10CFR50.73

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

May 30, 2003

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 03-341
NAPS: JHL
Docket No.: 50-338
License No.: NPF-4

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Power Station Unit 1.

Report No. 50-338/2003-003-00

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,

DAHeal

D. A. Heacock, Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
Atlanta, Georgia 30303-8931

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

IE22

NRC FORM 366 (7-2001) 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 7-31-2004

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Manual Reactor Trip Due to Loss of Electro-Hydraulic Control System Pressure

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-
04	19	2003	2003	-- 003 --	00	05	30	2003	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	1
POWER LEVEL (10)	74 %

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. A. Heacock, 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
X	TG	V	F130	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)

On April 19, 2003, at 2008 hours, with Unit 1 operating in Mode 1, at 74 percent power following a refueling and reactor vessel head replacement outage, a manual reactor trip was initiated. The manual reactor trip was initiated due to a loss of turbine electro-hydraulic control (EHC) system pressure which caused turbine control valves to drift shut. The cause of the EHC low pressure was a diaphragm failure on the auto stop oil side of the turbine auto stop oil EHC interface valve. At 2323 hours, a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B). An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A). This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A) for a condition that resulted in actuation of any engineered safety feature including the reactor protection system. This event posed no significant safety implications because the Reactor Protection System and Engineered Safeguards Features System functioned as designed. Therefore, the health and safety of the public were not affected by this event.

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

1.0 DESCRIPTION OF THE EVENT

On April 19, 2003, at 2008 hours, with Unit 1 operating in Mode 1, at 74 percent power following a refueling and reactor vessel head replacement outage, a manual reactor trip was initiated due to a loss of turbine electro-hydraulic control (EHC) system (EIIS System TG) pressure which caused turbine (EIIS System TA) control valves (EIIS Component V) to drift shut. Operators initiated a manual reactor trip (EIIS System JC) to mitigate the transient and in anticipation of an eventual turbine trip-reactor trip signal from the turbine throttle valve closure or steam generator (EIIS System AB, Component SG) low level reactor trip signal. The reactor was manually tripped and the turbine trip initiation signal was "Reactor Trip/Turbine Trip". There were no structures, systems, or components that were inoperable at the start of the event that contributed to the event. A description of the event is provided below.

Unit 1 was being returned to 100% power following a refueling and reactor vessel head replacement outage. Unit 1 power had been stabilized at 74 percent power while incore flux mapping was in progress. At 2006 hours, the Unit 1 operator at the controls (OATC) received an EHC system low pressure alarm. Shortly thereafter, an EHC system low level alarm was received. The operator also noted that the standby EHC pump, 1-TM-P-3, had automatically started. Field operators were dispatched to investigate the reason for the EHC low pressure condition. Operators in the vicinity of the high pressure turbine enclosure noted auto stop oil coming out of the enclosure doors. Investigation inside the enclosure determined that the EHC to auto-stop oil interface valve, 1-EH-TV-100, was leaking oil from the valve actuator bonnet. Local attempts to stop the leakage were unsuccessful. Within moments of dispatching the operators to investigate, the control room team noted that the turbine throttle valves #2 and #4 were closing. In addition, the alarms for steam flow – feed flow mismatch with low steam generator levels were received. The OATC recommended a manual reactor trip be initiated due to the plant transient and concurrence was received from the senior reactor operator.

Control room personnel responded to the reactor trip in accordance with the Emergency Procedure, 1-E-0, Reactor Trip or Safety Injection. All automatic safety features responded as expected. The post trip response progressed smoothly and within approximately 4 minutes the operations crew transitioned to 1-ES-0.1, Reactor Trip without Safety Injection.

Following the reactor trip, the Reactor Coolant System (RCS) (EIIS System AB) pressure increased to approximately 2301 psig due to the loss of load caused by the #2 and #4 throttle valves going closed. This caused the output of the Master Pressure Controller, 1-RC-PCV-1444J, (EIIS Component PCV) to rise above the Power Operated Relief Valves (PORVs) (EIIS Component RV) lift setpoint of 92.5 percent, causing Pressurizer PORV, 1-

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RC-PCV-1455C, to open for approximately 2 seconds to reduce RCS pressure. When RCS pressure decreased below the PORV lift setpoint, the Pressurizer PORV closed as designed. This pressure increase is expected following a loss of load from 74 percent power. RCS pressure then decreased to approximately 1999 psig and RCS temperature decreased to approximately 544.8°F, before recovering to the “no-load” value of 547°F and a pressure of 2235 psig. The steam dumps (EIIS System SB, Component TCV) functioned normally in T_{avg} mode. This pressure drop is expected during a reactor trip due to the temperature-dependent shrink of the primary system following the reactor trip. Pressurizer level dropped to approximately 24.8 percent before recovering to its no-load value of approximately 28 percent. This level drop is normal and expected—a consequence of the temperature-induced shrinkage following any reactor trip. Recovery of pressurizer level was within the capability of the normal letdown and charging alignment (EIIS System CB/BQ).

A non-emergency four-hour report was made to the NRC Operations Center, on April 19, 2003, at 2323 hours, pursuant to 10CFR50.72(b)(2)(iv)(B) for any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical. During the event, the accident mitigation system actuation circuitry (AMSAC) and the auxiliary feedwater (AFW) system (EIIS System BA) actuated as designed. A non-emergency 8-hour notification was also made to the NRC, at 2323 hours, on April 19, 2003, in accordance with 10CFR50.72(b)(3)(iv)(A) for any event or condition that results in valid Engineered Safety Function (ESF) (EIIS System JE) actuation.

Unit equipment responded as expected with a few discrepancies. The discrepancies included:

The “A” main feedwater pump’s, 1-FW-P-1A, (EIIS System SJ, Component P) amps indicated low (170 amps) and feedwater header pressure decreased after closing the discharge motor operated valve (MOV) (EIIS Component MOV) for the "C" main feedwater pump, 1-FW-P-1C, in preparation for securing the pump to reduce secondary system pressure. Pump amps remained low after re-opening the discharge MOV for the "C" main feedwater pump. The “A” main feedwater pump was secured. Testing was subsequently performed to ensure the disc had not separated from the stem for the “A” main feedwater pump discharge valve. No abnormalities were noted and the pump was returned to service.

Several secondary relief valves lifted immediately following the trip. Most of the valves reseated soon after lifting; however, 1-SV-RV-111A - 1A Feedwater Heater Tube Side and 1-SV-RV-111B – 1B Feedwater Heater Tube Side relief valves lifted and stayed open for an extended time. These valves did not reseat after system pressure was reduced below the lift setpoint. These relief valves were subsequently replaced.

Main turbine journal bearing #7 vibrations as indicated in the Main Control Room were

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erratic. Vibrations indicating as high as 14 mils following the reactor trip. Local vibration measurements for the #5, #6 and #7 bearings were 1.1 mils, 1.2 mils and 0.3 mils respectively. The main turbine journal bearing #7 vibration monitor, 1-TM-VBM-105, was replaced.

Intermediate range nuclear instrumentation detector N36 indication (EIIS System JD, Component DET) decreased appropriately following the reactor trip; however, as power continued to decrease near the end of its indication (below the P6 setpoint), N36 showed signs of being undercompensated. Compensating voltage was adjusted satisfactory and N36 was returned to service operable.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications because the reactor protection system and ESF systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

This event is being reported pursuant to 10 CFR 50.73 (a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature including the reactor protection system.

3.0 CAUSE

The cause of the EHC low pressure was a diaphragm failure of the auto stop oil side of the turbine auto stop oil – EHC interface valve. Investigation revealed that the diaphragm tore at two bonnet bolts 180 degrees apart from each other. This was a new diaphragm replaced during the refueling outage and the only tears were in the vicinity of the bolt holes. The center region of the diaphragm was in good condition. The tear around the bolt holes caused a leak path for auto stop oil. Increased oil leakage from the top of the diaphragm caused auto stop oil pressure to decrease, allowing the interface valve to unseat and begin dumping EHC fluid. This caused the EHC pressure to drop and the eventual need to manually trip the unit. The cause of the tear in the diaphragm at the bolt hole is under investigation.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the event in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Control Room personnel stabilized the plant using ES-0.1 Reactor Trip Recovery. All safety systems responded appropriately. The unit was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Post-Trip review was conducted on April 19, 2003, at 2235, hours by station management with the control room team.

The failed diaphragm was replaced and the valve's operation was verified satisfactory prior to returning the unit to service.

The North Anna Unit 2 turbine auto stop oil – EHC interface valve, 2-EH-TV-200, was inspected for similar concerns associated with the diaphragm. The inspection identified that the diaphragm also appears to be slightly distorted at similar locations. The valve has been operating satisfactorily since maintenance was last performed during the 2001 Unit 2 refueling outage. A non-routine surveillance was initiated to monitor the condition of the valve.

6.0 ACTIONS TO PREVENT RECURRENCE

A root cause evaluation is being performed to investigate the cause of the event. Corrective actions will be performed as necessary following completion of the evaluation to prevent recurrence.

7.0 SIMILAR EVENTS

LER 50-338/86-002-00 documents a reactor trip from 100 percent power generated by a low-low level in "B" steam generator caused by closure of the turbine governor valves. The closure of the turbine governor valves was attributed to problems associated with the control system.

LER 50-338/89-014-00 documents a reactor trip from 90 percent power due to loss of electro-hydraulic control (EHC) system pressure. The turbine trip solenoid operated valve 20-ET o-ring failed.

LER 50-338/89-017-00 documents a reactor trip from 7 percent power generated by a low-low level in "B" steam generator caused by electro-hydraulic control (EHC) system pressure transients. The EHC system pressure transient was caused by leaking turbine overspeed protection circuitry (OPC) valves.

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8.0 ADDITIONAL INFORMATION

Component failure information:

Description: Type 655 actuators are pressure-actuated, spring-and-diaphragm actuators used in conjunction with various valves to provide control for a wide variety of pressure regulation applications.

Manufacturer: Fisher-Rosemount

Model No.: Type 655-D

North Anna Unit 2 was in Mode 1 at 100 percent power and was not affected by this event.