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Technical Specifications: 4.12.E.1
4.12.E.3

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Steam Generator Inspection Reports

In accordance with Technical Specification 4.12.E.1, the following steam generator tube plugging and sleeving information is provided for the information of the NRC staff.

Following the recent inservice inspection of the Unit 1 steam generators, 21 tubes were plugged for the first time, one tube which was previously sleeved was plugged and 158 tubes were sleeved. The percentage of tubes plugged is 2.18% on steam generator 11. The percentage of tubes plugged (including the percentage plugged equivalent for the sleeved tubes) is 3.97% on steam generator 12. The inspection results are summarized in Attachment 1.

This information will be expanded upon in the Inservice Inspection Report for Unit 1 which will be submitted within 90 days of the end of the current refueling outage. Also Table 4.3-13 of the Prairie Island Updated Safety Analysis Report will be updated in the next revision.

The results of the inspection of Steam Generator 12 were classified as Category C-3 in accordance with Technical Specification 4.12 because more than 1% of the inspected tubes in Steam Generator 12 were defective. The NRC Staff was informed of the Category C-3 classification by telephone on November 13, 1992. The 30 day special report on the Category C-3 steam generator inspection, required by Technical Specification 4.12.E.3, is provided as Attachment 2 to this letter.

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If you have any questions concerning this information please call.

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c: Regional Administrator - Region III, NRC
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Attachments: 1. Steam Generator Tube Plugging/Sleeving Summary

2. Prairie Island No. 12 Steam Generator Category C-3 Tube Inspection Special Report

ATTACHMENT 1

STEAM GENERATOR TUBE PLUGGING/SLEEVING SUMMARY

STEAM GENERATOR TUBE PLUGGING/SLEEVING SUMMARY
(Prairie Island Technical Specification 4.12.E.1 15-Day Report)

Steam Generator No. 11 Summary

New Indications Plugged This Outage:	1
Total Tubes Plugged:	74
Total Tubes Sleeved:	0

11 Steam Generator % Plugged: 2.18%

New Indications:

One tube was plugged for thinning at the cold leg tube support plate.

Tube Plug Inspection:

A visual plug inspection was done this outage. There were no unusual indications found.

Tube Plug Removal:

Per the requirements of NRC Bulletin 89-01, five hot leg plugs from heat NX5222 were removed due to susceptibility to primary water stress corrosion cracking. These plugs were replaced with Alloy 690 mechanical plugs.

Supplemental Tubesheet Examination:

Because of the large number of indications found by the 100% rotating pancake coil (RPC) probe inspection of the tubesheet region in Steam Generator 12, 20% of the tubesheet region in Steam Generator 11 was inspected by rotating pancake coil as recommended by the EPRI PWR Steam Generator Examination Guidelines. The result for all of the tubes inspected in this 20% sample was No Detectable Degradation.

Steam Generator No. 12 Summary

New Indications Plugged This Outage:	20
New Indications Sleeved This Outage:	158
Sleeved Tubes Plugged This Outage:	1
Total Tubes Plugged:	120
Total Tubes Sleeved:	319

12 Steam Generator % Plugged + % Sleeved Equivalent: 3.97%

Plug Removal: No plugs were removed in response to NRC Bulletin 89-01 during this outage.

New Indications:

178 defective tubes were identified.

One tube was plugged for thinning at a cold leg tube support plate.

The remaining 177 defective tubes all had degradation in the tubesheet region. These defects were all identified during a supplemental 100% examination of the tubesheet region by the rotating pancake coil probe. This supplemental examination was performed in order to best identify t'.se tubes which have incipient degradation in the tubesheet region and which could leak during the next fuel cycle. The 177 defective tubes had the following indications:

- For 19 of these tubes, the only type of indications found were axial indications at the roll transition zone which are representative of primary water stress corrosion cracking (PWSCC).
- Six tubes contained axial indications associated with both roll transition zone primary water stress corrosion cracking and tubesheet crevice secondary side stress corrosion cracking (IGA/SCC).
- The remaining 152 tubes contained single or multiple axial indications in the lower half of the tubesheet crevice region associated with the secondary side stress corrosion cracking (IGA/SCC) occurring in the tubesheet of 12 steam generator.

The results of this inspection of Steam Generator 12 were classified as Category C-3 by Technical Specification 4.12 because more than 1% of the inspected tubes in Steam Generator 12 were defective. The NRC Staff was informed of the Category C-3 classification by telephone on November 13, 1992. The 30 day special report on the Category C-3 steam generator inspection is provided as Attachment 2 to this submittal.

Defective Lower Sleeve Weld:

One sleeve at tube position R18C38 was plugged with a welded tubesheet plug due to pin hole leaks in the lower sleeve weld. Leakage was identified during the fill for the leak test at 700 psig. See detailed discussion below.

Defective Welded Sleeve Plug Weld:

Because this was the second time (first time was in June 1991) that a defective lower sleeve weld was identified by leakage with the static head of water pressure, the lower sleeve welds of the 162 previously (April 87, August 88, and January 90) installed sleeves were examined with ABB Combustion Engineering's lower weld visual inspection tool after brushing.

One additional defective weld in the hot leg at tube position R19C32 was identified. A tiny circle of boric acid was found on the outside of the only welded sleeve plug installed in Steam Generator 12. This welded sleeve plug was removed and replaced with a welded tubesheet plug. Water drained from the tube when the welded plug was removed indicating some leakage from the primary system into the plugged tube had occurred through the defect in the weld. Tube R19C32 did not leak during the fill and hydro at 700 psig.

Tube Plug Inspection:

The plug/sleeve inspection in 12 steam generator was satisfactory.

Tube Leakage Identification:

The primary to secondary leak rate in Unit 1 was .001 GPM prior to shutdown for refueling. Just prior to shutdown on October 24, 1992 there was a step increase from 140 CPM to 350 CPM on the air ejector monitor, but a leak rate was not assigned (estimated to be .002 GPM). After shutdown, the water in Steam Generator 12 contained radioactivity, but there was none in Steam Generator 11. None of the defective tubes identified by the eddy current examination were candidates for significant leakage. However, during the leak test of Steam Generator 12, tube R18C38 (hot leg) dripped one drop per 12 minutes with static head and 1 drop per 3 minutes at 700 psig. We believe tube R18C38 was the source of leakage prior to shut down.

The eddy current data was reviewed for this tube from three inspections. The results of that review show degradation of the parent tube occurring beneath the sleeve:

- January 1990 Bobbin Coil Examination: Evaluation finds no detectable degradation. This tube was sleeved in January 1990 as a preventive sleeving measure.

- June 1991 Cross-wound Bobbin Coil Examination: An indication in the parent tube has now become apparent at 1.9 volts and about two inches above the roll transition.

- October 1992 Cross-wound Bobbin Coil Examination: The indication has grown to 3.6 volts.

Thus, the primary to secondary side leakage path was through the defect in the lower sleeve weld and, then, through the stress corrosion crack in the tubesheet region of the original tube. It is probable that this tube was the source of the leakage and increased air ejector radiation monitor readings during the shutdown of Unit 1 on October 24, 1992.

ATTACHMENT 2

PRAIRIE ISLAND NO. 12 STEAM GENERATOR

CATEGORY C-3 TUBE INSPECTION

SPECIAL REPORT

Prairie Island No. 12 Steam Generator Category C-3 Tube Inspection Special Report

Purpose

This report fulfills the special reporting requirements of Prairie Island Technical Specification 4.12.E.3. A special report is required whenever the results of a steam generator tube inservice inspection falls into Category C-3. The results of an inspection are classified as Category C-3 if more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Summary

An inservice inspection was conducted on the Unit 1 steam generators from November 9 through November 21, 1992. As a result of the eddy current inspection and a leak test on Steam Generator 12, it was found that 5.44% (179 of 3289) of the inspected tubes contained defects requiring repair. Therefore, the results of the inspection of Steam Generator 12 were classified as Category C-3. Twenty-one of these tubes were plugged and the remaining 158 tubes were repaired by installing welded tubesheet sleeves. Repairs were completed on December 2, 1992.

The results of the inspection of Steam Generator 11, where only one defective tube was identified, were Category C-2.

Background

Table 1 provides Prairie Island Nuclear Generating Plant data which is significant for the steam generators. Prairie Island has had several steam generator tube leaks due to secondary side intergrannular attack and stress corrosion cracking (IGA/SCC) as shown in Table 2. At present, secondary side intergrannular attack and stress corrosion cracking has been positively identified in only one of the four steam generators at Prairie Island, steam generator 12. Tube pulls have been conducted, as shown in Table 3, to characterize indications at cold leg tube support plates and in the hot leg tube sheet region.

Cause of Tube Degradation

The major cause of the degradation of tubes in Steam Generator 12 is secondary side intergrannular attack and stress corrosion cracking. This cause was identified by metallurgical examination of three hot leg sections of the Inconel 600 tubing removed from Steam Generator 12 in January, 1985 (Reference 1). The degradation is characterized as single or multiple axial indications. Except for the early years, these axial indications are located in the lower one-half of the tubesheet crevice region.

Rotating pancake coil (RPC) examination of the tube samples, plus experience gained from other utilities, provides a tool to confirm the type of degradation occurring in the tubesheet region. Rotating pancake coil examinations of all tubes with non-quantifiable indications in the tubesheet region has been done routinely since February, 1987. The rotating pancake coil examination results have confirmed the type of degradation as secondary side intergrannular attack and stress corrosion cracking.

During the October, 1992 inspection, 25 tubes with indications representative of primary side stress corrosion cracking (PWSCC) at the roll transition region were also identified.

Apparent Large Increase in the Number of Defective Tubes

The apparent increase in the number of defective tubes identified in Steam Generator 12 compared to previous inspections and compared to the tube leak inspection in September 1992 was due to conducting an examination of the entire hot leg tube sheet region of Steam Generator 12 (tube end to top of tubesheet in all tubes not sleeved or plugged) using the rotating pancake coil. The decision to conduct the 100% examination of the tubesheet region of Steam Generator 12 was primarily driven by the need to do every thing possible to prevent another tube leak outage during the 18 months of operation planed for the next fuel cycle. Since the rotating pancake coil eddy current inspection technique is known to have improved detection because the small pancake coil has a better signal-to-noise ratio in the presence of secondary side extraneous variables, the ability to identify additional defective tubes would reduce the probability of a tube leak outage (Reference 2). However, due to variable chemistry environments and times to initiate cracks, crack growth may still occur in the tubesheet crevice region sufficient to produce tube leakage prior to the end of a cycle, from tubes which appear to be non-degraded at the beginning of the cycle.

It was anticipated that the rotating pancake coil examination would identify some defects based on the experience at other utilities who have performed 100% rotating pancake coil inspections. The actual percentage of defective tubes identified in steam generator 12 by the rotating pancake coil was 5.2%.

Even though there appears to be a large increase in the degradation of Steam Generator 12, we believe this increase is due to the increased sensitivity of the inspection method and not due to any significant increase in the rate of degradation.

Effect on Examination of Steam Generator 11

Due to the high numbers of defective tubes found by the rotating pancake coil examination of Steam Generator 12, a 20 % Rotating pancake coil examination, based on guidelines in Reference 2, was conducted in Steam Generator 11, even though there has not been a valid tubesheet stress corrosion cracking type indication found in Steam Generator 11, to date. No detectable degradation of Steam Generator 11 tubes in the tubesheet region was found by the 20 % Rotating pancake coil examination.

Remedial Actions

Prairie Island has had several steam generator tube leaks due to secondary side intergrannular attack and stress corrosion cracking, as shown in Table 2. Northern States Power has participated in utility funded research on steam generator related issues beginning with the Steam Generator Owners Group II in 1982 and continuing with the EPRI funded Steam Generator Reliability Group. Remedial actions to reduce and/or prevent tube degradation due to secondary side intergrannular attack and stress corrosion cracking are being used by the industry with only limited success. Prairie Island has evaluated, and implemented the remedial actions as described below:

- Reduce Operating Temperature:

  Prairie Island has been a low temperature plant having operated with T_{hot} at 590°F since startup. This has slowed, but not eliminated, growth of intergrannular attack and stress corrosion cracking in the Prairie Island steam generators. Additional temperature reduction has not been warranted.

- Chemistry Control:

  Prairie Island has used state of the art analytical equipment since startup and has followed both the original equipment manufacturer's water chemistry guidelines as well as the EPRI secondary water chemistry guidelines (Reference 3).

  The amounts of material found from hideout return tests during each refueling shutdown have been small. One steam generator is sludge lanced each outage. In recent years, the amounts of sludge removed by sludge lancing have ranged between 30 and 80 pounds per steam generator.

- Conduct Crevice Flushing Operations with Boric Acid:

  Prairie Island started crevice flushing in 1986 on a refueling basis. Boric acid was added to the crevice flushing procedure in 1987. The crevice flushing procedure has been reduced to 24 hours because only a small amount of contaminants are being removed.

- On-line Addition of Boric Acid:

  Following the report of favorable laboratory results in 1986, Prairie Island began on-line addition of boric acid in Unit 1 in March 1987. The effectiveness of this remedial action remains controversial within the industry (EPRI IGA/SCC workshops in May 1991 and December 1992). However, the incidence of intergrannular attack and stress corrosion cracking at Prairie Island has lagged that of other similar units who have not used boric acid. Prairie Island will continue to use boric acid until such time as an inhibitor of equal or greater effectiveness is justified for on-line use. One of the recommended boric acid practices, low power soaks, has not been implemented at Prairie Island.

- Use of Other Chemical Inhibitors:

  At the present time, NSP supports EPRI research for other chemical inhibitors. Our current evaluation centers around the use of titanium compounds to inhibit the growth of intergrannular attack and stress corrosion cracking.

- Use of Preventative Sleeving:

  Preventive sleeving is one method of reducing the probability of tube leak outages. The down side of preventive sleeving is the inability to follow the degradation mechanism and the reduction in the ability to examine tube support plate intersection above the sleeves. We have made the strategic decision to sleeve on an as-needed basis, to insure that we are able to best follow the tube support plate problems and to reduce our overall cost of steam generator repair and maintenance.

- High Hydrazine and Molar Ratio Control:

  These two remedial actions have been used with some success in Japan. Historically, Prairie Island has maintained relatively high hydrazine levels. In May, 1992, feedwater hydrazine control was raised to 150 ppb.

  Molar ratio control is being evaluated at this time.

100% Eddy Current Examinations

Starting in 1982, 100% of the full length of all tubes in service have been examined at Prairie Island each refueling outage. This 100% examination is performed with the bobbin coil and the rotating pancake coil for row 1 U-bends. In addition, all tubes with indications which can not be quantified, such as undefined residual signals and distorted indications in the tubesheet and hot leg tube support plate regions, are examined with the rotating pancake coil probe due to its higher sensitivity. Repair decisions, in those cases, are based on the rotating pancake coil results.

References

1. EPRI NP-4745-LD, Examination of Tubes R4C19HL, R6C18HL, and R16C33HL from Steam Generator 12 of the Prairie Island Nuclear Station Unit 1.

2. EPRI NP-6201, PWR Steam Generator Examination Guidelines, Revision 2.

3. EPRI NP-6239, PWR Secondary Water Chemistry Guidelines, Revision 2.

Table 1: PRAIRIE ISLAND PLANT DATA

Location: On Mississippi River near Red Wing Minnesota

Nuclear Steam Supply System: Westinghouse 2-Loop 560 MWE

Steam Generators: Westinghouse Model 51
Mill-Annealed Alloy 600 Tubing
Open Tubesheet Crevices - 2 3/4" hard roll at bottom of tube

Circulating Water: Mississippi River/Cooling Towers

Secondary Systems Tubing: Stainless Steel/Carbon Steel

Startup Dates : Unit 1 - December 16, 1973
Unit 2 - December 21, 1974

Effective Full Power Days : Unit 1 - 5593 EFPD's
Unit 2 - 5506 EFPD's

HOT LEG TEMPERATURE: 590 degrees Fahrenheit

Table 2: Forced Outages due to Steam Generator Tube Leaks

- Tube Rupture in No. 11 Steam Generator Due to Loose Part - 1979
- Tube Leaks in No. 12 Steam Generator due to IGA/SCC - 1980, 1983, 1984, 1987 and 1992
- Tube Leak in No. 21 Steam Generator due to Plugging Error - 1981

Table 3: Tube Pulls

- January 1980, No. 22 Steam Generator - One Tube for Thinning at Cold Leg Tube Support Plate
- January 1985, No. 12 Steam Generator - Three Tubes for Tubesheet/Hot Leg Tube Support Plate IGA/SCC