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 3

TITLE (4): Reactor Trips From Low Steam Generator Level

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 3	8 4	8 4	— 0 0 1 —	0 0					0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	[X] 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)
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	B A	F C V	L 2 0 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 13, 1984, with Unit 2 at 100 percent power, a Reactor Trip-Turbine Trip occurred at 1530 due to the receipt of an "A" Steam Generator Feed Flow-Steam Flow Mismatch coincident with a Low Steam Generator Level signal. Plant parameters warranted the trip due to the closure of the "A" Main Feed Regulating Valve.

The cause of the closure of the Main Feed Regulating Valve was an instantaneous loss of power to "A" Steam Generator Feedwater Control System cards in Process Rack 6. The supply power loss was a result of the reinstallation of the redundant power supply. The loss of power caused the local manual-auto valve station in the Control Room to switch to manual with zero demand which subsequently closed the "A" Main Feed Regulating Valve.

A second Reactor Trip occurred at 1702 due to low-low water level in the "C" Steam Generator. The reactor was not critical at the time of the trip.

A third Reactor Trip occurred from 1% power at 2036 due low-low water level in the "C" Steam Generator. Both trips were due to manual feedwater control difficulties at low power.

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

On March 13, 1984, with Unit 2 at 100 percent thermal power, a Reactor Trip-Turbine Trip occurred at 1530 hours due to the receipt of an "A" Steam Generator Feed Flow-Steam Flow Mismatch coincident with a Low Steam Generator Level signal. Actual plant parameters warranted the trip due to the closure of the "A" Main Feed Regulating Valve (EIIS Identifier FCV). Subsequent to the trip, Steam Generator level control was regained via the "A" Feed Regulating Bypass Valve and Auxiliary Feed, all plant parameters remained normal for the post trip condition and within bounds of the safety analysis.

The cause of the closure of the "A" Main Feed Regulating Valve was an instantaneous loss of supply power (EIIS Identifier EJ) to "A" Steam Generator Feedwater Control System (EIIS Identifier JB) cards in Process Rack 6. To ensure an uninterrupted power source to Process Rack 6, two independent direct current power supplies one rated at 26 volts D.C., and another at 24 volts D.C. are provided. The 26 volt D.C. power supply had failed earlier in the day and had been replaced with a spare. During the final steps of the procedure when the power supply output fuses (EIIS Identifier FU) were reinstalled in the rack a momentary loss of D.C. supply power occurred. The instrument technicians involved observed that the indicator lights for power available to the cards flickered. The instantaneous loss of power caused the local manual-auto valve station in the Control Room to switch to manual with zero demand which subsequently closed the "A" Main Feed Regulating Valve. Although the Control Room Operator immediately responded by attempting to reopen the valve in the manual mode, the response time of the controller when at zero demand precluded opening prior to the unit trip.

After the full power trip, all three auxiliary feedwater pumps started on low-low steam generator level. Once the level in the "A" Steam Generator was restored, the operator discovered that the discharge valve (EIIS Identifier FCV) would not go closed to throttle the flow from the steam driven pump. In order to stop the flow from this pump, the operator closed the steam inlet valves to the pump. To further reduce the feedwater flow on all three steam generators (which were all over 55% at this point and increasing) the main feedwater lines were manually isolated. The level in the "A" Steam Generator began to decrease after reaching approximately 75%. The "B" and "C" Steam Generators were at 58% level and decreasing at this point. All three steam generators continued to decrease in level during the next 20 minutes. As the levels approached 30%, the manual isolation valves were opened about ½ turn at a time in order to slowly re-establish normal feedwater flow and maintain steam generator level. At 1702, the "C" Steam Generator dropped below the low-low level reactor trip setpoint of 18% causing a Reactor Trip. The reactor was not critical at the time of the trip and only the two shutdown banks were withdrawn. The "B" Steam Generator was at 22% at the time of trip.

The discharge valve for the steam driven auxiliary feedwater pump (MOV-FW-200D) was returned to service at 1732. The valve would not close due to the torque switch contacts remaining open with the valve fully open. The contacts were burnished to ensure good electrical conductivity and the valve was stroked and timed satisfactorily.

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

At 1840 the reactor was critical with all systems returned to normal startup conditions. At 1841 a loss of normal feedwater, condensate and condenser vacuum occurred due to high-high level in the 5B Feedwater Heater (EIIS Identifier HX). This high level was later determined to be caused by two leaking tubes. The 5B Feedwater Heater level was only above the high-high level setpoint for less than 30 seconds. Normal feedwater was restored at 2010. MOV-FW-200D again failed to close and was tagged out at 2019. During the process of restoring normal feedwater, the main feedwater lines were isolated again. When auxiliary feedwater was secured at 2010, the level in all three steam generators began to drop. The main feedwater lines were being unisolated slowly in an attempt to maintain steam generator level. At 2036, the "C" Steam Generator level dropped below the low-low level trip setpoint and caused a reactor trip. The reactor was operating at approximately 1% power prior to the trip. All systems functioned normally except that MOV-FW-200D was still tagged out. Loose contacts were found on the torque switch of MOV-FW-200D. The contacts were tightened and the valve was again stroked several times and returned to service satisfactorily at 2111.

A review of the operation of the trip systems actuations was conducted after the trip via the sequence of events recorder output and computer printouts. All equipment performed as required except MOV-FW-200D as discussed above.

Vepco

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

April 5, 1984

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

Serial No. N-84-004
NO/CLF: 11
Docket No. 50-339

License No. NPF-7

Dear Sirs:

Pursuant to North Anna Power Station Technical Specifications, the Virginia Electric and Power Company hereby submits the following License Event Report applicable to North Anna Unit No. 2.

Report No. LER 84-001-00

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
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