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U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Unit 1
Docket 50-282
License No. DPR-42

Response to Request for Additional Information on 2006 Prairie Island Nuclear Generating Plant (PINGP) Unit 1 Steam Generator Tube Inspections

In an email dated December 12, 2006, the Nuclear Regulatory Commission (NRC) Staff made the following request:

> By letters dated September 1, 2006 (ML062550530), and May 25, 2006 (ML061450543), Nuclear Management Company (the licensee) submitted information summarizing the results of the 2006 steam generator tube inspections at Prairie Island Unit 1. These inspections were performed during the twenty fourth refueling outage (1R24). In addition to this report, the U.S. Nuclear Regulatory Commission staff summarized additional information concerning the 2006 steam generator tube inspections at Prairie Island Unit 1 in a letter dated July 10, 2006 (ML061680006).
>
> The NRC staff has reviewed the information the licensee provided and determined that additional information is required in order to complete the evaluation. The additional information being requested is enclosed. From discussions with you, the root cause evaluation is not complete. Please provide your estimated date of completion of the root cause evaluation and arrange a teleconference to discuss the staff request upon availability of the root cause evaluation report.
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> 1. On May 18, 2006, the Nuclear Regulatory Commission (NRC) staff conducted a phone call with representatives from Prairie Island to discuss the steam generator tube inspections during their 24th refueling outage. During the call there was a discussion of a root cause analysis that was commenced in response to finding more wear indications than expected. Please discuss the scope and results of that analysis.

In addition, please discuss any planned corrective actions in response to the results. Also, please discuss whether there is any axial or radial pattern for the wear indications detected. If so, discuss the significance.

2. During the operating cycle prior to the 24th refueling outage, please discuss whether there were any chemical excursions in the steam generators (e.g., from a leaking condenser tube). If so, discuss the extent of the chemical excursion, the corrective action and any long term implications for the steam generators.

Nuclear Management Company, LLC (NMC) Response

The NMC response was discussed during a telecon on August 23, 2007. The responses provided below are based largely on that discussion. As discussed previously with the NRC Staff, the vendor root cause evaluation is not yet complete and is not required by the NRC Staff at this time.

Response to Question 1:

The NMC staff completed an apparent cause evaluation (ACE) which is summarized as follows:

> The cause of the unexpected eddy current indications is a lack of industry operating experience showing tube support plate (TSP) wear in similar steam generators. Because there was no industry experience, a specific tube standard for this type of wear could not be developed. Determination of the reason there was not any industry experience is one of the tasks of the AREVA Root Cause Analysis. From preliminary evaluations of PINGP steam generator TSP wear using the French Method of bobbin probe data analysis, the following statement is made:
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> > "If TSP wear equivalent to that detected at PINGP Unit 1 was present at nuclear power plants in France, it would not be detected and consequently it would not be reported applying the French data analysis rule to the bobbin probe technique."

In addition the ACE notes:

> **Corrective Actions:**
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> There are several possible scenarios for the steam generator TSP wear and corrective actions will be based on the applicable scenario. These scenarios are:
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> - This is an expected event (tube wear on TSPs) and the wear will not change in number of tubes affected or size of the individual indications after the initial operating cycle(s). This could be the reason TSP wear has

not been seen in France or other countries which have AREVA steam generators.

- This is unique to the 56/19 replacement steam generators installed at PINGP, either through their design, installation, or operating conditions. However the TSP wear will not affect a large number of tubes and continued operation will not be significantly affected. The results of the AREVA Root Cause Analysis should provide more information on this.
- This is unique to the PINGP steam generators and there is a significant potential that it could affect the long term operation of the steam generators. The results of the AREVA Root Cause Analysis could lead to modifications of the Unit 1 steam generators (similar to the anti-vibration bar replacement on the original steam generators) and design changes to the potential Unit 2 replacement steam generators.

With multiple scenarios there are multiple corrective actions. Furthermore, since the corrective actions will be based in a large part on the results of the AREVA Root Cause Analysis, which is not done, all corrective actions are not presently known. Therefore the corrective actions are broken into two groups: those that are known now; and those which will be known after the root cause analysis is done.

Actions that are known now:

- Update plant procedure 1H25.1, "Unit 1 Assessment of Steam Generator Tube Degradation Mechanism", with the results of the 1R24 Steam Generator Tube Inspections and the Condition Monitoring Operational Assessment.
- Update the Steam Generator Strategic Plan with the results of the 1R24 Steam Generator Tube Inspections and the Condition Monitoring Operational Assessment.
- Obtain TSP standards to be used for sizing TSP-tube defects.
- Monitor AREVA steam generator operating experience at other plants. Because small but significant differences exist between steam generator inspection and operation practices in the United States and elsewhere, NMC should remain in the Framatome Reactors Owners Group (FROG) Steam Generator Technical Committee (SGTC) which provides an informational exchange between nuclear power plants with AREVA steam generators. Presently PINGP has the only 56/19 AREVA replacement steam generator in this country. Any information from other sources would be of great benefit.
- Make a decision on whether to inspect the steam generators during the 1R25 outage. The results from the 1R24 Steam Generator Tube Inspections and the Condition Monitoring Operational Assessment provide a single data point; inspecting the steam generators during the next outage may provide answers to the following questions:
    - Will more tubes have indications?

- Will the indications grow during future operating cycles and by how much?

Subsequent to the completion of the ACE, NMC decided to inspect the steam generators during the 1R25 outage (the next Unit 1 refueling outage). The inspection scope is provided in Table 1 below.

Table 1
1R25 Outage Steam Generator Inspection Scope

INSPECTION SCOPE	PROBE TYPE	S/G 11	S/G 12
Full Length	Bobbin	100%	100%
Rows 1 through 9 U-Bends	MRPC[1]	0%	0%
Hot Leg Tubesheet	MRPC	0%	0%
Cold Leg Tubesheet	MRPC	0%	0%
Post In Situ Pressure Test	MRPC	100%	100%
Tubes with indications requiring additional inspections	MRPC	~200[2]	~150[2]

[1] Motorized Rotating Pancake Coil
[2] Approximate number of tubes based on 1R24 inspection results

NMC has not identified an axial or radial pattern of wear indications with respect to locations within the steam generator tube bundles. The locations and numbers of indications vary between the two steam generators. The 11 Steam Generator had 57 indications on 44 tube hot legs adjacent to tube support plates from TSPs 1 through 7 with the majority adjacent to TSPs 3 and 4 and with all indications on the periphery of the bundle. The 12 Steam Generator had 7 indications on 6 tubes adjacent to TSPs which were on both the hot and cold legs with all indications on the periphery of the tube bundle.

The AREVA Root Cause analysis also considered anti-vibration bar (AVB) wear indications and concluded they were caused by a few unexpected gap configurations that have led to early initiation of wear that will stop to significantly propagate as soon as the gap configuration has stabilized. The 11 Steam Generator had 9 indications on 5 tubes adjacent to anti-vibration bars (AVBs) from AVB 3 through AVB 7. The 12 Steam Generator had 32 indications on 16 tubes adjacent to AVBs from AVB 2 through AVB 9. The replacement steam generators have 8 Tube Support Plates and 9 Anti-vibration bars. Because the numbers and locations of indications vary significantly between the generators, the location of indications within the bundle is not thought to be significant.

Response to Question 2:

There was a small condenser leak (estimated less than eight gallons per day) that was corrected by plugging two condenser tubes. None of the EPRI secondary chemistry

guideline action levels were exceeded as a result of the leak. The indications of concern on the Unit 1 steam generators correlate strongly to mechanical wear, that is, chemistry is not expected to contribute to the indications observed during the inspection. No long term steam generator implications are anticipated as a result of the condenser tube leak.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

Joel P. Sorensen for M.D.W.

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cc: Administrator, Region III, USNRC
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