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Technical Specification 5.6.7.3

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

**PRAIRIE ISLAND NUCLEAR GENERATING PLANT**
Docket No. 50-282 License No. DPR-42

**2002 Unit 1 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 1 steam generator tubing is provided for the information of the NRC Staff.

The results of the inspection of 11 Steam Generator and 12 Steam Generator were classified as Category C-3 in accordance with Technical Specification 5.6.7.3 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 November 22, 2002. In accordance with Technical Specification 5.6.7.3, the 30 day special report on the Category C-3 steam generator inspection results is provided as Attachment 1 to this letter.

In this letter, we have made no new Nuclear Regulatory Commission commitments. Please contact Jeff Kivi (651-388-1121) if you have any questions related to this letter.

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c: Regional Administrator - Region III, NRC
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Attachments:
1. Prairie Island Unit 1 Steam Generator Category C-3 Tube Inspection Special Report

## ATTACHMENT 1

## PRAIRIE ISLAND UNIT 1 STEAM GENERATOR CATEGORY C-3 TUBE INSPECTION SPECIAL 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 1 steam generators. Unit 1 steam generator replacement is scheduled for the Fall of 2004.

### Summary of Inspection Results

The inservice inspection for Unit 1 Steam Generators occurred from November 19, 2002 through November 27, 2002. The inservice inspection consisted of inspection of 100% of the full length of tubing with the bobbin coil (except row 1 and 2 u-bends and sleeves), 100% of the hot leg tubesheet region, 100% of the sleeves (25% full length and the remaining 75% of the upper joint) and the row 1 and 2 u-bends with mechanical rotating probes with +Point™ coil.

As a result of the eddy current inspections, 2.9% (93 of 3217) of the inspected tubes in 11 Steam Generator contained defects requiring repair. Twenty-two of these tubes were plugged and the remaining seventy-one tubes were left in service using the F* and voltage based alternate repair criteria.

As a result of the eddy current inspections, 6.3% (190 of 3011) of the inspected tubes in 12 Steam Generator contained defects requiring repair. One hundred seventy eight of these tubes were plugged and the remaining twelve were left in service using the F* and voltage based alternate repair criteria.

### Investigation Into Causes of Major Tube Degradation

There are two major causes of tube degradation in Unit 1 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 ODSCC) is occurring in the hot leg tubesheet crevice region, at the top of the hot leg tubesheet, and in the hot leg tube support plate intersection. This was confirmed by metallurgical examination of three tube samples removed from Steam Generator 12 in January 1985. This was also confirmed by examination of a parent tube section removed during the sleeve pulls in 1996. In addition, three tubes were removed for GL 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 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 primary water stress corrosion cracking 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 EPRI 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 100 to 125 ppb as measured in the feedwater system.

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

Conduct Crevice Flushing Operations with Boric Acid: Prairie Island employed crevice flushing from 1986 to 1999.

On-line addition of Boric Acid: Prairie Island began on-line addition of boric acid to Unit 1 steam generators in 1987.

Use of other chemical inhibitors: Titanium chelate has been added since January 1994.