NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

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

TITLE (4)

Loss of RHR - Failed Level Indication

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 0 0 0
1 0	1 6	8 4	8 4	— 0 0 8	— 0 0	1 1	1 4	8 4		0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
5	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0 0 0	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)	X 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
E. Wayne Harrell	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	I S V	G 2 5 5	N							

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)

ABSTRACT

On October 16, 1984, with North Anna Unit 2 in Mode 5 a complete loss of Residual Heat Removal (RHR) capability occurred when both RHR pumps were unable to operate due to the introduction of air into the RHR system. The incident occurred during the drain down of the Reactor Coolant System (RCS), when the level of the RCS was being monitored via a standpipe off the centerline of one of the RCS loops. The isolation valve to which the standpipe was attached became clogged sometime during the drain down and falsely indicated 64 inches above centerline when in fact the level was below the RHR suction line (below centerline). Subsequently, letdown from the RCS was isolated and makeup initiated. RHR capability was regained 2 hours after initiation of the event. RCS level indication was moved to an alternate tap off loop centerline and indicated satisfactorily.

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NRC Form 366
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
North Anna Power Station	0 5 0 0 0 3 3 9	8 4	— 0 0 8	— 0 0	0 2 OF 0 2

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

On October 16, 1984, with North Anna Unit 2 in Mode 5 a complete loss of Residual Heat Removal (RHR) (EIIS System Identifier BO) capability occurred when both RHR pumps were unable to operate due to the introduction of air into the RHR system. The incident occurred at 1153 hours during the drain down of the Reactor Coolant System (RCS) (EIIS System Identifier AB). The level of the RCS was being monitored via a standpipe off the centerline of one of the RCS loops when the event occurred. The isolation valve (EIIS Component Identifier ISV) to which the standpipe was attached became clogged sometime during the drain down and falsely indicated 64 inches above centerline when in fact the level was below the RHR suction line (below centerline). Upon introduction of air, the "A" RHR pump began cavitating. The redundant pump was started and also experienced cavitation. A RCS level problem was assumed due to previous RHR pump cavitation problems. Letdown from the RCS was secured and makeup initiated to restore level. During the make up process, venting of the RHR system was conducted and RHR returned to service at 1355 hours. The standpipe level indication was moved to an alternate tap off loop centerline and indicated 12" above centerline. The clogged valve was closed, blank flanged and will be repaired during the next refueling outage.

Since this event occurred at the end of the Unit 2 refueling outage, the core decay heat load was relatively low. Reactor core exit temperature did increase from approximately 90 to 125 degrees Fahrenheit during this event. Since the RHR system at North Anna is independent of the Low Head Safety Injection System, the LHSI System could have been aligned to supply alternate cooling.

Vepco

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

November 14, 1984

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

Serial No. N-84-027
NO/CLF: 11
Docket No. 50-339

License No. NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following License Event Report applicable to North Anna Unit No. 2.

Report No. LER 84-008-00

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,

E. Wayne Harrell
Station Manager

Enclosures (3 copies)

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 2900
Atlanta, Georgia 30303

IE22
1/1