November 26, 2003

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 - SUMMARY OF CONFERENCE CALL WITH NUCLEAR MANAGEMENT COMPANY, LLC REGARDING THE 2003 STEAM GENERATOR INSPECTIONS (TAC NO. MC0503)

Dear Mr. Solymossy:

On September 25, 2003, the Nuclear Regulatory Commission (NRC) staff participated in a conference call with the Nuclear Management Company, LLC (the licensee), regarding the steam generator (SG) tube inspection activities at Prairie Island Nuclear Generating Plant, Unit 2, during refueling outage 22 (RFO-22). The conference call was voluntary on your part and occurred after the majority of the tubes had been inspected, but before the SG inspection activities were completed. A summary of the conference calls is provided in Enclosure 1 and the handout provided by the licensee in support of the conference call is provided in Enclosure 2.

This completes the NRC staff's efforts under TAC No. MC0503.

If you have any questions regarding this matter, please contact me at (301) 415-4106.

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

Enclosures: 1. Summary of Conference Call
2. Licensee Handout

cc w/encls: See next page

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DISTRIBUTION:

PUBLIC	RBouling	WBateman	KKarwoski	WRuland
PDIII-1 Reading	OGC	KRiemer, RIII	ALund	Tmensah
LRaghavan	ACRS	JJacobson, RIII	TMarsh/ELeeds	
AMcMurtray	PLouden	MHolmberg, RIII	*Provided summary input by memo	

OFFICE	PDIII-1/PM	PDIII-1/LA	EMCB/SC*	PDIII-1/SC
NAME	AMcMurtray	THarris for RBouling	ALund	LRaghavan
DATE	11/26/03	11/24/03	11/03/03	11/25/03

ADAMS Accession No. ML033210569

OFFICIAL RECORD COPY

Prairie Island Nuclear Generating Plant,
Units 1 and 2

cc:

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SUMMARY OF NUCLEAR REGULATORY COMMISSION STAFF

CONFERENCE CALL WITH NUCLEAR MANAGEMENT COMPANY, LLC

ON SEPTEMBER 25, 2003

REGARDING STEAM GENERATOR INSPECTION RESULTS AT

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

On September 25, 2003, the Nuclear Regulatory Commission (NRC) staff participated in a conference call with Nuclear Management Company, LLC (the licensee), regarding the ongoing steam generator tube inspection activities at Prairie Island Nuclear Generating Plant, Unit 2. The licensee discussed the topics based on the agenda that the NRC staff provided in a letter dated September 10, 2003 (ADAMS Accession No. ML032380406). The licensee provided written material in support of the conference call (See Enclosure 2 to the letter forwarding this enclosure). Additional details discussed during the conference call are summarized below.

The licensee inspected the full length (i.e., hot leg tube end to cold leg tube end) of all active steam generator tubes using the bobbin coil probe(s). In addition, the licensee inspected various tube regions and special interest indications using the motorized rotating pancake coils (MRPC). This included the U-bend, top of the tubesheet, plugs, re-rolls, dents, and previously identified degradation. Of particular interest, the licensee inspected the U-bend region (7th tube support plate on the hot leg to the 7th tube support plate on the cold leg) of 100 percent of the tubes in rows 1 through 11 with the MRPC probe. This inspection scope was based, in part, on the recent operating experience at Diablo Canyon, Unit 2, which indicated a potential for circumferential cracking in higher row U-bends. The licensee stated that an engineering analysis indicated that rows 1 through 11 were most susceptible to degradation based on temperatures and residual stresses.

As of September 25, 2003, at 12:00 AM, the licensee had completed approximately 60 percent to 70 percent of the eddy current data analysis. In steam generator 21, the licensee detected indications in the cold leg due to thinning, new and old anti-vibration bar (AVB) wear indications, and hot leg crevice indications. In steam generator 22, the licensee detected indications in the cold leg due to thinning, new and old AVB wear indications, hot leg crevice indications, top of tubesheet indications, and U-bend indications. Further details on these findings are as follows:

> In 1985, the licensee replaced the original factory-installed AVBs with stainless steel AVBs at Prairie Island Unit 2. The steam generator tubes contain “old AVB wear indications” at the old AVB contact points and “new AVB wear indications” at the new AVB contact points. All the AVB wear indications are monitored and dispositioned in accordance with the technical specifications. However, they are tracked separately because the “old” indications are expected to be stable in terms of flaw growth (in depth).
>
> The indications in the “hot leg crevice” in the tubesheet consist of both primary water stress corrosion cracking and outside diameter stress corrosion cracking and are axial in

nature. The range of lengths, 0.01 inches to 3.01 inches, is typical for flaws in this region at Prairie Island Unit 2.

The indications at the top-of-tubesheet are axial indications in the sludge pile region.

Two U-bend circumferential indications were detected in row 1 near the tangent point of the U-bend region in steam generator 22. The circumferential extent was approximately 130 to 137 degrees and the maximum depth was about 60 percent to 62 percent through wall.

The licensee stated that the inspection results for both steam generators fall into category 3 (C-3) as defined in the plant technical specifications. The licensee satisfied the reporting requirements in the plant technical specifications for the category 3 inspection results during this telephone call.

During this outage, the licensee reported all dents greater than 2 volts from the bobbin coil examination. During past outages, only dents greater than 5 volts were reported. In addition, the licensee has identified an increased number of dents at the 6th and 7th tube support plate intersections in rows 1 and 2. The licensee believes that this increased denting was caused by the heat treatment process performed on rows 1 and 2 U-bends during the 2000 refueling outage. This same issue was also identified at Unit 1. The licensee has not detected any cracking in the dents.

The licensee performed a foreign object search and retrieval (FOSAR) inspection of one steam generator during this outage and identified several potential loose parts (PLP) that are located on the top of the tubesheet on the hot leg side. All these PLPs were present during the last inspection and there is no tube wear associated with the PLPs. The licensee inspects 100 percent of the hot leg top of tubesheet to the top of the sludge pile with MRPC every outage and 20 percent of the cold leg top of tubesheet (to +1 inch) every other outage to inspect for wear from the PLPs.

The licensee planned to perform up to 10 in-situ pressure tests. At the time of the conference call, two circumferential indications located in row 1 U-bends were selected as candidates. The licensee stated that these indications did not meet the in-situ test screening criteria and were only being tested as a conservative measure.

The staff requested that the licensee contact the NRC if they identified any of the following conditions as they completed their steam generator inspection activities:

(1) New degradation mechanisms.
(2) Failure of a pressure test or leakage during an in-situ pressure test.

The staff did not identify any other issues requiring followup.