10CFR50.73

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

April 25, 2000

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

Serial No.:	00-211
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 Unit 2.

Report No. 50-339/2000-002-00

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,

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

JE22

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 2	05000339	1 OF 4

TITLE MANUAL REACTOR TRIP DUE TO LOSS OF A REACTOR COOLANT PUMP

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	04	2000	2000	002	00	04	25	2000	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	1
POWER LEVEL (10)	7%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	x	50.73(a)(2)(i)		50.73(a)(2)(viii)
	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)
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
D	EK	DG	F010	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 April 4, 2000, at 1146 hours, with Unit 2 in Mode 1 operating at approximately 7% power, a manual reactor trip was initiated to comply with Technical Specification 3.4.1.1 due to the loss of the "C" reactor coolant pump. While preparing to install the potential transformer fuses for recovery of the 2C Station Service Bus (SSB), the drawer for the "F" emergency transfer bus was opened. This caused the "C" Reserve Station Service Transformer (RSST) to unload, de-energizing the 1H and 2J emergency busses and also both Unit's "C" SSBs. The "C" Reactor Coolant Pumps on both units tripped due to the loss of the "C" SSBs. Both unit's SSB power was being provided from the RSST busses at this time. This event is reportable pursuant to 10CFR50.73 (a)(2)(i)(A) and (a)(2)(iv). The loss of the emergency busses was the result of personnel error when the incorrect potential transformer fuse drawer was opened. This event posed no significant safety implications because the Unit's 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. Non-emergency one and four hour reports were made at 1225 hours on April 4, 2000, pursuant to 10 CFR 50.72(b)(1)(i)(A) and (b)(2)(ii).

1.0 DESCRIPTION OF THE EVENT

On April 4, 2000, North Anna Unit 2 was in Mode 1, approximately 7 percent power, undergoing a restart from an automatic reactor trip that occurred on April 3, 2000, from a secondary feeder cable (EIIS System EA, Component CBL5) failure from the 2C Station Service Transformer (SST) (EIIS Component XFMR) to the 2C Station Service Bus (EIIS Component BUS). Preparations were in progress for returning the 2C SST to service. While preparing to install the Unit 2 potential transformer (PT) fuses for the 2C Station Service Bus, the drawer for the "F" emergency transfer bus PT fuses was opened. This caused the "C" Reserve Station Service Transformer (RSST) to unload, de-energizing the 1H and 2J emergency busses (EIIS System EB, Component BUS) and also both Unit's "C" Station Service busses (SSB). The "C" Reactor Coolant Pumps (RCP) (EIIS System AB, Component P) on both units tripped due to the loss of the "C" Station Service Busses. Both unit's SSB power was being provided from the RSST busses at this time. The Unit 2 reactor was manually tripped at 1146 hours, due to the lost reactor coolant pump, to comply with Technical Specification 3.4.1.1. Technical Specification 3.4.1.1 requires the unit to be placed in hot standby within 1 hour. The 2J Emergency Diesel Generator (EDG) (EIIS System EK, Component DG) auto started and supplied power to the 2J emergency bus. The 1H EDG failed to automatically start and the 1H emergency bus remained de-energized.

With Unit 1 in Mode 5, the "A" Residual Heat Removal (RHR) (EIIS System BP, Component P) pump tripped when the 1H emergency bus was de-energized. With the loss of forced Reactor Coolant System (RCS) cooling, the emergency procedure, 1-AP-11, was entered and forced cooling was restored in approximately 2 minutes when the "B" RHR pump was placed in service. Secondary cooling remained available during the loss of forced cooling and the RCS temperature temporarily decreased due to losing the heat input from the running RCP. At 1219 hours, the "C" RSST and 1H busses were restored. Once the "C" Station Service Busses were re-energized via the "C" RSST, the Unit 1C RCP was started at 1236 hours. With the "C" RSST back in service, the Unit 2 "J" emergency bus was placed back on normal feed and the 2J EDG was shutdown. Unit 2 "C" RCP was started at 1446 hours on April 4, 2000.

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 2100 psig, pressurizer level decreased to approximately 13.5 percent, and RCS temperature decreased to 532 degrees F. Pressurizer pressure, level and RCS temperature returned to their normal programmed values. All Unit 2 Engineered Safety Feature (ESF) equipment responded as designed.

At 1225 hours, a 1 hour non-emergency report was made to the NRC Operations Center pursuant to 10 CFR 50.72 (b)(1)(i)(A) for a plant shutdown required by the Technical Specifications for having less than 3 RCP's operating in Modes 1 or 2. At this time, a 4 hour non-emergency report was also presented pursuant to 10 CFR 50.72(b)(2)(ii) for actuation of the RPS equipment as a result of manually tripping the unit.

At 1805 hours, the reactor trip breakers for Unit 2 were closed and a reactor start-up initiated. The unit entered Mode 2 at 2247 hours and the reactor was critical at 2314 hours.

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 with the exception of the 1H EDG. However, two independent offsite power supply circuits were operable and capable of supplying power to safety related equipment. The station blackout diesel was also operable. The Unit 1J EDG and "B" train of RHR were operable during the event. Therefore, the health and safety of the public were not affected by this event.

Non-emergency one and four hour reports were made to the NRC Operations center at 1225 hours on April 4, 2000 pursuant to 10 CFR 50.72(b)(1)(i)(A) and (b)(2)(ii). This event is reportable pursuant to 10 CFR 50.73 (a)(2)(i)(A) and (a)(2)(iv).

3.0 CAUSE

The loss of the "C" RSST was the result of opening the fuse drawer for the "F" emergency transfer bus, de-energizing the 1H and 2J emergency busses and both Unit's "C" Station Service busses. The cause of opening the incorrect potential transformer fuse drawer was the result of personnel error.

Operations personnel, a licensed Control Room Operator and non-licensed Control Room Operator, in conjunction with a Station Electrician, were assigned to clear equipment tags and install station service line PT fuses, respectively. The workers were instructed to report to the normal switchgear room and wait for a Senior Reactor Operator (SRO) and Electrical Supervisor to arrive to perform a detailed pre-job brief. Prior to the arrival of the SRO, the workers performed a walkdown of the clearance boundary and were searching for the equipment tags and fuses associated with the "C" SSB. While searching for the equipment tags and fuses, before starting any maintenance activities, the incorrect PT fuse drawer was opened. Several contributing factors led up to opening the incorrect PT line fuse drawer: 1) actions were taken prior to the planned pre-job brief, 2) workers failed to ensure they were at the proper component, 3) workers failed to understand or heed the caution labels posted on the cubicle, 4) workers failed to follow procedure regarding the location of the fuses, and 5) a questioning attitude among the workers was not evident.

The 1H EDG failed to start automatically as a result of the number 3 cylinder being hydraulically locked due to being full of oil. A separate Licensee Event Report is being issued to report the 1H EDG automatic start failure.

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 stable shutdown condition. The post trip response progressed as expected, and the Operators transitioned to 2-ES-0.1, Post Trip Recovery.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Post Trip Review meeting was conducted on April 4, 2000, 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.

A root cause evaluation is being performed regarding the manual reactor trip including personnel errors. The Operations personnel involved were removed from normal duties until a remediation plan could be determined and completed. A change was initiated to the controlling procedure to enhance the directions regarding clearing tags and locating fuses. This experience will be provided to Operations personnel during re-qualification training. The contributing factors to this event were discussed with station electricians to re-enforce proper maintenance practices.

6.0 ACTIONS TO PREVENT RECURRENCE

Upon completion of the root cause evaluation, corrective actions not already completed will be performed as necessary.

7.0 SIMILAR EVENTS

No events were identified regarding manual reactor trips due to opening the potential transformer fuse drawers and de-energizing the emergency busses and the station service busses.

8.0 ADDITIONAL INFORMATION

Unit 1 was in Mode 5 for a scheduled refueling outage. This event affected Unit 1 resulting in the failed automatic start of the 1H EDG. Failure of the 1H EDG is being reported under a separate Licensee Event Report. (Component Description: Emergency Diesel Generator, Manufacturer: Fairbanks Morris, Model No.: 38TD8.125)