NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION, UNIT 1	0 5 0 0 0 3 3 8	1 OF 0 4

TITLE (4)

STEAM GENERATOR TUBE DEFECTS

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	1 4	8 7	8 7	—	0 1 0	— 0 1	0 5	2 4	8 8		0 5 0 0 0
											0 5 0 0 0

OPERATING MODE (9)	6	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 1 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		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
G. E. KANE, STATION MANAGER	AREA CODE 7 0 3 8 9 4 - 5 1 5 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	
X	A B	T B G	W 1 2 0	Y							

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

During the 1987 refueling outage on Unit 1, standard eddy current bobbin probe inspections in the 'A', 'B', and 'C' Steam Generators identified greater than one percent of the initial sample group to be defective. All tubes exhibiting: (1) clear indications of being defective (i.e. greater than 40 percent "thru wall" indication), (2) "distorted indications" at the tube support locations, and (3) indications in the vicinity of the tubesheet that were identified using the 8X1 probe and confirmed with the rotating pancake coil probe, were removed from service. The defects identified in the Steam Generators are reportable pursuant to 10CFR50.73(a)(2)(v)(C) as required by Technical Specification 4.4.5.5.c.

In order to facilitate determination of the nature and cause of the tubesheet indications, two tubes were removed from the 'A' Steam Generator for further nondestructive and destructive examination. Preliminary results of the examinations performed on the two tubes revealed circumferential pressurized water stress corrosion cracking in the expansion transition region of both tubes at the top of the tubesheet.

The health and safety of the general public were not affected during this event.

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

During the 1987 refueling outage on Unit 1, standard eddy current (E/C) bobbin probe inspections in the 'A', 'B', and 'C' Steam Generators (S/G's) identified greater than one percent of the initial sample group to be defective. Accordingly, a prompt report to the NRC was made in compliance with Technical Specification 4.4.5.5.c. Additionally, one hundred percent of the inservice tubes on each S/G were inspected using the bobbin probe, as defined by the pre-established inspection plan and Table 4.4-2 of Technical Specification 4.4.5.2. The inspection results are given in Table 1. The defects identified in the 'A', 'B', and 'C' S/G's are reportable pursuant to 10CFR50.73(a)(2)(v)(C) as required by Technical Specification 4.4.5.5.c.

In addition to standard E/C bobbin probe inspections, E/C inspections were also performed on accessible hot leg tubes, in the tube sheet area, using an 8X1 probe and a rotating pancake coil (RPC) probe. Tubes with indications identified by the 8X1 probe were further inspected with the RPC probe to confirm the indications.

Those tubes exhibiting (1) clear indications of being defective (i.e greater than 40 percent "thru wall" indication), (2) "distorted indications" at the tube support plate locations, and (3) indications in the vicinity of the tubesheet that were identified using the 8X1 probe and confirmed with the RPC probe, were removed from service. A summary of tubes plugged to date is presented in Table 1. No reactor core safety concerns exist since each S/G is bounded by the Safety Analysis tube plugging limit of 12 percent.

In order to facilitate determination of the nature and cause of the tubesheet indications, two tubes were removed from the 'A' S/G for further nondestructive and destructive examination. Destructive and non destructive examinations (NDE) examinations were conducted on the removed tube samples. The NDE included visual and dimensional examination, eddy current inspections, X-ray radiography, and silastic mold observation. The second phase of examination included scanning electron microscopy (SEM), metallography, and burst testing. Preliminary results of the examinations performed on the two tubes revealed circumferential pressurized water stress corrosion cracking in the expansion transition region of both tubes at the tubesheet top location. Minor outside diameter intergrannular corrosion within the first support plate region of one tube and just above the top of the tube sheet region of both tubes was also identified.

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

A demonstration program for the Thermal Stress Relief procedure, to arrest further tube degradation at tube support plates, was conducted during this outage. Partial implementation of the Thermal Stress Relief program is preliminarily scheduled for the Fall 1988 Unit 2 refueling outage. The continued use of Thermal Stress Relief on Unit 1 is currently being evaluated. Tube sleeving, as an alternative to tube plugging, has been successfully utilized in the industry and is being considered as a corrective measure at North Anna Power Station.

The Steam Generator tube inspection testing requirements for the upcoming Unit 1 and 2 refueling outages is under development.

Previous similar events have occurred at North Anna Power Station on Unit 1 during the 1985 refueling outage as reported in LER 85-020-01.

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

TABLE 1

S/G	Clear* Indications	Distorted** Indications	Tube*** Sheet Indications	Total Tubes Plugged This Outage	Previous Plugs	Total Tubes Plugged	Current % Plugged
A	25	43	15	83	126	209	6.17
B	14	37	11	62	116	178	5.25
C	30	78	9	118****	153	271	7.99

* Clear Indications (defective) - greater than 40 percent "thru wall" indications.

** Distorted Indications - tube support plate indications of undetermined "thru wall" depth.

*** Tubesheet indications - not evident on standard eddy current testing which were identified by the 8X1 probe and confirmed by the RPC probe.

**** Includes one tube that was erroneously plugged.

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117

May 24, 1988

U. S. Nuclear Regulatory Commission
Document Control Desk
016 Phillips Building
Washington, D.C. 20555

Serial No. N-87-013A
NO/DEQ: nih
Docket No. 50-338

License No. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following updated Licensee Event Report applicable to North Anna Unit 1. This Licensee Event Report has been updated to include the destructive and nondestructive examination results.

Report No. LER 87-010-01

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very truly yours,

G. E. Kane
Station Manager

Enclosure

cc: U. S. Nuclear Regulatory Commission
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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