September 2, 2004

Daniel J. Malone
Site Vice President
Palisades Nuclear Plant
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: PALISADES NUCLEAR POWER PLANT, REQUEST FOR ADDITIONAL INFORMATION RE: STEAM GENERATOR TUBE INTEGRITY ASSESSMENT FROM THE 2003 REFUELING OUTAGE (TAC NO. MC2747)

Dear Mr. Malone:

By letter to the Nuclear Regulatory Commission (NRC), dated April 13, 2004, supplemented by letter dated June 28, 2004, the licensee submitted a report that summarizes the steam generator tube integrity assessment performed at Palisades Nuclear Power Plant during the 2003 refueling outage.

NRC staff is reviewing your request and has determined that the information identified in the enclosure to this letter is needed to complete its evaluation. The enclosed request was discussed with Ms. Amy Hazelhoff during a conference call on August 17, 2004. Your response to this request for additional information is requested within 90 days from the date of this letter. Ms. Hazelhoff has agreed to this request.

If you have any questions, please contact me at (301)415-1345 or email jfs2@nrc.gov.

Sincerely,

/RA/

John F. Stang, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: Request for Additional Information

cc w/encls: See next page

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DATE	8/31/04	8/30/04	08/26/04	8/31/04

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REQUEST FOR ADDITIONAL INFORMATION FOR

NUCLEAR MANAGEMENT COMPANY

PALISADES STEAM GENERATOR TUBE

INSPECTION REPORT FOR THE 2003 OUTAGE

DOCKET NO. 50-255

By letter dated April 13, 2004 (ML041100667), supplemented by letter dated June 28, 2004 (ML041890415), Nuclear Management Company, LLC, the licensee for Palisades, submitted a report that summarizes the steam generator (SG) tube integrity assessment performed during the 2003 refueling outage. In reviewing these two letters and their attachments, the Nuclear Regulatory Commission (NRC) staff determined that the responses to the following questions are required to complete the evaluation.

1. Provide a general description of the replacement SGs, Include in this general description the total number of tubes, tube diameter, tube wall thickness, tube pitch, tube support thickness and any other noteworthy design characteristics from a SG tube integrity standpoint (e.g., full length stress relieved of the row 1 through 10 tubes). In addition, provide sketches of the steam generators that depict the tube support naming conventions and tubesheet maps that depict the rows and columns of the tubes.

2. On page 7 of your in-service inspection report, you indicated that 100 percent of rows 1, 2 and 3 U-bends were inspected in SG E-50A. You also indicated that only 25 percent of rows 1 and 2 U-bends were inspected in SG E-50B and that no expansion of this sample was required. Given that you identified primary water stress corrosion cracking (PWSCC) in the U-bend region of a row 2 tube in SG E-50A and the potential for PWSCC to develop in the U-bend of other low row tubes, discuss the technical basis for not expanding the U-bend inspection to include 100 percent of the tubes in rows 1, 2, and 3 in SG E-50B. Discuss how this was accounted for in your operational assessment.

3. On page 8 of your in-service inspection report, you indicated that one tube (R135C100) was plugged due to a restriction to the passage of a +Point™ rotating pancake coil (RPC). Describe the nature and location of the restriction. Include in your response a discussion of the extent of the restriction and whether or not it was service induced. What was the largest size probe to be passed through the tube during this outage and previous outages? If the restriction was service induced, discuss the basis for not expanding the inspection to include 100 percent of the tubes.

4. On page 9 of your in-service inspection report, you indicated that after tube plugging, there were 7 tubes non-expanded in SG E-50A and 1 in SG E-50B that remained in

service. Then, on page 32, you stated that the non-expanded tube population is 6 tubes in SG E-50A and 1 in SG E-50B. On page 33 you indicated that only 7 tubes remained in service with no expansion in the tubesheet region. Clarify how many non-expanded tubes remain in service in each SG.

5. On pages 15, 16, and 18 of your in-service inspection report, you indicated that 25 tubes in SG E-50A and 21 tubes in SG E-50B were plugged. You then stated that after 8 cycles of operation, 57 additional tubes in SG E-50A and 51 in SG E-50B have been plugged, which brings the total number of tubes plugged to 365 and 360, respectively. Please clarify how many tubes were plugged prior to placing the SGs in service and clarify how many tubes were plugged prior to the 2003 refueling outage.

6. With respect to the eggcrate wear indications, address the following:

   On page 26 of your in-service inspection report, you indicated that 100 tubes in SG E-50A and 68 tubes in SG E-50B were reported as having a distorted support indication (DSI) from bobbin, were +Point™ RPC tested and reported as VOL. On the same page, you stated that the remaining DSI reported by bobbin were at supports. These included 10 DSI in SG E-50A and 45 DSI in SG E-50B, which were +Point™ RPC tested and no degradation was found. Please clarify this discussion. Is it implying that 110 DSI's were reported in SG E-50A and of these 110, 100 were attributed to wear and 10 had no degradation?

   In addition, please clarify an apparent discrepancy in Table 16. Table 16 indicates that 114 tubes in SG E-50A had wear indications while the text indicates 100.

   Provide the technical basis for determining when wear at eggcrates should be RPC inspected. Specifically, address in your response whether there is any data that indicates any appreciable change in the bobbin coil signal will result from an axial or circumferential flaw that is present in a wear scar.

7. On page 39 of your in-service inspection report, you indicated that a sodium intrusion event has introduced sodium into the crevices. Discuss this event in detail or if already provided to NRC, please provide references to documents where this is discussed.

8. On page 8 of your in-service inspection report, you indicated that stay rod locations are areas of high residual stress and susceptible to stress corrosion cracking. Discuss why this area is considered to be susceptible to high residual stresses and what degradation mechanisms are more prone to appear.

9. Regarding your disposition of the freespan differential (FSD) indications:

   Discuss what constitutes a "significant change" in the signals when compared to baseline and your technical basis for this criteria (e.g., Is it based on test repeatability?).

Approximately 260 tube locations were inspected with an RPC probe because of a new FSD or because an existing FSD changed. Clarify whether the indications that required RPC testing were considered new or changed. If new indications, discuss the cause. If changed indications, discuss the reason for the change.

Clarify the number of tubes with FSD indications that required RPC testing. Table 19 implies that only 1020 of the 1619 FSDs were dispositioned based on history review.

10. You indicated that no changes were noted in the bobbin coil examinations of dings. You further indicated that a bobbin coil review was performed for all free span dings greater than or equal to 2 volts. Clarify how many dings are present in your steam generators (e.g., “x” dings greater than 2 volts, “y” dings greater than 5 volts). Clarify that 95 percent of all dings (greater than 2 volts) are located at vertical straps. Clarify whether circumferential cracking has been observed at dings at Palisades or at other similarly designed and operated units. If it has been observed, discuss the basis for not expanding the scope of inspection to 100 percent. Based on the material provided, the staff inferred that for dings whose voltages are between 2 and 5 volts, the 2003 bobbin coil data was compared to baseline data to determine if any change occurred. If this is the case, discuss the technical basis for the approach including whether the bobbin coil signal from a ding would change significantly if a flaw (axial or circumferential) were to initiate at this location. The staff notes that many plants with mill annealed Alloy 600 tubes inspect 100 percent of all dents/dings that have bobbin voltages greater than 5 volts.

11. In your assessment of tube wear, discuss how you accounted for degradation in the tubes you did not inspect. In addition, clarify the discussion regarding the 67 percent through wall wear indication detected in 1999 (p. 26). Was the wear associated with this tube caused by some local support condition such that it necessitated inspection of surrounding tubes?

12. On page 28 of your in-service inspection report, you indicated that an axial u-bend outside diameter stress corrosion cracking (ODSCC) indication was screened. Was an axial ODSCC indication detected in the U-bend or is this a typographical error?

13. Discuss whether any primary to secondary leakage has been observed during the current cycle.

14. For all tubes with possible loose parts indications, discuss whether a visual inspection was performed at these locations. Were any loose parts left in the SG? If so, discuss whether a tube integrity evaluation was performed.