10 CFR 50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

October 19, 1993

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

NAPS:AML
Docket No. 50-339
License No. NPF-7

Dear Sirs:

Pursuant to North Anna Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/93-006-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the 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. R. D. Mc Whorter
NRC Senior Resident Inspector
North Anna Power Station

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
North Anna Unit 2	05000 339	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)
										05000
09	21	93	93	006	00	10	19	93		05000

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

OPERATING MODE (9): 6							
POWER LEVEL (10): 0%							
	20.402(b)		20.405(b)		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 (Include Area Code)
Greg Kane, Station Manager	(703) 894-2101

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	AB	HX	A109	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 15 single-spaced typewritten lines) (16)

During the 1993 Unit 2 refueling outage, Steam Generator (S/G) tube inspections were performed using the standard eddy current bobbin probe and rotating pancake coil probe. The results of the tests showed that greater than one percent of the tubes in service in each of the three S/Gs had pluggable indications. Per Technical Specification (TS) 4.4.5.2, "C" S/G was classified as C-3 at 0930 hours on September 21, 1993, "A" S/G was classified C-3 at 1015 hours on September 22, 1993, and "B" S/G was classified C-3 at 1310 hours on September 25, 1993. All tubes with pluggable indications were taken out of service. This event is reportable pursuant to 10 CFR 50.73 (a)(2)(v)(c). Four-hour reports were made to the NRC in accordance with 10 CFR 50.72 (b)(2)(i) and TS 4.4.5.C at 1135 hours on September 21, 1993, for "C" S/G, at 1107 hours on September 22, 1993, for "A" S/G and at 1334 hours on September 25, 1993, for "B" S/G.

The primary cause of the tube degradation is believed to be primary water stress corrosion cracking and stress corrosion cracking originating in the outside diameter of the tube.

No significant safety consequences evolved as a result of the event because the number of tubes plugged in each S/G is below the Safety Analysis tube plugging limit. Therefore, the health and safety of the public were not affected as a result of the event.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 – FACILITY NAME 8 TOTAL – DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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

APPROVED OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
North Anna Unit 2	05000 339	93	— 006 —	00	2 OF 3

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

1.0 Description of the Event

During the 1993 Unit 2 refueling outage, Steam Generator (S/G) (EIIS System Identifier AB, Component Identifier HX, Vendor Identification W120) tube inspections were performed using the standard Eddy Current (E/C) bobbin probe and Rotating Pancake Coil (RPC) probe. The tests revealed that greater than one percent of the inservice tubes in each of the three S/Gs had pluggable indications. Per Technical Specification (TS) 4.4.5.2, "C" S/G was classified as C-3 at 0930 hours on September 21, 1993, "A" S/G was classified C-3 at 1015 hours on September 22, 1993, and "B" S/G was classified C-3 at 1310 hours on September 25, 1993. All tubes with pluggable indications were taken out of service. This event is reportable pursuant to 10 CFR 50.73 (a)(2)(v)(c). Four-hour reports were made to the NRC in accordance with 10 CFR 50.72 (b)(2)(i) and TS 4.4.5.C at 1135 hours on September 21, 1993, for "C" S/G, at 1107 hours on September 22, 1993, for "A" S/G and at 1334 hours on September 25, 1993 for "B" S/G.

Standard E/C bobbin probe and RPC probe inspections were performed on 100% of the available S/G tubes. The E/C bobbin probe inspections included the full length of each tube, while the RPC probe inspections were performed in the hot leg tubesheet areas. Additionally, augmented inspections were performed using the RPC probe for (1) confirmation of E/C bobbin probe calls, (2) selected U bends, and (3) hot leg tube support plate areas. Tubes exhibiting either greater than 40% "thru-wall" indications or circumferential indications were removed from service.

2.0 Significant Safety Consequences and Implications

No significant safety consequences evolved as a result of the event because the number of tubes plugged in each S/G is below the Safety Analysis tube plugging limit. Therefore, the health and safety of the public were not affected as a result of the event.

3.0 Cause of the Event

The primary cause of the tube defects is believed to be primary water stress corrosion cracking (PWSCC) and stress corrosion cracking originating in the outside diameter of the tube (ODSCC).

4.0 Immediate Corrective Actions

All of the tubes with pluggable indications were removed from service.

5.0 Additional Corrective Actions

An Engineering evaluation of the current S/G plugging levels has been performed to substantiate the next cycle of operation.

The TS 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.

The results of the S/G inspection will be provided to the NRC in accordance with TS 4.4.5.5.a.

6.0 Actions to Prevent Recurrence

S/G chemistry control and sludge reduction will continue to be used as a means of controlling secondary tube degradation.

The Reactor Coolant System programmed T_{AVG} will continue to be programmed at a level that will reduce the rate of PWSCC.

7.0 Similar Events

LER 50-339/90-004-00 documents a C-3 condition on Unit 2 "A" and "C" Steam Generators.

LER 50-339/92-005-00 documents a C-3 condition on Unit 2 "A" and "C" Steam Generators.

8.0 Additional Information

Unit 1 was operating at 100% power (Mode 1), and was not affected during the event.