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

10 CFR 50.73

February 7, 1992

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

Serial No. N-92-01
NAPS:WCH
Docket Nos. 50-338
License Nos. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. 50-338/92-001-00

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,

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 (5-89) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
North Anna Power Station Unit 1	0 5 0 0 0 3 3 8	1 OF 3

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 1	1 0	9 2	9 2	- 0 0 1	- 0 0	0 2	0 7	9 2		0 5 0 0 0
										DOCKET NUMBER(S) 0 5 0 0 0

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 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 - 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)

		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 1992 mid cycle Steam Generator (S/G) tube inspection outage on Unit 1, one hundred percent of the accessible tubes in the "B" and "C" Steam Generators (S/Gs) were inspected using the standard eddy current (E/C) bobbin probe. In "A" S/G, all but one tube (due to tubing restrictions) have been inspected. Actions on the remaining tube are being evaluated to complete the inspection. Additionally, inspections are being performed using a rotating pancake coil (RPC) probe.

As a result of these inspections, greater than 1% of the tubes in each S/G were identified as having pluggable indications. These inspection results required the three S/Gs to be classified as Category C-3. All defective tubes are being removed from service. The defects identified in the S/Gs are reportable pursuant to 10CFR50.73(a)(2)(v)(C). Four hour reports were made pursuant to 10CFR50.72(b)(2)(i).

This event posed no significant safety implications because the primary to secondary leakage rates were closely monitored and were relatively steady and low in magnitude. In addition, a comprehensive Safety Analysis was performed to justify the current operating cycle and to address the potential for S/G tube degradation. This eventually included an assessment of potential indications left inservice. 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 mid cycle Steam Generator (S/G) outage on Unit 1, S/G (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. As a result of these inspections, greater than 1% of the tubes in each of the three S/Gs were identified as having pluggable indications. These inspection results required the three S/Gs to be classified as Category C-3. Accordingly, prompt notifications to the NRC were made in compliance with Technical Specification 4.4.5.5.c and pursuant to 10CFR50.72(b)(2)(i). The defects identified in the three S/Gs are reportable pursuant to 10CFR50.73(a)(2)(v)(C) as required by Technical Specification 4.4.5.5.c.

Standard E/C bobbin probe inspections were performed on one hundred percent of the inservice tubes on "B" and "C" S/Gs. These inspections covered the full length of each tube. In "A" S/G, all but one tube (due to tubing restrictions) have been inspected. Actions on the remaining tube are being evaluated to complete the inspection. Additionally, inspections are being performed using a rotating pancake coil (RPC) probe. These inspections are being performed in the area of the tube support plates. The total number of tubes being inspected in each S/G is 2925 in "A", 2963 in "B" and 2738 in "C".

2.0 Significant Safety Consequences and Implications

This event posed no significant safety implications because the primary to secondary leakage rates were closely monitored and were relatively steady and low in magnitude. In addition, a comprehensive Safety Analysis was performed to justify the current operating cycle and to address the potential for S/G tube degradation. This eventually included an assessment of potential indications left inservice. 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

As an initial corrective action, tubes exhibiting degradation exceeding 40% through wall are being 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 is being performed on the growth rates of the circumferentially oriented indications in several tube sheet and tube support plate locations to substantiate the next period of operation. A Safety Analysis for subsequent operation and any required TS change package will be prepared and approved by the NRC prior to restart.

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

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

The three S/Gs are scheduled to be replaced in 1993.

7.0 Similar Events

Previous similar events have occurred at North Anna Power Station on Unit 1 during the 1985, 1987, 1989 and 1991 refueling outages as reported in LERs 85-020-01, 87-010-01, 89-004-00 and 91-003 respectively.

8.0 Additional Information

Unit 2 was not affected by this event.