks that are between the non-return check valves and the faulted steam generator, the analysis will include steam flow from the intact steam generator through the pressure balancing line and out the break until the non-return check valve closes on the faulted loop. The non-return check valve is assumed to close one second after forward flow ceases. The licensee states that this is consistent with the Prairie Island licensing basis. These assumptions are conservative and, therefore, the NRC staff finds the licensee's approach acceptable.

3.7 Modeling of the Feedwater System

The modeling of the feedwater system is important since it determines the amount of water added to the faulted steam generator and therefore the amount of mass discharged to the containment. The licensee states that the feedwater system model is consistent with the current Prairie Island licensing basis. The main feedwater pumps continue to run at full capacity until a feedwater isolation signal is received. At this time the pumps are tripped and coast down. The main feedwater regulator and bypass valves are opened at a rate faster than the non-emergency opening rate and close at a rate slower than the emergency rate after receipt of the feedwater isolation signal. The unisolated portions of the feedwater line empty into the faulted steam generator.

The licensee states that the auxiliary feedwater system is modeled such that the maximum amount of water is injected into the faulted steam generator until the auxiliary feedwater pumps are tripped by the automatic runout protection system.

These assumptions are conservative and, therefore, the NRC staff finds the licensee's approach acceptable.

3.8 Single-Failure Assumption

The licensee states that the single-failure sensitivity studies include assuming the failure of a safeguards train of equipment, the failure of a feedwater regulator valve to the faulted steam generator to close, and the failure of the main feedwater pump to trip. In comparing this list with the single-failures discussed in the Prairie Island USAR, the NRC staff noted that the failure of auxiliary feedwater runout protection was not addressed in the licensee's March 25, 2004, application. In its June 2, 2004, supplemental letter the licensee responded that previous analyses performed by NMC showed that failure of the auxiliary feedwater pump automatic runout protection was not a limiting failure. Therefore, the NRC staff considers the licensee's treatment of the single-failure criterion to be acceptable since it considers the significant single-failures.

3.9 Break Flow Model

The break flow is modeled using the Moody critical flow correlation.[4] This correlation has been found to be conservative for critical (maximum) flow from large pipes and is, therefore, acceptable for this case.

3.10 Primary-to-secondary Heat Transfer

The licensee states that the primary-to-secondary heat transfer is maximized in the reactor core response calculations. Since the same assumption maximizes energy discharge into the containment, this is conservative and is, therefore, acceptable.

3.11 Distribution of Reactivity Due to the Difference in the Faulted Loop and the Non-faulted Loop Reactor Coolant Temperatures

A steam line break with a negative moderator temperature coefficient has the potential to result in a return to power which adds additional heat to the steam generator fluid discharged to the containment. The assumption previously used in BAW-10169-A was a 50 percent/50 percent faulted/unfaulted loop reactivity weighting. This applied to a four-loop plant with three intact loops and one faulted loop. The combination of modeling three intact loops as one and the 50 percent/50 percent reactivity weighting results in the flow from one faulted loop having three times the reactivity impact as the flow from any of the three intact loops. To maintain the same weighting for a two-loop plant such as Prairie Island, the 50 percent/50 percent weighting corresponds to a weighting of 75 percent/25 percent, faulted/unfaulted. Since the licensee is maintaining the same weighting proportion as previously approved by the NRC, the NRC staff finds this acceptable.

4.0 SUMMARY

The licensee proposes to use the same models and assumptions for the containment response to a main steamline break accident as used for the reactor core response. Since all the input and models that maximize the core response also maximize the containment response, this is acceptable. The NRC staff has reviewed some of the important input and modeling assumptions to assure that they are suitably conservative for the containment response. The NRC staff's conclusion is that they are. Based on this review, the NRC staff finds the licensee's proposal to be acceptable.

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.

[4] F. J. Moody, "Maximum Flow Rate of Single Component, Two-Phase Mixture," American Society of Mechanical Engineers Paper 64-HT-35, 1964.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 (69 FR 22881). 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.

Principal Contributor: R Lobel

Date: August 19, 2004

cc:

Jonathan Rogoff, Esquire
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