NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Unit 1	0 5 0 0 0 2 8 2	1 OF 0 6

TITLE (4) Unit Shutdown Resulting from Steam Generator Tube Leakage

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 2	2 0	8 7	8 7	— 0 0 2 —	0 0	0 1	2 7	8 8		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 7

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A) Voluntary
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Arne A. Hunstad, Staff Engineer	6 1 2 3 8 8 - 1 1 2 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	A B	S G	W 3 5 1	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 20, 1987, Unit 1 was at 97% power, coasting down to refueling. At 2045, a normal shutdown to cold shutdown was started due to a steam generator primary-to-secondary leak which had increased over a period of several days to 0.33 GPM as measured by xenon activity at the condenser air ejector. Radiochemistry analysis of steam generator blowdown samples on February 20th confirmed that the leak was in No. 12 Steam Generator. On February 23, R16C41 was identified as the leaking tube using a camera mounted in the channelhead. Leakage was 1 drop per 29 seconds at 225 psig nitrogen on the secondary side and 13 drops per minute at 450 psig. A 100% eddy current test program was done; 19 tubes were plugged.

The Prairie Island Westinghouse Model 51 steam generators were fabricated with a 2 1/4 inch rolled region at the bottom of the tubesheet and an open crevice between the tube and tubesheet hole for the remainder of the 22-inch thick tubesheet. Impurities in steam generator bulk water concentrate in the crevice regions and can cause intergranular corrosion of the mill-annealed Alloy 600 steam generator tubing. A caustic crevice environment is the most likely cause of this type of corrosion at Prairie Island.

Technical Specification 3.1.C.6 requires unit shutdown to cold shutdown and an inservice steam generator tube inspection whenever primary-to-secondary leakage exceeds 1.0 gallons per minute. This limit was not exceeded during this event. This report is being provided voluntarily.

IE22

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
	0 5 0 0 0 2 8 2	8 7	— 0 0 2	— 0 0	0 2 OF 0 6

TEXT (If more space is required, use additional NRC Form 366A's) (17)

DESCRIPTION OF THE EVENT

On February 20, 1987, Unit 1 was at 97% power, coasting down to refueling. At 2045, a normal shutdown to cold shutdown was started due to a steam generator (EIIS Identifier SG) primary-to-secondary leak which had increased to 0.33 GPM as measured by xenon activity at the condenser air ejector. Significant event chronology is as follows:

February 17 at 0030: Condenser air ejector monitor (EIIS Identifier MON) 1R-15 reading 70 CPM, which is normal

February 18 at 1015: 1R-15 removed from service due to erratic readings and spiking

at 1030: Air ejector grab sample taken at 0905 reported to Control Room

February 19 at 0440: Air ejector grab sample taken at 0440 reported to Control Room

at 1441: 1R-15 returned to service after replacing the scintillation detector. Reading ~150 CPM.

at 1705: 1R-15 reached alarm setpoint (500 CPM)

February 20 at 0030: 1R-15 reading 400 CPM

at 0900: Steam generator tube leak contingency plans discussed with Operations Committee

in P.M.: 1R-15 setpoint raised to 2000 CPM

at 1935: 1R-15, after stable reading at 400 CPM since Feb. 19, quickly increased to 3000 CPM

at 2045: Began Unit 1 shutdown to cold shutdown; 1R-15 reading ~4000 CPM

at 2055: Steam generator blowdown monitor 1R-19 alarming

at 2100: 1R-15 reading 6190 CPM

at 2336: Unit 1 off line

February 21 at 0111: Unit 1 borated to cold shutdown

at 1703: Unit 1 at cold shutdown

Table 1 provides the 1R-15 readings from the Emergency Response Computer System (ERCS) every 10 minutes from 1600 to 2400 on February 20th. The continuous plotting of 1R-15 readings on the ERCS displays in the control room provided a valuable and vivid diagnostic tool for the operators during this event.

The following data shows the progression of the leak:

Date	Time	Method	Rate
Feb. 7	2359	Equilibrium Tritium Leak Rate	0.0004 GPM
Feb. 15	0400	Equilibrium Tritium Leak Rate	0.0002 GPM
Feb. 18	0905	Instantaneous Gas Leak Rate(Xe)	0.005 GPM
Feb. 19	0440	Instantaneous Gas Leak Rate(Xe)	0.005 GPM
Feb. 19	1715	Instantaneous Gas Leak Rate(Xe)	0.021 GPM
Feb. 20	0415	Instantaneous Gas Leak Rate(Xe)	0.015 GPM
Feb. 20	2000	Instantaneous Gas Leak Rate(Xe)	0.18 GPM
Feb. 20	2100	Instantaneous Gas Leak Rate(Xe)	0.33 GPM

Radiochemistry analysis of steam generator blowdown samples at 2125 on February 20th confirmed that the leak was in No. 12 Steam Generator.

On February 2?, R16C41 was identified as the leaking tube using a camera mounted in the channelhead. Leakage was 1 drop per 29 seconds at 225 psig nitrogen on the secondary side and 13 drops per minute at 450 psig.

CAUSE OF THE EVENT

The Prairie Island Westinghouse Model 51 steam generators were fabricated with a 2 1/4 inch rolled region at the bottom of the tubesheet and an open crevice between the tube and tubesheet hole for the remainder of the 22-inch thick tubesheet. Impurities in steam generator bulk water concentrate in the crevice regions and can cause intergranular corrosion of the mill-annealed Alloy 600 steam generator tubing. A caustic crevice environment is the most likely cause of this type of corrosion at Prairie Island.

Based on the results of the January 1985 tube pull (EPRI Report NP-4745-LD) and the rotating pancake coil data, intergranular attack/stress corrosion cracking in the tubesheet crevice region is the degradation mechanism.

This type of corrosion has been reported in varying forms in steam generators at other plants.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Prairie Island Unit 1	0 5 0 0 0 2 8 2	8 7	— 0 0 2 —	0 0	0 4 OF 0 6

TEXT (If more space is required, use additional NRC Form 366A's) (17)

ANALYSIS OF THE EVENT

The maximum leak rate measured by xenon activity was 0.33 GPM for the 2100 sample. The estimated maximum leak rate based on the air ejector monitor readings occurred at 2130 and is calculated as 0.37 GPM. As power is reduced the leak rate decreased due to increasing steam generator pressure. The location of the leak was in the lower half of the crevice.

The leaking tube had no detectable degradation during the March 1986 Refueling Outage Inspection, both on bobbin coil data and rotating pancake coil data. Degradation in another tube, R9C25, which did have a rotating pancake coil indication from March, had grown, but the tube was not leaking.

1R-15 appeared to behave erratically during the initial leakage increase. The photomultiplier tube was replaced on February 19th. Activity levels did increase during the period of time that 1R-15 was out of service. Grab samples were taken daily which showed increased activity.

Review of the data shows that there may have been an initial leakage increase on February 15, 1987. However, that increase was so small that it would not have generated any concern.

Since very little radioactive material was released, this event presented no additional risk to the health and safety of the public. Total dose to an individual would have been less than 0.0001 mRad from the airborne releases resulting from the leakage.

Technical Specification 3.1.C.6 requires unit shutdown to cold shutdown and an inservice steam generator tube inspection whenever primary-to-secondary leakage exceeds 1.0 gallons per minute. This limit was not exceeded during this event.

This report is being provided voluntarily.

CORRECTIVE ACTIONS

Unit 1 was shut down when primary-to-secondary leakage increased above 0.1 GPM.

A 100% eddy current inspection using a 0.740-inch bobbin probe was done of the tubesheet to 01H region of No. 12 steam generator. R16C41 had multiple eddy current indications from 4.2 to 9.7 inches above the hot leg end of the tube. All tubesheet indications were also examined using the rotating pancake coil probe for further identification and resolution. Nineteen tubes were plugged. One of these pluggable indications was at the top of the tubesheet and the rest were in the lower region of the tubesheet.

Boric acid addition which had been planned to begin following the refueling outage was started, instead, on March 9, 1987.

From research funded by vendors, the Steam Generator Owners Groups, EPRI, and foreign utilities, the following list of remedial actions for steam generator secondary side IGA/SCC has been developed by the Steam Generator Project Office at EPRI.

1. Crevice flushing with boric acid: This is being done.

2. Soaking at operating temperature: This is done only as schedules allow, but based on hideout return data this may be of limited benefit.

3. On-line boric acid addition: This is being done

4. High hydrazine treatment: This is used by some Japanese plants, but has not gained acceptance in the U.S. and is not in use at Prairie Island.

5. Reduced power/temperature operation: This is normally recommended as a last ditch effort to prevent mid-cycle shutdowns. This has been done at other plants. This is not being recommended for Prairie Island at this time.

6. Sleeving: Twenty seven sleeves have been installed in No. 12 steam generator. Installation of 200 sleeves is planned for September 1988.

7. Maintain steam generator chemistry within the Owners' Group guidelines: This is being done.

The steam generator replacement option is also being considered.

Two additional recommendations which were reinforced by the steam generator tube rupture event at North Anna in July are:

1. Increase the required sampling frequency from once per day to once per 8 hour shift when the air ejector monitor is out of service. This will be done.

2. Install N-16 gamma detectors on the main steam lines to give a redundant and earlier identification of steam generator tube leakage. This is being investigated.

ADDITIONAL INFORMATION

Component: Westinghouse Model 51 Steam Generator Serial Number 1102

Steam Generator Tubing: 7/8 inch O. D., 0.050 inch wall thickness, mill-annealed Alloy 600, fabricated at Blairsville.

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
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Prairie Island Unit 1	0 5 0 0 0 2 8 2	8 7	0 0 2	0 0	0 6 OF 0 6

TEXT (If more space is required, use additional NRC Form 366A's) (17)

There have been 3 similar primary-to-secondary tube leaks in No. 12 steam generator originating from intergranular corrosion in the tubesheet crevice region:

July 1980: LER 80-018
December 1983: No LER written
October 1984: LER 84-010

NRC FORM 366A
(9-83)

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

January 27, 1988

10 CFR Part 50
Section 50.73

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

Unit Shutdown Resulting From Steam Generator
Tube Leakage LER 1-87-002

The Licensee Event Report for this occurrence is attached.

This is a voluntary report submitted for the information of the NRC Staff.

David Musolf

David Musolf
Manager - Nuclear Support Services

c: Regional Administrator - III, NRC
Sr Resident Inspector, NRC
NRR Project Manager, NRC
MPCA
Attn: Dr J W Ferman

Attachment

IE22
1/1