NSP

Northern States Power Company

Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

June 17, 1994 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, 35 tubes were plugged for the first time, one tube which was sleeved during this inspection was plugged and 117 tubes were sleeved. The percentage of tubes plugged is 2.45% on steam generator 11. The percentage of tubes plugged (including the percentage plugged equivalent for the sleeved tubes) is 4.89% 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 May 26, 1994. Further information on the results of the Unit 1 steam generator inspections was provided to the NRC Staff by telephone on May 31, 1994 and June 14, 1994. 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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This letter contains no new NRC commitments. If you have any questions concerning this information please call.

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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

Steam Generator No. 11 Summary

New Indications Plugged This Outage:	9
Total Tubes Plugged:	83
Total Tubes Sleeved:	0

11 Steam Generator % Plugged: 2.45%

New Indications:

Nine defective tubes were identified with the following types of degradation:

1. Wastage:

   Four tubes were plugged for thinning at the cold leg tube support plate.

2. Secondary Side SCC/IGA in Tubesheet Region:

   Two tubes were plugged for axial indications in the hot leg tubesheet crevice region.

3. Secondary Side SCC/IGA in Sludge Region at Top of Tubesheet:

   One tube was plugged for an axial indication 2 inches above the hot leg tubesheet.

4. Primary Water Stress Corrosion Cracking:

   Two tubes were plugged for axial indications at the roll transition zone in the hot leg.

Tube Plug Inspection:

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

Tube Plug Removal:

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

Supplementary Tubesheet Rotating Pancake Coil (RPC) Examination:

Secondary side stress corrosion cracking and roll transition zone cracking were identified for the first time in 11 Steam Generator. 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. However, one tube which was not part of this 20% sample (but which was examined by RPC because of a bobbin coil undefined indication in the tubesheet region) did contain axial indications representative of secondary side stress corrosion cracking. A decision was then made to examine the remaining 80% of the hot leg tubesheet region with RPC. Four additional tubes with tubesheet region axial indications were identified during this inspection.

Steam Generator No. 12 Summary

New Indications Plugged This Outage:	26	
New Indications Sleeved This Outage:	118	
Sleeved Tubes Plugged This Outage:	1	(weld equipment failure)
Total Tubes Plugged:	146	
Total Tubes Sleeved:	436	

12 Steam Generator % Plugged + % Sleeved Equivalent: 4.89%

New Indications:

143 defective tubes were identified with the following types of degradation:

1. Secondary Side SCC/IGA in Tubesheet Region:

   110 tubes were plugged or sleeved for axial indications in the hot leg tubesheet crevice region. These single or multiple axial indications in the lower half of the tubesheet crevice region are associated with the secondary side IGA/SCC corrosion occurring in the tubesheet of 12 steam generator.

2. Primary Water Stress Corrosion Cracking:

   Thirty tubes contained axial indications at the roll transition zone in the hot leg. One of these thirty tubes contained both an indication at the roll transition zone and a secondary side IGA/SCC indication. One tube contained an axial indication in the row 1 U-bend region.

3. Wear at Anti-Vibration Bars:

   Two tubes were plugged due to wear indications at the new anti-vibration bar locations.

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 May 26, 1994.

The 30 day special report on the Category C-3 steam generator inspection is provided as Attachment 2 to this submittal.

Supplementary Tubesheet Rotating Pancake Coil (RPC) Examination:

In order to best identify those tubes which have minor degradation in the tubesheet region and which could leak during the next fuel cycle, a complete examination of the tubesheet region of all unsleeved and unplugged tubes in steam generator 12 was conducted. The results of this examination were used to develop the tube repair lists.

Tube Plug Inspection:

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

Tube Plug Removal:

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

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 consisting of 100% full length bobbin coil and 100% of tubesheet region MRPC was conducted on Unit 1 Steam Generator 12 from May 21 through May 28, 1994. As a result of the eddy current inspection 4.38% (143 of 3268) of the inspected tubes in Steam Generator 12 contained defects requiring repair. Twenty-six of these tubes were plugged and the remaining 117 tubes were repaired by installing welded tubesheet sleeves. Less than 10% of the inspected tubes were degraded. Repairs were completed on June 5, 1994.

An inservice inspection consisting of 100% full length bobbin coil and 100% of tubesheet region MRPC was conducted on Unit 1 Steam Generator 11 from May 28 through June 4, 1994. As a result of the eddy current inspection .27% (9 of 3314) of the inspected tubes in Steam Generator 11 contained defects requiring repair. These nine tubes were plugged. The inspection results in Steam Generator 11 were Category C-2. Repairs were completed on June 5, 1994.

Background

Table 1 provides Prairie Island Nuclear Generating Plant data which is significant for the steam generators.

The current status of each steam generator at Prairie Island is shown in the attached Table 2: "Prairie Island Steam Generator Tube Plug and Sleeve Status."

In 1980 and 1985 tube samples were removed 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 (IGA/SCC or ODSCC). 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 (MRPC) of the tube samples plus experience gained from other utilities provides a tool to confirm the type of degradation occurring in the tubesheet region. MRPC examinations of all tubes with non-quantifiable indications in the tubesheet region has been done routinely since February, 1987. The MRPC results have confirmed the type of degradation as secondary side IGA/SCC.

Also, during the October, 1992 and the current inspection, tubes with indications representative of primary water stress corrosion cracking (PWSCC) at the roll transition region have also been identified.

Comparison of Number of Defective Tubes in Steam Generator 12, May 1994 to October 1992

The number of defective tubes identified in Steam Generator 12 compared to the October 1992 has decreased somewhat. In addition, some of the indications identified using the current somewhat larger pancake coil RPC probe, upon review this year with hindsight, were present in the October 1992 inspection. A review of 1992 data for 177 RPC indications identified in 1994 in 12 Steam Generator tubesheet region was done. The results are summarized as follows:

New:	39
Small Change	23
Large Change	41
No Change	74

This indicates that the secondary side stress corrosion cracking growth has not increased.

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 a tube leak outage during the 18 months of operation planned for the next fuel cycle, as well as a comparison to the previous 1992 100% RPC examination.

Effect of 12 Steam Generator Results on Examination of Steam Generator 11

Due to the high numbers of defective tubes found by the RPC examination of Steam Generator 12, a 20 % sample of Steam Generator 11, based on guidelines in Reference 2, was examined using RPC. No indications were found by this 20% sample.

Expansion of RPC Examination to 100% in 11 Steam Generator

Due to one ODSCC indication in the tubesheet region of 11 Steam Generator found by supplemental RPC done for non-quantifiable bobbin coil indications, a decision was made to expand the 20% RPC sample to a 100% RPC examination of the tubesheet of 11 Steam Generator in order to give assurance of a leak free operating cycle and to establish a baseline for future steam generator decisions and inspections. The final result of the RPC examination was:

- 2 tubes with PWSCC at Roll Transition Zone
- 2 tubes with ODSCC in tubesheet crevice region
- 1 tube with ODSCC 2 inches above top of tubesheet in region of hard sludge. (This indication was in R28C30, HL, TSH +2.1 to +2.5 inches and was a Single Axial Indication)

In addition, 4 tubes were plugged due to thinning at the Cold Leg Tube Support plates.

During the May 31, 1994 phone call with the NRC Staff, an indication in Steam Generator 11 R14C47, hot leg was discussed. This was a 42% call at TSH +1.5 inches. This was subsequently reevaluated in a second set of data and reclassified as an Undefined Indication. Subsequent examination by RPC showed No Detectable Degradation at this location.

Indications at Sleeve Welds

Summary

When using rotating I-coil and pancake eddy current probes, volumetric indications were found in the upper weld of 27 Combustion Engineering welded tubesheet sleeves. These indications were characterized as shallow and on the inside surface. The visual examination showed the welds to be acceptable. There were some weld artifacts on the weld surface which could explain the eddy current indications. All of the welds were examined this outage by ultrasonic testing and found to be acceptable. The indications are not due to corrosion since they were found in sleeves from 3 different outages, 1987, 1992, and 1994. All sleeves with these indications were left in service.

Details of Sleeve Examination

A. A new type of eddy current examination was conducted on the Combustion Engineering welded tubesheet sleeves installed in 12 Steam Generator using a new probe called the "I"-coil RPC probe. In the past, only a cross-wound bobbin coil had been used to examine the sleeves. The I-coil probe was a Zetec Model 610ZR with 2 coils, one axially wound and one circumferentially wound. The coils are a slightly larger diameter than normal which enhances the ability to see degradation in the parent tube. Twenty of the 319 installed sleeves contained indications in the region of the upper sleeve weld. The new sleeves installed this outage were also examined with the I-coil RPC probe. Seven of the new sleeves had similar indications. None of these indications were identified by this year's cross-wound bobbin coil examination.

B. The sleeve material is Alloy 690.

C. The attached Table 3 provides the list of sleeves with indications at the weld. The attached Figure 1 is a drawing of the Combustion Engineering Tubesheet Sleeve installed at Prairie Island.

D. The ECT indications identified in the sleeve weld region by the I-coil probe were classified as PWA (possible weld anomaly) or SWI (sleeve weld indication).

E. Each sleeve which had an I-coil indication was also examined by a conventional 3-coil RPC probe, designated as Zetec 620ZR. This probe provides better resolution than the I-coil. All of the indications were classified as VOL (volumetric) by the 3-coil RPC. The indications were also characterized as shallow and located on the inside surface.

F. The indications were all located at the sleeve upper weld. Four of these sleeves were installed in the April 1987 outage and sixteen were installed in October 1992 outage. Seven were installed this outage.

Technical Justification for Repair and Use-as-Is:

A. References

1. Combustion Engineering, CEN-294-P, Prairie Island Steam Generator Tube Repair using Leak Tight Welded Sleeves.

B. Investigation was done using ultrasonic testing and visual examination.

1. Visual inspection per ABBCE Procedure 00000-NSS-061 Rev. 9, *Visual Examination Procedures for Steam Generator Plug Welds and Tube-Sleeve Welds*, of the upper sleeve welds of the subject tubes plus 5 tubes with NDD (No Detectable Degradation) by the RPC probes. All visual examinations were acceptable. There are visual indications which could explain the ECT indications.

2. Ultrasonic examination was done of both the old and new sleeves per ABBCE Procedure 00000-NSS-062 Rev. 9, *Procedure for the Ultrasonic Examination of Steam Generator Tube to Sleeve Upper Welds*. All welds were found acceptable.

3. I-Coil RPC examination was done of three pre-production weld samples to identify similar indications. No indications were found which were similar to those found in the installed sleeves.

4. The original Ultrasonic Examination Results were reviewed. Only one tube contained a UT indication which could explain the ECT indications.

5. Baseline I-Coil RPC Examination of the new sleeves was done. For those with indications, 3-coil RPC was also done. Seven of the 117 new sleeves had indications classified as volumetric, shallow, and on the inside surface.

6. Combustion Engineering was requested to provide assistance in the disposition of the sleeves containing the weld anomalies. They recommended that the sleeves remain in service and stated that "ABB Combustion Engineering is confident that these sleeves meet structural integrity and leak tightness requirements."

7. In a telephone call on Tuesday, May 31, 1994, the NRC Staff requested a nominal size of these indications. Approximate sizes were assigned by Zetec EDDYNET CRACKMAP Analysis. The CRACKMAP program oversizes the flaw due to "look ahead", "look behind", and size of the coil. With the relatively large coil size, the indication is seen by the coil when the coil is some distance from the actual artifact.

   The average size is .17 high by .28 inches in circumferential extent. Again, these are relative numbers only, since a thorough study involving destructive examination of actual sleeve samples containing these indications would be necessary to calibrate these numbers. The nominal width of the inside surface of the weld is 3/16 inch. The minimum average height of the weld at the sleeve-tube interface is 0.080 inches.

C. The sleeves with the welding artifacts, as identified by eddy current RPC probes, are acceptable based on acceptable visual examination, ultrasonic examination, and indication location. There is currently no known primary side degradation which is volumetric in nature. In addition, since these indications are found in the both the oldest and newest sleeve welds, they are not a result of corrosion.

D. Combustion Engineering has been requested to keep NSP appraised of further developments for the examination of sleeve welds. NSP will conduct future examinations of the sleeve welds using RPC probe technology.

Remedial Actions

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 to the present EPRI funded Steam Generator Management Project. Remedial actions to reduce and/or prevent tube degradation due to secondary side intergrannular attack and stress corrosion cracking have been used by the industry with only limited success. Prairie Island has evaluated, and in most cases, implemented the following remedial actions:

- Reduced 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 shutdowns have been small. Steam generators are sludge lanced every other outage on a cycling basis. Thirty-eight pounds was removed from 12 Steam Generator this outage.

- High Hydrazine and Molar Ratio Control:

  These two remedial actions have been used with success in Japan. For a U.S. plant, Prairie Island has maintained relatively high hydrazine levels for a long time. In May, 1992, feedwater hydrazine control was raised to 125 +/- 25 ppb.

  Molar ratio control has been attempted by adjustments to steam generator blowdown resin ratios during the last operating cycle. Molar ratio control is still being evaluated. The object of molar ratio control is to maintain the cation to anion ration (sodium to chloride) at less than one so that free sodium hydroxide can not form in the crevice regions.

- Conduct Crevice Flushing Operations with Boric Acid:

  Prairie Island started crevice flushing in 1986 using two days each outage. Since then we have added boric acid to the crevice flushing procedure. The time has been reduced to 24 hours since only a small amount of contaminants are being removed.

- On-line Addition of Boric Acid:

  Following 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). Improvements in the eddy current technology can make these comparisons difficult, since a different set of tubes would have been identified for everyone if RPC inspections had been available and/or used in the past. 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.

- Use of Other Chemical Inhibitors:

  At the present time, NSP supports EPRI research for other chemical inhibitors. Our current evaluations centers around the use of titanium compounds to inhibit the growth of IGA/SCC. A titanium chelate, TYZOR LA Titanate has been added since January 1994 to Unit 1 steam generators.

- 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. NSP has 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.

- Detailed Inspection Plans:

  Although not a recommendation for remedial actions, but rather a current inspection guideline, 100% of the full length of all tubes in service are routinely examined at Prairie Island. This was started in 1982. In addition, all tubes with indications which can not be quantified, such as UDI's, DSI's, MBM's (in the tubesheet) are examined with the rotating pancake coil probe due to its higher sensitivity. Repair decisions, in those cases, are based on the RPC results. It is anticipated that the tubesheets of both Unit 1 steam generators will be examined 100% by RPC in the future.

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 3.

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 - 5942 EFPD's
(as of December 31, 1993) Unit 2 - 5810 EFPD's

HOT LEG TEMPERATURE: 590 degrees Fahrenheit

Table 2: Prairie Island Steam Generator Tube PLUG & SLEEVE Status

DEFECT TYPE	SG11	SG12	SG21	SG22
Cold Leg TSP Thinning	**31**	17	61	122
Antivibration Bar Wear	27	**3**	9	30
Tubesheet Sec Side IGA/SCC Only	**5**	**469**	0	1
Roll Transition PWSCC Only	**2**	**48**	8	4
RTZ PWSCC and Sec Side IGA/SCC	0	**11**	0	0
Hot Leg Tube Support Plate ?	0	1	0	0
U-Bend PWSCC	1	**1**	0	0
Loose Parts	7	0	2	2
Free Span	**5**	1	2	1
Other	5	5	4	3
Total Tubes Defective	**83**	**556**	86	163
Total Tubes Plugged	**83**	**146**	86	163
Tubesheet Sleeves (IGA/SCC)*	0	* **436**	0	0
% Equivalent Plugged	**2.45%**	**4.89%**	2.54%	4.81%

*Includes 26 preventive sleeves

As of June 5, 1994

Table 3
(Page 1 of 2)

Cumulative Indications Report
Prairie Island Unit 1
Steam Generator 12, Hot Leg

ROW	COL	LEG	EXTENT BEG	EXTENT END	REM	REEL	PROBE	LOCATION		CURRENT VOLTS	CURRENT DEG	CURRENT %	CH
5	48	H	STH	BUH	S	00033	620ZR					NDD	
		H	STH	BUH	S	00021	610ZR	BUH+	0.4	5.07	17	SWI	P3
8	48	H	STH	BUH	S	00016	610ZR	BUH+	0.3	7.21	27	SWI	P3
		H	STH	BUH	S	00033	620ZR	BUH+	0.3	13.35	28	VOL	1
14	50	H	STH	BUH	S	00019	610ZR	BUH+	0.4	6.58	19	SWI	P3
		H	STH	BUH	S	00033	620ZR	BUH+	0.4	14.86	18	VOL	1
15	50	H	STH	TEH	S	00102	640ZX					NDD	
		H	STH	BUH	S	00100	610ZR	BUH+	0.5	26.93	30	SWI	P1
		H	STH	BUH	S	00101	620ZR	BUH+	0.6	22.63	52	VOL	P1
21	51	H	STH	BUH	S	00033	620ZR	BUH+	0.4	22.45	45	VOL	1
		H	STH	BUH	S	00018	610ZR	BUH+	0.5	7.00	38	SWI	P3
7	52	H	STH	TEH	S	00103	640ZX					NDD	
		H	STH	BUH	S	00105	620ZR					NDD	
		H	STH	BUH	S	00104	610ZR	BUH+	0.4	8.40	15	SWI	P1
3	56	H	STH	BUH	S	00018	610ZR	BUH+	0.4	25.36	11	PWA	1
		H	STH	BUH	S	00033	620ZR	BUH+	0.4	10.56	27	VOL	1
8	57	H	STH	BUH	S	00019	610ZR	BUH+	0.5	16.54	11	SWI	P3
		H	STH	BUH	S	00033	620ZR	BUH+	0.5	29.39	23	VOL	1
9	59	H	STH	BUH	S	00019	610ZR	BUH+	0.4	15.02	22	SWI	P3
		H	STH	BUH	S	00033	620ZR	BUH+	0.4	34.91	31	VOL	1
7	63	H	STH	BUH	S	00018	610ZR	BUH+	0.4	18.29	22	SWI	1
		H	STH	BUH	S	00033	620ZR	BUH+	0.5	15.08	19	VOL	1
16	64	H	STH	BUH	S	00018	610ZR	BUH+	0.4	18.18	35	SWI	1
		H	STH	BUH	S	00033	620ZR	BUH+	0.4	13.67	75	VOL	1
7	65	H	STH	TEH	S	00102	640ZX					NDD	
		H	STH	BUH	S	00100	610ZR	BUH+	0.5	6.30	26	SWI	P1
		H	STH	BUH	S	00101	620ZR	BUH+	0.5	30.37	7	VOL	P1
6	66	H	STH	BUH	S	00101	620ZR					NDD	
		H	STH	TEH	S	00102	640ZX					NDD	
		H	STH	BUH	S	00100	610ZR	BUH+	0.4	18.12	5	PWA	P1

Table 3
(Page 2 of 2)

Cumulative Indications Report
Prairie Island Unit 1
Steam Generator 12, Hot Leg

ROW	COL	LEG	EXTENT BEG	END	REM	REEL	PROBE	LOCATION	CURRENT VOLTS	DEG	%	CH
23	67	H	STH	BUH	S	00018	610ZR	BUH+ 0.4	12.84	27	SWI	1
		H	STH	BUH	S	00033	620ZR	BUH+ 0.5	21.23	30	VOL	1

Figure 1

Sleeve Installation
Prairie Island Unit 1
Steam Generator 12