March 29, 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 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING STEAM GENERATOR TUBE INSPECTION SUMMARY REPORTS FROM THE SPRING 2002, AND FALL 2004 REFUELING OUTAGES (TAC NO. MC0907)

Dear Mr. Solymossy:

By letters dated March 5, March 19, and May 19, 2002, and letters dated October 24, and November 7, 2003, and January 7, 2004, the Nuclear Management Company, LLC (NMC), submitted steam generator tube inspection summary reports in accordance with the Prairie Island Nuclear Generating Plant Technical Specifications. These inspection reports were from the spring 2002 and fall 2003 refueling outages for Unit 2. The Nuclear Regulatory Commission staff finds that the additional information identified in the enclosure is needed.

A draft of the request for additional information was e-mailed to Mr. J. Kivi (NMC) on February 5, 2004. During a phone call on March 23, 2004, a mutually agreeable response date of June 21, 2004, was established.

Please contact me at (301) 415-4106 if future circumstances should require a change in this response date.

Sincerely,

/RA/

Anthony C. McMurtray, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-306

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
REGARDING STEAM GENERATOR TUBE INSPECTION SUMMARY REPORTS
FROM THE SPRING 2002 AND FALL 2003 REFUELING OUTAGES
PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2
DOCKET NO. 50-306

By letters dated March 5, 2002 (ML020720552), March 19, 2002 (ML020990197), May 31, 2002 (ML021560092), October 24, 2003 (ML033040378), November 7, 2003 (ML033210114), and January 7, 2004 (ML040200107), Nuclear Management Company (NMC), the licensee, submitted the 2002 and 2003 steam generator (SG) tube inspection summary reports for the Prairie Island Nuclear Generating Plant (PINGP) Unit 2. Additional information concerning these inspections was summarized by the Nuclear Regulatory Commission (NRC) staff in letters dated April 12, 2002 (ML021050465) and November 26, 2003 (ML033360758). In order for the NRC staff to complete its review of these reports, responses to the following questions are requested:

1. Please discuss whether the dent signals attributed to the U-bend heat treatment performed in 2000 have changed in magnitude since your 2002 inspection. If so, discuss the implications to tube integrity and your postulated root cause. Please discuss the scope (number of tubes inspected) and results of any rotating probe inspections performed at these dented locations.

Unit 2 Inservice Inspection Summary Report, Interval 3, Period 3
Refueling Outage Dates 2-1-2002 to 3-2-2002
Cycle 21/ 6-7-2000 to 3-2-2002 (May 31, 2002)

1. During the 2002 inspection, several tubes were plugged due to permeability indications (ex. in Table VIII, SG 21, tubes in Row (R)40 Column (C)50, R42 C50 and R34 C55). Discuss whether these indications have changed with time or whether these tubes were plugged based on new criteria implemented during the 2002 inspection.

SG Inspection Results - 15-Day Report (October 24, 2003)

1. Per the PINGP technical specification, Section 5.5.8, the F* distance (not including eddy current test uncertainty) is 1.07 inches and the EF* distance (not including eddy current uncertainty) is 1.67 inches. To apply the F*/EF* repair criteria, at least 1.07 inches (or 1.67 inches, as appropriate) from the bottom of the hardroll transition must be free of degradation. In the report, several tubes (ex. in SG 21, tubes in R2 C13, R1 C15 and R21 C42) are reported with indications that range to what appears to be less than 1 inch from the bottom of the hardroll. Please clarify where the measurements are taken from and the eddy current uncertainty assumed when determining flaw location relative to the bottom of the hardroll transition. Provide the technical basis for the eddy current uncertainty used (or provide a reference to a specific document if previously submitted to the NRC). In addition, please clarify why some indications do not have the “elevation to” field filled out or have a zero in the field (ex. in SG 21 the tube in R3 C3).

ENCLOSURE

2. On page 1 of the report, NMC indicates that several single axial indications and multiple axial indications were no longer detectable (ex. in SG 21, tubes in R3 C3 and R14 C3). You indicated that this may have resulted from the rerolling process. Please discuss whether this reduction in detectability or voltage is observed in the “post installation” inspections (i.e., in the inspections performed in the outage in which the roll repair is made). Also discuss how it is confirmed that these indications have not grown into the F* distance given the “reduced” detectability in this region. Please discuss whether indications which were no longer detectable in one outage were subsequently detected in a future outage. If so, discuss whether these indications have grown (either in length or depth). Discuss your technical basis for leaving these tubes in service (including any tube pull results confirming your ability to detect degradation in these tubes).

3. The cover letter of the report notes that 17 tubes were repaired in 2003, and it appears that new indications are occurring in tubes that have been rerolled and/or that existing indications have grown into the F*/EF* distance. Discuss whether tubes with such indications have adequate tube integrity. The NRC staff notes that the F*/EF* distance assumes that the tube is undegraded in that region. Given that it appears that flaws have been found in this region (following a cycle (or more) of operation), there is a potential that these flaws may affect pullout resistance or leakage integrity. Please provide your technical basis for your conclusions.

4. For tubes with rerolls installed, please discuss whether any indications of denting or any restrictions have been observed between the roll expansions. The NRC staff notes that operating experience with the diode effect has indicated the potential for water to enter a crevice, expand, and subsequently deform a tube or sleeve. If dents/restrictions have been observed, discuss the implications to tube integrity (e.g., address whether the dents/restrictions reduce the resistance to pullout and/or affect leakage integrity of the joint).

Unit 2 Inservice Inspection Summary Report, Interval 3, Period 3
Refueling Outage Dates 9-13-2003 to 10-10-2003
Fuel Cycle 21: 3-3-2002 to 10-10-2003 **(January 7, 2004)**

1. On page 1, footnote 2, of Section 7 of this report, NMC indicated that INR signals at tube support plates that are greater than or equal to 1.5 volts are inspected with a rotating probe. Please clarify what “INR” signals are. Please compare this type of indication to a distorted tube support indication or a non-quantifiable indication at a tube support. Please briefly discuss the basis for the 1.5 volt criteria. Discuss the results of the rotating probe examinations of these signals.

2. Please summarize the results of your 2003 inspection including a discussion of the degradation mechanisms observed, the results of your insitu pressure testing, and the results of your condition monitoring and operational assessment. For example, discuss whether any indications were found in the dents, the results of your rotating probe inspection in the U-bend areas, the results of your plug examination, etc. Include in this discussion, the results of any post-insitu pressure test inspections and your assessment of these results.

3. Regarding indications of wear at the previously removed antivibration bars, please discuss whether there has been any change in the eddy current indications from the time the original antivibration bars were removed to the 2003 inspection (i.e., discuss whether the indications are growing, including indications in the u-bend area).

4. Several indications in the U-bend area that have been left in service appear to be in the free span (e.g., SG 21, Row 11, Column 33). Please clarify the nature of the indications reported in the U-bend area (i.e., are all freespan indications attributable to wear at the “old” anti-vibration bars)?

5. Please clarify the nature of the volumetric indications plugged during the 2003 inspection. (See page 10 of Section 7 of this report)

cc:

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