October 28, 2004

Mr. Joseph M. Solymossy
Site Vice-President
Prairie Island Nuclear Generating Plant
Nuclear Management Company
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT 2 - STEAM GENERATOR TUBE INSPECTION SUMMARY REPORTS FOR 2002 AND 2003 OUTAGES (TAC NO. MC0907)

Dear Mr. Solymossy:

By letters dated March 5, 2002 (ML020720552), March 19, 2002 (ML020990197), May 31, 2002 (ML021560092), October 24, 2003 (ML033040378), November 7, 2003 (ML033210114), January 7, 2004 (ML040200107), June 21, 2004 (ML041740330), and September 3, 2004 (ML042590288), Nuclear Management Company (the licensee) submitted reports summarizing the steam generator tube inspections performed at Prairie Island Nuclear Generating Plant, Unit 2 during refueling outages 21 and 22, which were performed in 2002 and 2003, respectively. Additional information concerning these inspections was summarized by the U.S. Nuclear Regulatory Commission staff in letters dated April 12, 2002 (ML021050465), January 27, 2003 (ML030080581), and November 26, 2003 (ML033360758).

As discussed in the enclosed evaluation, the staff concludes that the licensee provided the information required by their technical specifications. In addition, the staff did not identify any technical issues that warranted follow up action at this time.

If you have any further questions, please contact me at (301) 415-8371.

Sincerely,

/RA/

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

Docket No. 50-306

cc/w encl: See next page

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SAFETY EVALUATION OF 2002 AND 2003 REFUELING OUTAGE

STEAM GENERATOR TUBE INSPECTION REPORTS

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING COMPANY, 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), January 7, 2004 (ML040200107), June 21, 2004 (ML041740330), and September 3, 2004 (ML042590288), Nuclear Management Company (the licensee) submitted reports summarizing the steam generator (SG) tube inspections performed at Prairie Island Nuclear Generating Plant (PINGP), Unit 2, during refueling outages 21 and 22, which were performed in 2002 and 2003, respectively. Additional information concerning these inspections was summarized by the U.S. Nuclear Regulatory Commission staff in letters dated April 12, 2002 (ML021050465), January 27, 2003 (ML030080581), and November 26, 2003 (ML033360758).

The two SGs at PINGP, Unit 2, are Westinghouse model 51 SGs. Each SG contains 3,388 mill annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.875-inch and a nominal wall thickness of 0.050-inch. The tubes were roll expanded into the tubesheet at both ends for approximately 2.75-inch (i.e., they are expanded for only a fraction of the tubesheet thickness and are considered partial depth hard-rolled tubes). The tubes are supported by a number of carbon steel tube support plates. The original anti-vibration bars were removed and replaced. The tubes installed in rows 1 and 2 were subjected to an in-situ thermal stress relief in May 2000. To repair defects, many tubes have been roll expanded into the tubesheet region above the original factory roll expansions. The hot-leg temperature at PINGP, Unit 2, has been approximately 590 degrees Fahrenheit since commencement of initial operation. There are no sleeves installed in the Unit 2 steam generators as of 2003.

In addition to the depth-based tube repair criteria, the licensee is also authorized to apply the voltage-based tube repair criteria for predominantly axially-oriented outside diameter stress corrosion cracking at the tube support plate elevations. Although authorized to implement the voltage-based repair criteria, the licensee has not found it necessary to implement these criteria since few, if any, indications subject to this repair criteria have been identified at Unit 2. In addition, the licensee is authorized to leave flaws within the tubesheet region in service provided they satisfy the F*/EF* repair criterion. The major cause of degradation within the tubesheet region is primary water stress corrosion cracking at the roll transition zones. Secondary side intergranular attack and outside diameter stress corrosion cracking have also been observed at this location.

ENCLOSURE

With respect to the scope of the inspections in 2002 and 2003, the licensee performed full length bobbin coil inspections of 100 percent of the tubes in both of the steam generators. In addition, a rotating probe equipped with a +Point™ coil was used to inspect from 3 inches

above the hot-leg tubesheet to the tube end (i.e., the entire portion of the tube in the tubesheet was inspected). In addition, a rotating probe equipped with a +Point™ coil was used to inspect the U-bend region of 100 percent of the row 1 and 2 tubes in 2002 and 2003.
In 2003, a +Point coil was also used to inspect the U-bend region of 100 percent of the tubes in rows 3 through 11 and the entire portion of the tube in the cold-leg tubesheet for 20 percent of the tubes. Additional rotating probe examinations were performed at other locations as discussed in the licensee's submittals.

The licensee provided the scope, extent, methods, and results of their SG tube inspections in the documents referenced above. The licensee also described corrective actions (i.e., tube plugging or repair) taken in response to the inspection findings. There were several findings to note as a result of the review of the 2002 and 2003 inspection reports:

> Two Westinghouse Alloy 600 explosive plugs installed in 1980 exhibited signs of leakage in 2002. The source of the leak could not be confirmed (i.e., it could not be determined if the leak was from a gap between the plug and the tube wall or from a flaw in the plug material). Neither tube was swollen and both plugs were replaced with Westinghouse Alloy 690 welded tubesheet plugs.
>
> Of the 5 indications of cracking in the row 1 and 2 U-bends identified at Unit 2 since the commencement of operation, one was identified in 1997 (an axial indication), one was identified in 2001 (a circumferential indication), and three were detected in 2003 (circumferential indications). All indications were at the tangent point. Following the 2002 outage, the staff summarized several recommendations concerning flaw detection in the U-bend region. These recommendations are contained in a letter dated January 27, 2003 (ML030080581).
>
> Following the thermal stress relief to the U-bend region of the tubes in rows 1 and 2, dent-like tube indications occurred near the two uppermost tube support plates of the tubes in these rows. In 2003, only two dents that exceeded the licensee's 2 volt dent reporting threshold were identified. The size of the 2003 dents were similar to previous examinations performed after the stress relief (i.e., the changes in the magnitude of the dents were within the expected variability associated with the eddy current examination). Additional details regarding the denting observed at Units 1 and 2 following the U-bend stress relief are contained in a letter dated April 29, 2004 (ML041180217).
>
> No cracks have ever been identified in the dents at PINGP, Units 1 and 2.

In the documents reference above, the licensee also clarified its inspection and repair practices for flaws located within the tubesheet region. Specifically, the licensee indicated that:

> With eddy current uncertainty included, the F* and EF* distances are 1.27- and 1.87-inches, respectively. The eddy current uncertainty is 0.2-inches.
>
> The uncertainty in the length of the F* re-rolls is controlled by the length of the rolls in the roller. This length can be measured more accurately than the eddy current distance. The effective length of the rolls in the roller used to install the F* re-rolls is 1.25-inches, and the effective length of the rolls in the roller used to install the EF* re-rolls is 2.0 inches. Given the limited uncertainty in the dimension of the rolls, the F* and EF*

distance (1.07- and 1.67-inches) is assured based on the effective length of the rolls and by not allowing any degradation in the re-rolls or elevated re-rolls.

The F* criteria only applies to re-rolls below the mid-plane of the tubesheet. EF* could be applied to the lower region of the tubesheet; however, in practice the EF* is not applied below the midplane of the tubesheet.

As a result of observing minor leakage from the original rolled region in tubes with rerolls, the licensee instituted a practice of rerolling the original rolled region in tubes in which new rolls (i.e., rerolls) were to be installed. This practice seals off the axial cracks found in the original rolled region thereby limiting the leakage from the crack(s). This rerolling process may alter the crack such that it becomes electrically conductive and may result in the flaw being no longer detectable. These indications are referred to as safety analysis diagram and maintenance assembly-disassembly indications for single and multiple axial indications, respectively. When indications are no longer detectable, the location of the indication is calculated based on the original location of the indication.

New indications have been observed in re-rolls (i.e., rolls above the original roll) after one cycle of operation and some preexisting cracks below the F* and EF* distances have exhibited some minor growth. In addition, some flaws have been observed in the non-hard rolled portion (i.e., unexpanded portion) of the tube between the original roll and the re-roll (i.e., rolls above the original roll). No indications of denting or restrictions (that would prevent the passage of eddy current probes) have been observed between the roll expansions.

The reference point for indications within the tubesheet depends on the number of rolled tube expansions. For tubes that have not been rerolled, the location of indications within the tubesheet are reported from the lower edge of the expansion transition. For tubes that have been rerolled, indications are reported from the lower edge of the lower expansion transition of the upper most reroll (refer to Figure 1 and page 3 of 10 of the June 21, 2004 letter).

Based on a review of the information provided, the staff concludes that the licensee provided the information required by their technical specifications. In addition, the staff concludes that there are no technical issues that warrant follow-up action at this time since the inspections appear to be consistent with the objective of detecting potential tube degradation and the inspection results appear to be consistent with industry operating experience at similarly designed and operated units.

cc:

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