10CFR50.73

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

December 23, 1999

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

Serial No.: 99-607
NAPS: JHL
Docket No.: 50-339
License No.: NPF-7

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/99-004-00

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,

WRMatthews

W. R. Matthews
Site Vice President

Commitments contained in this letter: None

Enclosure

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

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NRC FORM 366 (6-1998) | U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (
MANUAL REACTOR TRIP DUE TO LOSS OF FEEDWATER PUMP SUCTION PRESSURE AND AUXILIARY FEEDWATER ACTUATION

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 NAME	DOCUMENT NUMBER 05000-
12	02	1999	1999	004	00	12	23	1999	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	98 %		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
W. R. Matthews, Site Vice President	(540) 894-2101

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE	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)**

On December 2, 1999, at 1647 hours, with Unit 2 in Mode 1 operating at approximately 98% power, the reactor was manually tripped due to the loss of two of three main feedwater pumps. The main feedwater pumps tripped on low suction pressure due to the loss of discharge flow from the "A" high pressure heater drain pump following a secondary plant transient. The auxiliary feedwater pumps automatically actuated following the reactor trip to restore inventory in the steam generators. This event is reportable pursuant to 10CFR50.73 (a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).

The cause of the event is attributable to the failure of a feedwater heater level control switch to properly actuate. A root cause evaluation of the event is being performed.

This event posed no significant safety implications. The reactor was safely shut down and ESF equipment operated properly. The health and safety of the public were not affected.

1.0 DESCRIPTION OF THE EVENT

On December 2, 1999, at 1647 hours, with Unit 2 in Mode 1 operating at approximately 98% power, the reactor was manually tripped due to the loss of the "A" and "B" main feedwater pumps (2-FW-P-1A and B) (EIIS System - SJ, Component - P). The "C" main feedwater pump (2-FW-P-1C) remained in operation during the event.

The initiation of the event (occurring with Unit 2 at 100% power) was the closure of the "1A" feedwater heat exchanger (2-FW-E-1A) (EIIS Component - HX) cascade valve (2-SD-LCV-203A) (EIIS System - SN, Component - LCV) to the "2A" feedwater heat exchanger (2-FW-E-2A). This resulted in a 2-FW-E-1A Hi/Hi level alarm and a reduction in main feedwater pump suction pressure. The third condensate pump (2-CN-P-1C) (EIIS System - SD) was previously tagged, due to concern for a potential failure of the suction expansion joint (EIIS Component - EXJ) which was visibly deformed, and was unavailable to completely restore feedwater suction pressure. The Hi/Hi level in the 2-FW-E-1A resulted in the closure of the first point extraction steam motor-operated valve (2-ES-MOV-201A) (EIIS System - SE, Component - V) causing cooler water to enter the steam generators (EIIS System - AB). The highest power level indicated by nuclear instrumentation was 101.2%. A ramp down to approximately 98% power was implemented. At this point, the "1A" feedwater heat exchanger cascade valve (2-SD-LCV-203A) was manually isolated.

While investigating the closure of 2-SD-LCV-203A, a 2-FW-E-2A Hi/Hi level alarm was received when the level switch cover was removed. The second point extraction steam motor-operated valve (2-ES-MOV-202A) isolated due to the Hi/Hi level alarm in the "2A" feedwater heat exchanger. A ramp down of 1% per minute was initiated. Approximately one minute later, 2-FW-P-1A tripped on low suction pressure and the "1B" main feedwater pump (2-FW-P-1B) automatically started. A 4% percent per minute ramp down was initiated. The "A" high pressure heater drain pump (2-SD-P-1A) (EIIS System - SN), which pumps heater drains into the feed pump suction line, was then manually stopped due to low and fluctuating discharge pressure. Main feedwater pump 2-FW-P-1B subsequently tripped on low suction pressure. At 1647 hours, the reactor was manually tripped due to the loss of two of three main feedwater pumps on low suction pressure. The auxiliary feedwater pumps automatically started following the manual reactor trip and were secured upon restoration of steam generator level. The "C" main feedwater pump remained in service during the event to provide feed flow to the steam generators. The steam dumps opened to the condenser (EIIS Component - COND) to remove excess heat.

Control Room personnel responded to the reactor trip in accordance with procedure 2-E-0, Reactor Trip or Safety Injection. Primary plant response was normal for this transient. Initially, Reactor Coolant System (RCS) pressure decreased to approximately 1970 psig,

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TEXT (If more space is required, use additional copies of NRC Form 366A) **(17)**

pressurizer level decreased to 24%, and RCS temperature decreased to approximately 546 degrees F. Pressurizer pressure, level and RCS temperature subsequently returned to their normal programmed values. All ESF equipment responded as designed.

On December 2, 1999, at 1811 hours, a four hour report was provided to the NRC per 10 CFR 50.72 (b)(2)(ii) due to the reactor trip and auxiliary feedwater pump actuation. This event is reportable pursuant to 10CFR50.73 (a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications. The reactor was safely shut down and ESF equipment operated properly. Therefore, the health and safety of the public were not affected at any time during this event. The ability to remove heat by normal secondary plant equipment remained in service during this event.

3.0 CAUSE

The cause of the event is attributable to the failure of a feedwater heater level control switch to properly actuate. A root cause evaluation of the event is being performed.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the reactor trip, Operations Emergency Procedure 2-E-0, Reactor Trip or Safety Injection was entered and immediate actions were performed to bring Unit 2 to a safe stable condition. The post trip response progressed as expected and the operators transitioned to 2-ES-0.1, Post Trip Recovery. Unit 2 was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A "Post Trip Review" meeting was conducted with station personnel, on December 2, 1999, to identify the cause of the event, identify abnormal or degraded indications occurring during the reactor trip, and to assess Unit readiness for return to operation.

Equipment malfunctions were appropriately dispositioned and Unit 2 was returned to power operations on December 3, 1999. A root cause evaluation of the event is being performed. Following completion of the root cause evaluation, any additional recommended corrective actions will be evaluated and implemented, as required.

6.0 ACTIONS TO PREVENT RECURRENCE

None.

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TEXT (If more space is required, use additional copies of NRC Form 366A) **(17)**

7.0 SIMILAR EVENTS

None.

8.0 ADDITIONAL INFORMATION

Unit 1 was at 100% power and not affected by this event.