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:

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

Mr. R. D. Mc Whorter 26-107 NRC Senior Resident Inspector North Anna Power Station

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No significant safety consequences evolved as a result of the event because the numbe 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	ABSTRA Duri perf The in e 4.4. class 1310 of s repo at 1 for The corr the No s of	ACT (I ng t orme resu ach 5.2, sifi hou ervi 135 "A" pri cosic tube sign:	Limit is the is ad us of t "C" ted () ted ()	a 1400 1993 of ti be the the S/G C-3 a mad son Sep This a mad cau rackin ant s lugge	spaces, I Unit 2 the sta he tes hree S was cl t 1015 ptember s event e to t Septer at 1334 se of ng and afety ed in	e, appro l refue andard ts sho /Gs ha lassifi hours r 25, t is re he NRC mber 2 4 hours the t stress conseq each and	wimate aling eddy wed d plu ied a on 1993. aport in 1, 19 s on ube s cor s/G	ny 15 sm outa curr that uggabl s C-3 Septe accor 993, 1 Septe degra crosic es ev is b	ngle-spac age, St ent bo greate le ind: at 09 mber 2 l tube: pursua dance for "C' adation on crac olved elow t	team bbin r th lcat 30 h 2, 1 s wi 12, 1 s wi 5, 1 s king as a the	Geni pro ions ions 993, th p. 0 10 G, at 993, bel 993, th cri safe	en ines)(76) erator (S be and rot one percen . Per Tec on Septer and "B" luggable : CFR 50.7 CFR 50.7 CFR 50.72 t 1107 how for "B" lieved to .ginating ult of the ety Analy	/G) tati totati chni mber S/G (b urs S/G. be in t e ev sis fect	tube ng pa f the cal S 21, was (2)(2)()(2)()(2)()(2)() on Se prim the out tube	in nca tu pec 19? cla ns v) i) pt (ary ary cla	aspectic ake coi ubes in cificat 93, "A" assified (c). F and TS ember 2 y waten ide dia ause th lugging	ons l pi sei S/(d C- aker 4. 2, : c st met e n g 1: t of	wer cobe (TS wa -3 a lou -hou 1.5. 1991 cress ar (umb(imin f +1)	e

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
B	UP TO 18 - FACILITY NAME 8 TOTAL - DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FAGILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	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	CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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
в	UP TO 18 - FACILITY NAME 8 TOTAL - DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
3	1	OPERATING MODE
10	3	POWER LEVEL
11.	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	CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

NRC FORM 306A		APPROVED OMB NO. 3150-0104 EXPIRES 5/31/85									
Ľ	ESTIMATED BURDEN PER REBPONSE TO COMPLY WITH THIS INFORMATE C.X.LECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURD ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH SHI 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20084-04 AND TO THE PAPERWORK REDUCTION PROJECT (3180-0104), OFFICE MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.										
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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.

U.S. NUCLEAR REGULATORY COMMISSION |

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

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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 (MINBS 7714), U.S. NUCLEAR REQULATORY COMMISSION, WASHINGTON, DC 20565-0001, AND TO THE PAPERWORK REDUCTION PHOJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDDET, WASHINGTON, DC 20503.

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NRC FORM 366A

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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 TAVG 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.