NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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 4

TITLE (4): REACTOR TRIP DUE TO 'C' S/G LEVEL SIGNAL

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 3	1 8	8 8	8 8	— 0 1 4 —	0 0	0 4	1 5	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3
POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
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
G. E. Kane, Station Manager	AREA CODE 7 0 3 8 9 4 - 5 1 5 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
				N					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 8	0 1	8 8

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 1705 hours on March 18, 1988, with Unit 1 at zero percent power (Mode 3), and the reactor trip breakers open, a reactor trip signal was generated when the 'C' steam generator (S/G) narrow range level decreased to 25 percent while the 'C' S/G channel IV Steam Flow-Feed Flow Mismatch bistable was in trip. The 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was placed in trip for troubleshooting FI-1495, 'C' S/G channel IV steam flow. This event is reportable pursuant to 10CFR50.73(a)(2)(iv). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

The cause of this event was failure to maintain the 'C' S/G level above 25 percent while the 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was in trip. As a corrective action, the 'C' S/G level was restored above 25 percent. A Human Performance Evaluation System (HPES) review will be performed with regard to this event.

The health and safety of the general public were not affected at any time during this event.

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1.0 Description of Event

At 1705 hours on March 18, 1988, with Unit 1 at zero percent power (Mode 3), and the reactor trip breakers open, a reactor trip signal was generated when the 'C' steam generator (S/G) narrow range level decreased to 25 percent while the 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was in trip. The 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was placed in trip when the 'C' S/G channel IV Steam Flow, FI-1495 (EIIS System Identifier SB, Component Identifier FT), was placed in trip for troubleshooting FI-1495, 'C' S/G channel IV steam flow. This event is reportable pursuant to 10CFR50.73(a)(2)(iv). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

On March 17, 1988, Unit 1 was ramped off line, and placed in Mode 3 in order to stroke test a containment isolation valve. During this rampdown, the 'A' S/G steam flow channel IV, FI-1475, was declared inoperable at 2147 hours. Channel IV was placed in trip after it was determined that it had deviated from the 'A' S/G steam flow channel III, FI-1474, by more than the channel check acceptance criteria. At 2205 hours, FI-1495 was placed in trip after it was determined that this channel had deviated from the 'C' S/G steam flow channel III, FI-1494, by more than the channel check acceptance criteria.

At 0511 hours on March 18, 1988, FI-1495 was returned to service, and FI-1475 was returned to service at 1555 hours. At 1600 hours, FI-1495 was again placed in trip, by Instrument personnel, for additional troubleshooting related to the earlier deviations between channel III and channel IV.

In order to calibrate or test FI-1495, per Instrument Calibration Procedure ICP-MS-1-F-1495, the bistables associated with the 'C' S/G channel IV Steam Flow are placed in the "trip" position. Therefore, the 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was placed in trip. When this bistable is in trip, a level of 25 percent in the 'C' S/G will generate a reactor trip signal. Normally, when this bistable is not in trip, and with no steam flow-feedwater flow mismatch, a reactor trip signal is only generated when the S/G level decreases to 18 percent on 2 out of 3 channels.

The logic for a Safety Injection signal is high steam flow sensed by 1 out of 2 channels on 2 out of 3 steam generators, coincident with 2 out of 3 RCS loop average temperatures less than 543 degrees F.

As part of the instrument calibration procedure, ICP-MS-1-F-1495, 'A', 'B', and 'C' steam generator steam flow channel IV bistables were placed in trip. Subsequently, the high steam flow portion of the Safety Injection signal logic was completed, and a Safety Injection signal would have been generated if RCS average temperature decreased to 543 degrees F on 2 out of 3 loops.

While Unit 1 was in Mode 3, feedwater flow was being supplied to the steam generators by throttling the manual isolation valves which are in series with the bypass feedwater regulating valves. The feedwater temperature was approximately 80 degrees F. Introducing feedwater flow, of this temperature, into a steam generator which contains water at approximately 550 degrees F, causes the Reactor Coolant System (RCS) temperature to decrease rapidly.

For approximately 15 minutes prior to the event, both RCS temperature and 'C' S/G level were indicating a downward trend. The control room operator recognized that a S/G level of 25 percent, and not the normal trip setpoint of 18 percent level, would result in a reactor trip signal due to the Steam Flow-Feedwater Flow Mismatch bistable in trip. Subsequently, the control room operator began to gradually increase the feedwater flow to the 'C' S/G in order to restore the level without causing a significant decrease in RCS temperature. This action, while preventing the degradation of RCS temperature and a Safety Injection, did not prevent the level in the 'C' S/G from decreasing to 25 percent on 1 out of 2 level channels. Consequently, a reactor trip signal was generated. The RCS temperature was approximately 548 degrees F when the reactor trip signal occurred. A four hour report was made at 1742 hours in accordance with 10CFR50.72(b)(2)(ii).

2.0 Safety Consequences and Implications

No significant safety consequences resulted from this event because the reactor trip breakers were open, and the control rods were fully inserted at the time the signal was generated. While the unit is in Mode 3, the reactor trip signal associated with the Steam Flow-Feedwater Flow Mismatch coincident with Low Steam Generator Water Level is not required to be operable, by Technical Specifications. Throughout this event, the 'C' S/G channel III steam flow was fully operable, and channel IV was in trip in accordance with T.S. requirements.

The health and safety of the general public were not affected at any time during this event.

3.0 Cause of The Event

The cause of this event was failure to maintain the 'C' S/G level above the 25 percent reactor trip setpoint while the 'C' S/G channel IV Steam Flow-Feedwater Flow Mismatch bistable was in trip. Normally when this bistable is not in trip, and with no steam flow-feedwater flow mismatch, a reactor trip signal is generated when S/G level decreases to 18 percent on 2 out of 3 channels. Due to cold feedwater temperature, feedwater flow had to be increased gradually to prevent a significant RCS cooldown.

The constraint of a more restrictive band in which to control S/G level, the need to gradually increase feedwater flow in order to prevent an RCS cooldown, and the concern regarding a possible Safety Injection contributed to the S/G level not being maintained above the 25 percent reactor trip setpoint.

4.0 Immediate Corrective Action

As an immediate corrective action, the 'C' steam generator level was restored to above 25 percent.

5.0 Additional Corrective Actions

As an additional corrective action, a Human Performance Evaluation System (HPES) review will be performed with regard to this event. Any additional corrective actions identified as a result of this review will be evaluated and implemented as necessary.

6.0 Actions Taken to Prevent Recurrence

Any significant recommendations identified by the HPES review which could prevent recurrence of this type of event will be implemented as necessary.

7.0 Similar Events

There have been no similar events, at North Anna, in which a reactor trip signal has been generated with a Steam Flow-Feedwater Flow Mismatch bistable in trip. However, low feedwater temperature was a major contributor to the "Hi Hi steam generator level" reactor trip reported in LER 88-005-00.

8.0 Additional Information

An engineering review of the acceptance criteria for the channel check tolerance was performed, and it was determined that at low power, the channel check tolerance band was very conservative. This tolerance band was subsequently revised. With the new channel check acceptance criteria in place, the potential for placing the S/G steam flow channel in trip for troubleshooting will be reduced.

10 CFR 50.73

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

April 15, 1988

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

Serial No. N-88-017
NO/MLT: nih
Docket No. 50-338

License No. 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. LER 88-014-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very Truly Yours,

GEK

G. E. Kane
Station Manager

Enclosure

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

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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