10CFR50.73

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

April 17, 2003

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

Serial No.: 03-256
NAPS: MPW
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 Power Station Unit 2.

Report No. 50-339/2003-001-00

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,

for DAH

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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION , UNIT 2	05000 - 339	1 OF 4

TITLE (4)
Automatic Reactor Trip Due to Steam Flow - Feed Flow Mismatch

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-
03	31	2003	2003	-- 001 --	00	04	17	2003	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	1
POWER LEVEL (10)	100 %

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	SJ	FCV	W120	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 March 31, 2003, at 1259 hours with Unit 2 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "C" steam generator (SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "C" main feed regulating valve. This resulted in a reactor and turbine trip. Closure of the "C" main feed regulating valve was due a blown fuse on the valve driver card. At 1540 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) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system. This event posed no significant safety implications because the Reactor Protection 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 March 31, 2003, at 1259 hours with Unit 2 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "C" steam generator (SG) (EIIS System AB, Component SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "C" main feed regulating valve (MFRV) (EIIS System SJ, Component FCV). This resulted in a reactor and turbine trip.

At 1259 hours the Unit 2 operator at the controls received a process rack power supply failure alarm followed by a "C" steam generator steam flow-feed flow mismatch alarm. The operator noted an absence of indicating lights for the "C" feed reg. valve controller and that the valve indicated closed. The reactor automatically tripped from a steam flow-feed flow mismatch with low steam generator level initiating signal within seconds of the receipt of the alarms. 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. Initially, Reactor Coolant System (RCS) (EIIS System AB) pressure and temperature decreased to approximately 1983 psig and 544 degrees Fahrenheit. Subsequently, RCS pressure and temperature returned to their normal programmed values.

Following the reactor trip the Reactor Protection System (RPS) and all Engineered Safety Feature (ESF) (EIIS System JE) equipment responded as designed including proper operation of AMSAC, and the Auxiliary Feedwater System (AFW) (EIIS System BA). No other major equipment issues were noted.

At 1540 hours a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B) for an event causing actuation of the Reactor Protection System when the reactor is critical. An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A) for an event causing actuation of the Auxiliary Feedwater System.

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 reportable 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

Cause of the automatic reactor trip was the "C" SG low level coincident with a steam flow greater than feed flow mismatch. The initiating signal was caused by closure of the "C" MFRV. Closure of the "C" MFRV was the result of a failed driver card in the SG water level control system for "C" SG. The driver card failed (i.e., de-energized) as a result of a blown fuse.

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, the MFW System was placed in service to all three S/Gs and the AFW System secured and returned to normal AUTO/Standby alignment. Subsequently, Control Room personnel transitioned to 2-OP-1.5 in preparation for unit re-start.

5.0 ADDITIONAL CORRECTIVE ACTIONS

The "C" MFRV failed driver card was replaced with a refurbished card and a successful functional test was performed. The driver cards for the Unit 2 "A" and "B" MFRVs were inspected and the fuses were replaced. Functional tests were performed successfully. Unit 2 entered Mode 1 at 0256 hours on April 1, 2003. Unit 2 achieved 100 percent power at 0122 hours on April 2, 2003.

With Unit 1 defueled for a scheduled refueling outage the driver cards for the "A", "B", and "C" MFRVs and the by-pass valves were pulled, the fuses inspected and replaced as necessary.

6.0 ACTIONS TO PREVENT RECURRENCE

Unit 2 last experienced an automatic reactor trip from a MFRV driver card failure in 1991. Corrective actions implemented included developing preventive maintenance (PM) procedures to replace the MFRV driver cards every third refueling outage. The Unit 2 MFRV driver cards were last replaced in 1999. The next scheduled driver card replacement was due during the 2004 refueling outage. Following the 1999 driver card replacement, industry operating experience identified the need to inspect and replace fuses as part of the 7300 processor card refurbishment. Subsequently, the PM was revised to include this action in the next schedule PM. The current controls ensure driver card refurbishment's included proper inspections and necessary repairs.

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

LER N2-91-009-00 dated 10/10/91, documents an automatic reactor trip from "B" steam generator low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" main feed regulating valve.

8.0 ADDITIONAL INFORMATION

At the time of this event Unit 1 was defueled as part of a scheduled refueling outage.

Component failure information:
Description: Driver Card Mark No. 02-FW-FCY-2498
Manufacturer: Westinghouse
Model No.: 2837A16G03

Our Instrument and Calibration group is refurbishing these driver cards on site.