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AUTH.NAME AUTHOR AFFILIATION
KANSLER,M.R. Virginia Power (Virginia Electric & Power Co.)
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-002-00:on 930803,automatic reactor trip occurred due to low steam generator water level coincident w/steam/ feedwater flow mismatch resulting from spurious closure of main feedwater regulating valve A.W/930827 ltr.

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RIDS/ADDS

10CFR50.73

Virginia Electric and Power Company
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

August 27, 1993

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

Serial No.: 93-538
SPS:RCB
Docket No.: 50-281
License No.: DPR-37

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

REPORT NUMBER

50-281/93-003-00

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,

M. R. Kansler
Station Manager

Enclosure

cc: Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

030010

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1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Surry Power Station, Unit 2	05000 - 281	1 OF 5

TITLE (4) Unit 2 Automatic Reactor Trip Due to Low Steam Generator Water Level Coincident With Steam/Feedwater Flow Mismatch Resulting From Spurious Closure of "A" MFRV

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	03	93	93	-- 003 --	00	08	27	93		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	97%		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
M. R. Kansler, Station Manager	(804) 357-3184

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
B	AA	IL	M035	N		B	JB	FCV	W120	Y
B	SB	RV	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)

At 2005 hours on August 3, 1993, with Unit 1 at 100% power and Unit 2 at 97% power, Unit 2 experienced an automatic reactor trip. The trip occurred when the "A" Main Feedwater Regulating Valve (MFRV) unexpectedly closed, causing a feed flow/steam flow mismatch coincident with a low water level in the "A" steam generator. Reactor Protection System functions actuated as designed, and post-trip response was satisfactory. The reactor was placed in a safe, hot shutdown condition, and the health and safety of the public were not affected. The cause of the closure of the MFRV was traced to an erratic power supply in the manual/automatic control station. This report is required by 10CFR50.73(a)(2)(iv).

REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

1.0 DESCRIPTION OF THE EVENT

On August 3, 1993, at 2005 hours, with Unit 2 at 97% power, the Unit experienced an automatic reactor trip because of a steam flow/feed flow mismatch coincident with a low water level in the "A" steam generator, 2-RC-E-1A (EIIS-AB, SG). The transient occurred when the "A" Main Feedwater Regulating Valve (MFRV), 2-FW-FCV-2478 (EIIS-JB, FCV) unexpectedly closed. The turbine (EIIS-TA) and main generator (EIIS-TB) tripped as designed. The Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) also actuated as designed. The Auxiliary Feedwater Pumps, 2-FW-P-2, 2-FW-P-3A, and 2-FW-P-3B (EIIS-BA, P) automatically started on decreasing steam generator level as designed.

Control Room operators responded to the trip in accordance with emergency and other operating procedures. Plant response was as expected except for the following:

- Individual Rod Position Indicator (IRPI) rod bottom light (EIIS-AA, IL) for control rod M-10 was slow in illuminating. (The rod bottom light illuminated at approximately 2055 hours.)

- "C" Steam Generator Power Operated Relief Valve (PORV), 2-MS-RV-201C, (EIIS-SB, RV) indicated "intermediate" at 960 psig (design lift setting is 1035 psig). Subsequent investigation showed that the valve did not lift.

The Nuclear Regulatory Commission was notified in accordance with 10CFR50.72 at 2221 hours. This event is being reported pursuant to 10CFR50.73(a)(2)(iv) as an automatic actuation of the Reactor Protection System (RPS) (EIIS-JC).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

Upon receipt of the reactor trip, RPS actuations functioned as designed, and all control rods inserted into the core. The electrical buses transferred properly and off-site power was maintained throughout the event. The emergency diesel generators remained operable in automatic, but were not required to start. Station operating personnel acted promptly to place the plant in a stable, hot shutdown condition. The shutdown margin of reactivity was calculated and found to be satisfactory. The health and safety of the public were not affected.

3.0 CAUSE OF THE EVENT

The reactor tripped as designed when the "A" Steam Generator experienced a feed flow/steam flow mismatch coincident with a low steam generator water level. The cause of the decrease in feedwater flow and loss of water level was closure of the "A" MFRV. The valve's positioner received an unexpected demand to shut, and efforts on the part of the licensed Control Room Operator to take manual control and open the valve were unsuccessful in reversing the transient. The cause of the closure of the MFRV was traced to an erratic power supply in the manual/automatic control station.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Operators acted promptly to place the plant in a safe, shutdown condition in accordance with emergency and other operating procedures. The licensed Control Room Operator adjusted the "C" Steam Generator PORV set point to clear the intermediate indication.. The Shift Technical Advisor monitored the safety function status trees to verify that unit conditions were acceptable.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

- IRPI rod bottom light for control rod M-10.
  - Problems have been experienced with this indication for several years. The Nuclear Steam Supply System vendor has evaluated the condition and concluded that its cause is residual permeability in the control rod drive mechanism housing. This phenomenon is caused by a combination of factors, including material composition and the decrease in reactor coolant system temperature following a trip.
  - A hot rod drop test conducted subsequent to the reactor trip showed that control rod M-10 was fully operable.

- The "C" Steam Generator PORV control loop was investigated and a number of discrepancies were detected. The pressure controller and power supply were replaced and the transmitter was recalibrated.

- "A" Main Feedwater Regulating Valve.
  - An extensive troubleshooting effort was conducted in an attempt to identify the source of the signal which caused the "A" main were examined:
    - -- valve controller inputs
    - -- manual/automatic control station
    - -- turbine first stage pressure input summator
    - -- terminal connections on control system modules
    - -- valve actuator
    - -- valve positioner
    - -- electro/pneumatic converter.
  - A number of components were replaced, including:
    - -- flow controller
    - -- manual/auto control station

During examination of the MFRV manual/automatic control station which had been replaced, it was noted that the - 15 VDC power supply was behaving erratically. A failure of this power supply will cause the controller to fail to zero output and inhibit control responses in automatic or manual. Since a lack of controller response in automatic and manual was observed during the event, the controller was determined to be the cause of the event.

6.0 ACTIONS TO PREVENT RECURRENCE

A Root Cause Evaluation (RCE) was initiated immediately after the reactor trip. Recommendations of the RCE will be evaluated and implemented as appropriate.

7.0 SIMILAR EVENTS

LER S2-86-007 Manual reactor trip due to high steam generator level (metal debris between MFRV plug and valve seat).

LER S2-90-003 Manual reactor trip due to failure of "A" Main Feedwater Regulating Valve (blockage of positioner air supply inlet filter/orifice assembly.)

LER S2-90-004 Manual reactor trip following inadvertent grounding of the "A" Main Feedwater Regulating Valve control signal during testing.

8.0 MANUFACTURER/MODEL NUMBER

IRPI Rod Bottom Light	Spang & Co., Magnetics Div. Signal Conditioning Card, EPC-2NI-13, Part # E 2786
"C" Steam Generator PORV Controller	Westinghouse Hagan Model 4111080-001
MFRV Manual/Automatic Control Station	Westinghouse Hagan 7100 Series Control System Model 124 Controller