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U S Nuclear Regulatory Commission  
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Prairie Island Nuclear Generating Plant Unit 2  
Docket 50-306  
License No. DPR-60

2005 Unit 2 Steam Generator Category C-3 Inspection Results 30-Day Report

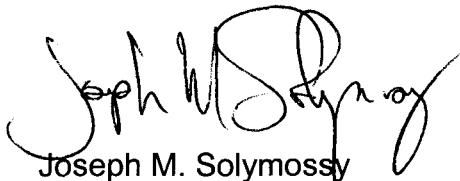
In accordance with Technical Specification 5.6.7.3 this special report, due to Category C-3 inspection results of the Unit 2 steam generator tubing, is provided for the information of the NRC Staff.

The results of the inspection of 21 Steam Generator and 22 Steam Generator were classified as Category C-3 in accordance with Technical Specification 5.5.8 because more than 1% of the inspected tubes in each Steam Generator were defective. The NRC Staff was informed of the Category C-3 classification by telephone on May 17, 2005.

The 30-day special report on the Category C-3 steam generator inspection results is provided as Enclosure 1 (Prairie Island Unit 2 Steam Generator Category C-3 Inspection Results 30-Day Report) to this letter.

Summary of Commitments

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



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Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC  
Project Manager, Prairie Island, USNRC  
Resident Inspector, Prairie Island, USNRC  
Minnesota, Department of Commerce

## ENCLOSURE 1

### 2005 UNIT 2 STEAM GENERATOR CATEGORY C-3 INSPECTION RESULTS 30-DAY REPORT

#### Purpose

This report fulfills the special reporting requirements of Prairie Island Technical Specification 5.6.7.3. This report is required whenever the steam generator tube inservice inspection finds more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective. This report summarizes the inspection results, the investigation into causes of major tube degradation and corrective measures. Corrective measures to prevent recurrence of Category C-3 inspections are discussed. Steam generator inspection results continue to exceed the category C-3 limits, so corrective measures do not prevent recurrence. However, careful inspections and repairs coupled with chemistry controls and low operating temperature provide assurance of safe and reliable operation of Unit 2 steam generators.

#### Summary of Inspection Results

The inservice inspection for Unit 2 Steam Generators occurred from May 10, 2005 through May 16, 2005. The inservice inspection consisted of inspection of 100% of the full length of tubing with the bobbin coil (except rows 1 through 4 u-bends), 100% of the hot leg tubesheet region, 25% of the hot leg Alloy 690 roll plugs, 100% of the rows 1 through 4 u-bends and 33% of the rows 5 through 8 u-bends with mechanical rotating probes with +Point™ coil.

As a result of the eddy current inspections, 6.0% (191 of 3158) of the inspected tubes in 21 Steam Generator contained defects requiring repair. 12 of these tubes were plugged and the remaining 179 tubes were left in service using the F\* and EF\* repair criteria.

As a result of the eddy current inspections, 4.8% (150 of 3138) of the inspected tubes in 22 Steam Generator contained defects requiring repair. Eight of these tubes were plugged and the remaining 142 were left in service using the F\* repair criteria.

#### Investigation into Causes of Major Tube Degradation

There are two major causes of tube degradation in Unit 2 steam generators:

- Secondary side intergranular attack and stress corrosion cracking and
- Primary water stress corrosion cracking.

Secondary side intergranular attack and stress corrosion cracking (IGA/SCC or ODS/SCC) is occurring in the hot leg tubesheet crevice region and at the top of the hot leg tubesheet. This was confirmed by metallurgical examination of three tube samples removed from the original Steam Generator 12 in January 1985. This was also confirmed by examination of

a parent tube section removed during the sleeve pulls in the original Steam Generator 12 in 1996. In addition, three tubes were removed from Unit 1 for Generic Letter 95-05 Voltage Based Repair Criteria in 1997 and ODSCC was confirmed at the hot leg tube support plates.

Primary water stress corrosion cracking (PWSCC) at the roll transition region has been confirmed by metallurgical examination of one roll transition zone removed during sleeve pulls in the original Steam Generator 12 in 1996.

### **Corrective Measures**

Prairie Island participates in utility funded research on steam generator related issues. Corrective measures to reduce and/or prevent tube degradation due to PWSCC and secondary side IGA/SCC have been used by the industry with limited success. Prairie Island corrective measures include:

Chemistry Control: Prairie Island follows both the original equipment manufacturer's water chemistry guidelines and the Electric Power Research Institute secondary water chemistry guidelines. The PWSCC degradation appears to be relatively independent of chemistry and occurs in regions of high residual stress.

High Hydrazine Control: Prairie Island maintains a hydrazine control band of eight times Condensate Oxygen Level to 20 ppb (parts per billion) in the feedwater system.

Feedwater pH Control - Alternate Amine: Prairie Island maintains MPA (3-methoxypropylamine) at seven to nine ppm (parts per million) in the steam generator blowdown.

Molar ratio control to reduce secondary side corrosion: Molar ratio control has been attempted by adjustments to steam generator blowdown resin ratios.

On-line addition of Boric Acid: Prairie Island uses on-line addition of boric acid to Unit 2 steam generators.