

March 21, 2008

Mr. Michael D. Wadley
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 - REVIEW OF
STEAM GENERATOR INSPECTION SUMMARY REPORTS 2006 STEAM
GENERATOR TUBE INSPECTIONS (TAC NO. MD5727)

Dear Mr. Wadley:

By letters dated December 18, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML063530271), December 26, 2006 (ADAMS Accession No. ML063600314), March 14, 2007 (ADAMS Accession No. ML070780140), and February 21, 2008 (ADAMS Accession No. ML080520187), Nuclear Management Company, LLC, submitted information summarizing the results of the 2006 steam generator tube inspections performed at Prairie Island Nuclear Generating Plant Unit 2 during refueling outage 24 (2R24). In addition to these reports, the Nuclear Regulatory Commission staff summarized additional information concerning the 2006 steam generator tube inspections at Prairie Island Unit 2 in a letter dated January 31, 2007 (ADAMS Accession No. ML070170446).

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 warrant follow up action at this time. If you have any further questions, please feel free to contact me at 301-415-8371 or mlc@nrc.gov.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-306

Enclosure:
Staff Evaluation

cc w/encl: See next page

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*Memo from A. Hiser to L. James

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July 2006

REVIEW OF STEAM GENERATOR INSPECTION SUMMARY REPORTS

PRAIRIE ISLAND UNIT 2

2006 STEAM GENERATOR TUBE INSPECTIONS

TAC NO. MD5727

DOCKET NO. 50-306

By letters dated December 18, 2006 (ML063530271), December 26, 2006 (ML063600314), March 14, 2007 (ML070780140), and February 21, 2008 (ML080520187), Nuclear Management Company, LLC, submitted information summarizing the results of the 2006 steam generator (SG) tube inspections performed at Prairie Island Nuclear Generating Plant (PINGP) Unit 2 during refueling outage 24 (2R24). In addition to these reports, the U.S. Nuclear Regulatory Commission (NRC) staff summarized additional information concerning the 2006 SG tube inspections at Prairie Island Unit 2 in a letter dated January 31, 2007 (ML070170446).

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 inches and a nominal wall thickness of 0.050 inches. 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 were no sleeves installed in the Unit 2 SGs prior to 2R24 and no sleeves were installed during 2R24.

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.

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.

The NRC staff has the following notes/observations as a result of reviewing the aforementioned documents submitted by the licensee:

- All crack-like indications detected were contained within the tubesheet, except for one indication in SG 22, which extended 0.08 inches above the top of tubesheet in the sludge pile region
- No degradation of secondary side components was observed during the visual inspection of SG 22.
- Several potential loose parts were identified in SG 21 and SG 22. Several of these could not be retrieved from the SGs. There was no tube damage associated with these loose parts.
- Dent-like indications at the uppermost TSPs, attributed to the thermal stress relief of the U-bend region, have been relatively stable in quantity and severity (i.e., bobbin voltage measurements) between inspections in 2005 and 2006.

Based on a review of the information provided, the NRC 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.