Vepco

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

10 CFR 50.73

April 9, 1992

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. N-92-11
NAPS:CSW
Docket Nos. 50-339
License Nos. NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/92-005-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours,

[signature] for
G. E. Kane
Station Manager

Enclosure:

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

Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station

IE22

NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
North Anna Power Station	0 5 0 0 0 3 3 9	1 OF 0 4

TITLE (4) Steam Generator Tube Defects

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 1	9 2	9 2	0 0 5	0 0	0 4	0 9	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 0 0 0

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

☐ 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)	☐ 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 AREA CODE	
G. E. Kane	7 0 3	8 9 4 - 2 1 0 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	H X	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☒ NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

During the 1992 Unit 2 refueling outage, one hundred percent of the accessible tubes in the "A", "B" and "C" Steam Generators (S/Gs) were inspected using the standard eddy current (E/C) bobbin probe. Additionally, augmented inspections were performed using a rotating pancake coil (RPC) probe.

As a result of these inspections, greater than 1% of the tubes in steam generators "A" and "C" were identified as having pluggable indications. These inspection results required the two S/Gs to be classified as Category C-3 in accordance with Technical Specification 4.4.5.2. All tubes with pluggable indications are being removed from service. The defects identified in the S/Gs are reportable pursuant to 10CFR50.73(a)(2)(v)(C) and Technical Specification 4.4.5.5.c. Four hour reports were made pursuant to 10CFR50.72(b)(2)(i) for S/Gs "C" and "A" on March 11, 1992 and March 13, 1992 respectively.

This event posed no significant safety implications since the number of tubes plugged in each steam generator is less than the Safety Analysis tube plugging limit. Therefore the health and safety of the general public was not affected due to this event.

1.0 Description of the Event

During the 1992 Unit 2 refueling outage, steam generator (EIIS System Identifier AB, Component Identifier HX, Vendor Identifier W120) tube eddy current (E/C) inspections were performed using the conventional bobbin probe and the rotating pancake coil probe. These inspections are required by Technical Specification 4.4.5.0. As a result of these inspections, greater than 1% of the tubes in steam generators 'A' and 'C' were identified as having pluggable indications. These inspection results required the two S/Gs to be classified as Category C-3. All tubes with pluggable indications are being removed from service. The defects identified in the S/Gs are reportable pursuant to 10CFR50.73(a)(2)(v)(C) and Technical Specification 4.4.5.5.c. Four hour reports were made pursuant to 10CFR50.72(b)(2)(i) for S/Gs 'C' and 'A' on March 11, 1992 and March 13, 1992 respectively.

Standard E/C bobbin probe inspections were performed on one hundred percent of the hot and cold leg inservice tubes on the three S/Gs. Additionally, augmented inspections were performed using a rotating pancake coil (RPC) probe. These augmented inspections were performed in the hot leg side, for all of the tubes for the top of the tube sheet transition zone, dented tube support plate intersections, selected undented tube support plate intersections and selected U bends. The expanded RPC program on undented intersections encompassed an initial sample on 'A' and 'B' generator resulting in an inspection that included 100% of the open tubes through the second support plate. On 'C' generator, inspection included 100% of the open tubes through the fifth tube support plate. The total number of tubes pluggable as a result of the inspection and maintenance activities in the two steam generator is 45 tubes in "A" S/G and 191 tubes in 'C' S/G. The percentage of tubes determined to be pluggable this outage in the 'A' S/G is 1.39% (45 pluggable/3244 examined). The percentage of tubes determined to be pluggable this outage in the 'C' S/G is 5.96% (191 pluggable/3205 examined). In addition, several indications were circumferential in nature in tube sheet and tube support plate locations.

Those tubes exhibiting (1) clear indications of being defective (i.e., greater than 40 percent "thru-wall" indication) and, (2) circumferential indications in the vicinity of the tubesheet or at the tube support plates that were confirmed using the RPC probe, are being removed from service.

2.0 Significant Safety Consequences and Implications

This event posed no significant safety implications since the number of tubes plugged in each steam generator is less than the Safety Analysis tube plugging limit. Therefore, the health and safety of the general public was not affected due to this event.

3.0 Cause of the Event

Most of the S/G tube degradation is believed to be caused by primary water stress corrosion cracking (PWSCC) and stress corrosion cracking originating in the outside diameter of the tube (ODSCC).

4.0 Immediate Corrective Actions

Tubes identified as defective were plugged and removed from service. Additionally, tubes with clear indications of cracks using the RPC probe, for which a percent through wall extent cannot be determined, are also being removed from service.

5.0 Additional Corrective Actions

An evaluation was performed of the growth rates on the circumferentially oriented indications in several tube sheet and tube support plate locations. The projected end of cycle average crack sizes have been determined acceptable with respect to the structural performance margin established per Regulatory Guide 1.121.

The Technical Specification surveillance requirement for primary to secondary leakage monitoring will continue to be applied. In addition, the conservative primary to secondary administrative leakage limits (50 gpd maximum in any individual steam generator) will continue.

The results of the S/G inspections will be provided in accordance with Technical Specification 4.4.5.5.a.

6.0 Actions to Prevent Recurrence

Steam generator chemistry control and sludge reduction will continue to be used as a means of controlling secondary tube degradation. The Reactor Coolant System programmed Tave (presently 580.8 °F) will continue to be programmed at a level that will reduce the rate of primary water stress corrosion cracking.

7.0 Similar Events

A Unit 2 Steam Generator Inspection resulting in C-3 categorization occurred during the 1990 refueling outage as reported by LER 90-004. Unit 1 Steam Generator Inspections resulting in C-3 categorizations occurred during the 1985, 1987, 1989, 1991 and 1992 outages as reported by LER's

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
North Anna Power Station	0 5 0 0 0 3 3 9	9 2	— 0 0 5	— 0 0	0 4 OF 0 4

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

8.0 Additional Information

Steam generator 'B' was found to have 14 pluggable tubes out of 3253 inservice using the same inspection techinques as for S/Gs 'A' and 'C'. This represents 0.43% of the inservice tubes, below the reportable threshold of 1%.

Unit 1 was not affected by this event.