Vepco VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117

10 CFR 50.73

June 11, 1991

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

Serial No. N-91-012
NAPS:MPW
Docket Nos. 50-338
License Nos. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. 91-012-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours,

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station

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1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION UNIT 1	0 5 0 0 0 3 3 8	1 OF 0 3

TITLE (4)

AUXILIARY FEEDWATER PUMP AUTO-START SIGNAL RECEIVED

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 NAMES	DOCKET NUMBER(S)
0 5	2 0	9 1	9 1	— 0 1 2 —	0 0	0 6	1 1	9 1		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)	4			
	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0 0 0	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
	20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
	20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
	20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
G. E. Kane, Statio… Manager	7 0 3	8 9 4 - 2 1 0 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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 fifteen single-space typewritten lines) (16)

At 0603 hours on May 20, 1991, with Unit 1 in Mode 4 (hot shutdown) and heating up in preparation for unit start-up, the Auxiliary Feedwater Pumps (AFWP) received an auto start signal when the circuit breakers for the "C" Main Feedwater Pump (FWP) opened due to a hi-hi level in the "C" Steam Generator (SG). The "C" SG hi-hi level was caused by opening the "C" Main Steam (MS) Non-Return Valve (NRV) with a sufficient differential pressure to induce a swell in SG water level. A four hour report was made in accordance with 10CFR50.72 (b)(2)(ii) at 0929 hours. This event is reportable pursuant to 10CFR50.73(a) (2) (iv) as an automatic actuation of an Engineered Safety Feature.

The event was caused by personnel error. Steam generator level returned to normal, feedwater(FW) isolation was reset, and the main feedwater pump breakers were re-closed to restore feed capability from the condensate system.

The AFWP's are not required until Mode 3 and were in "Pull-to-Lock". Therefore, an actual auto-start did not occur. Sufficient feedwater was available and all systems functioned as designed. The plant remained stable during the event. Therefore, the health and safety of the public was not affected at any time during this event.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

1.0 Description of the Event

At 0603 hours on May 20, 1991, with Unit 1 in Mode 4 (hot shutdown), the Auxiliary Feedwater Pumps (AFWP)(EIIS System Identifier BA, Component Identifier P) received an auto start signal when the circuit breakers for the "C" Main Feedwater Pump (FWP)(EIIS System Identifier SJ, Component Identifier P) opened due to a hi-hi level in the "C" Steam Generator (SG)(EIIS System Identifier AB, Component Identifier SG). The "C" SG hi-hi level was caused by opening the "C" Main Steam (MS) Non-Return Valve (NRV)(EIIS System Identifier SB, Component Identifier V) with a sufficient differential pressure to induce a swell in SG water level.

The event occurred while heating up the Reactor Coolant System (RCS)(EIIS System Identifier AB) in preparation for unit start-up. The Residual Heat Removal System (RHR)(EIIS System Identifier BP) had been secured in accordance with Operating Procedures and RCS temperature control transferred to the Steam Generators. At approximately 315 degrees operations personnel initiated actions to stabilize RCS temperature to allow a review of plant conditions and procedures to ensure Mode 3 entry was permissible. Steam generator blowdown was placed in service to help remove heat from the RCS and all MS NRV bypass valves were opened to pressurize the main steam header to allow a MS NRV to be opened. When RCS temperature increased to 330 degrees, the "C" MS NRV was opened to provide additional steam removal capability to the Main Condenser (EIIS System Identifier SG). Because the Main Steam Header (EIIS System Identifier SB) was not equalized in pressure with the SGs, the "C" SG level swelled to greater than 75 percent of the narrow range resulting in a feedwater (FW) isolation signal. This tripped the Main Feed Pump breakers, which were closed in the test position to allow feeding with the condensate pumps. The FW isolation signal also shut the main feed regulating valve bypasses, and isolated SG blowdown (EIIS System Identifier WI). The trip of the main feed pump breakers generated an AFWP start signal. The AFWP's are not required until Mode 3 and were in "Pull-to-Lock". Therefore, an actual auto-start did not occur. All systems functioned as designed. The plant conditions remained stable during the event.

2.0 Significant Safety Consequences and Implications

The AFWP's are not required until Mode 3 and were in "Pull-to-Lock" so an actual auto-start did not occur. Sufficient SG inventory existed to allow more than adequate time to re-establish MFW flow. AFW could have been quickly placed in service had this been necessary. The plant remained stable during the event. Therefore, the health and safety of the public was not affected at any time during this event.

3.0 Cause of the Event

The cause of the event was personnel error. Prior to opening 1-MS-NRV-101C the Shift Supervisor considered other options available for stabilizing the RCS temperature. These options included: stopping a reactor coolant pump (RCP), opening the SG Power Operated Relief Valves (PORV), opening all three MS NRVs at once, or allowing the RCS to heatup and

change modes. However, these options were not implemented for various operational considerations. The option of opening the "C" MS NRV was chosen because the "C" SG water level was the lowest of the three SGs. The Shift anticipated a rise in SG level, but did not expect the level to increase above 75 percent and result in the AFW auto start signal.

4.0 Immediate Corrective Actions

The "C" SG level returned to normal, FW isolation was reset, and the breakers for the "C" FWP were closed to restore feed capabilities from the condensate system.

5.0 Additional Corrective Actions

Steam generator blowdown was returned to service

6.0 Actions Taken to Prevent Recurrence

Administrative limitations requiring the HP release form has been amended to allow additional operating flexibility for SG PORVs.

Operations procedures controlling heatup and cooldown will be further enhanced to provide additional clarification for RCS heat removal techniques.

The Shift Supervisor was interviewed by Operations Management on the decision process.

The event will be included in the Licensed Operator Requalification Program Cycle training - Mods and Experiences.

7.0 Similar Events

Similar recent Licensee Event Reports (LER) involving automatic actuation of the auxiliary feed water pumps is as follows:

LER N2-90-003-00 — Auxiliary feedwater pumps auto-started when main feedwater pump breakers opened on Hi-Hi level in SG while cooling down. Cause was identified as personnel error and procedure inadequacy.

8.0 Additional Information

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