10CFR50.73

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

June 2, 2000

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

Serial No.:	00-277
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 Unit 1.

Report No. 50-338/2000-004-00

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,

D. A. Heacock
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

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

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION , UNIT 1	05000338	1 OF 4

TITLE (
AUTOMATIC REACTOR TRIP DUE TO MALFUNCTION OF GENERATOR OUTPUT BREAKER

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-
05	07	2000	2000	004	00	06	02	2000	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)	100%		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 or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

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	EP	BKR	ABB	Y					

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 May 7, 2000, at 0757 hours, with Unit 1 in Mode 1 operating at 100% power, a "Generator Lockout - Turbine Trip" occurred due to a malfunction (suspected ground) on the "A" phase of the generator output breaker, G-12. This resulted in an automatic reactor trip from a "Turbine Trip - Reactor Trip" signal. The Auxiliary Feedwater (AFW) System received an automatic start signal as expected for a full power reactor trip. Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, Reactor Coolant System (RCS) pressure decreased to 1930 psig, pressurizer level decreased to 22 percent, and RCS temperature decreased to approximately 543 degrees F. Pressurizer pressure, level, and RCS temperature returned to their normal programmed values. All Engineered Safety Feature (ESF) equipment responded as designed. 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.

1.0 DESCRIPTION OF THE EVENT

On May 7, 2000, at 0757 hours, with Unit 1 in Mode 1 operating at 100% power, a "Generator Lockout - Turbine Trip" occurred due to a malfunction (suspected ground) on the "A" phase of the generator output breaker (EIIS System EL, Component BKR), G-12. The turbine trip resulted in an automatic reactor trip from a "Turbine Trip - Reactor Trip" signal. A description of the event is described below.

On May 7, 2000, at 0750 hours, a "G-12 Trouble" alarm annunciated in the Unit 1 main control room. Operations personnel were dispatched to investigate. Upon arrival at the G-12 breaker, at approximately 0757 hours, smoke was observed coming from the "A" phase of the G-12 breaker. Before any action could be taken, the turbine tripped on a "Generator Lockout - Turbine Trip" signal. The reactor then tripped on a "Turbine Trip - Reactor Trip" signal with power greater that the P-8 permissive. The Auxiliary Feedwater System (AFW) (EIIS System BA) received an auto start signal due to low-low steam generator (EIIS System AB, Component SG) level as expected following a full power reactor trip. At 0800 hours, Fire Contingency Action procedure FCA-0, Fire Protections - Operations Response was entered for a fire at the G-12 breaker. The Fire Brigade Scene Leader (a licensed operator) responded to the fire at the G-12 breaker. At 0805 hours, the fire was reported out (self-extinguished). At 0819, the fire at the G-12 breaker re-flashed and the fire brigade was dispatched. At 0827, the fire was reported out (self-extinguished), however there was still smoke in the area. The fire brigade stayed in the area to monitor for potential re-flashes of the fire.

Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, Reactor Coolant System (RCS) (EIIS System AB) pressure decreased to 1930 psig, pressurizer level decreased to 22 percent, and RCS temperature decreased to approximately 543 degrees F. Pressurizer pressure, level, and RCS temperature returned to their normal programmed values. All Engineered Safety Feature (ESF) equipment responded as designed. The post trip response progressed as expected and the Operators transitioned to 1-ES-0.1, Post Trip Recovery. The plant was stabilized at no-load conditions. A non-emergency four-hour report was made to the NRC Operations Center, at 0945 hours, on May 7, 2000, pursuant to 10 CFR 50.72(b)(2)(ii) for an event that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

The unit equipment responded as expected with a few discrepancies. The discrepancies included: "B" Moisture Separator Reheater (MSR) flow control valve (EIIS System SB, Component FCV), 1-MS-FCV-104B, did not close when the reheater control reset button was depressed, 2) position indication for the "D" MSR flow control valve, 1-MS-FCV-104D, did not indicate closed with the valve actually in the closed position, 3) intermediate range nuclear instrumentation detector (EIIS System JD, Component DET), N-35, was under-compensated, and 4) detection of a ground on the I-III vital bus (EIIS System EF).

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 a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

3.0 CAUSE

The cause of the automatic reactor trip was a "Generator Lockout - Turbine Trip" due to a suspected ground on the "A" phase of the generator output breaker, G-12. The cause of the ground has not yet been determined.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the reactor trip in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Initially, RCS pressure decreased to 1930 psig, pressurizer level decreased to 22 percent, and RCS temperature decreased to approximately 543 degrees F. Pressurizer pressure, level, and RCS temperature returned to their normal programmed values. All ESF equipment responded as designed.

The post trip response progressed as expected and the Operators transitioned to 1-ES-0.1, Post Trip Recovery. The plant was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Post Trip Review meeting was conducted at 1200 hours on May 7, 2000, with station personnel to identify the cause of the reactor trip to prevent recurrence, to identify abnormal or degraded indications occurring during the reactor trip, and to assess Unit readiness for return to operation.

The G-12 breaker control power was de-energized to clear a ground on the I-III vital bus.

Instrument air was isolated to 1-MS-FCV-104B and D to close the valves. It was later determined that there was a bad gasket for the manual bypass handle on the positioner on 1-MS-FCV-104B. The gasket was replaced and the valve was stroked satisfactorily. The limit switch on 1-MS-FCV-104D was adjusted and the valve was stroked satisfactorily.

The source range detectors were manually energized because N-35 was under-compensated. N-35 was subsequently functionally tested satisfactorily.

The G-12 generator output breaker was tagged out for maintenance. The G-12 breaker was subsequently replaced, tested, and returned to service.

6.0 ACTIONS TO PREVENT RECURRENCE

A root cause evaluation is being performed regarding the automatic reactor trip. Corrective actions will be performed as necessary following completion of the evaluation.

7.0 SIMILAR EVENTS

No events were identified for an automatic reactor trip as a result of a malfunction of a generator output breaker.

8.0 ADDITIONAL INFORMATION

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

Component failure information:
Description: Generator Output Breaker, G-12
Manufacturer: Asea Brown Boveri
Model No.: DR36V175D5