

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

June 9, 1994

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

Serial No.: 94-360
SPS:MDK
Docket No.: 50-280
License No.: DPR-32

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 1.

REPORT NUMBER

50-280/94-006-00

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

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PDR ADOCK 05000280
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LICENSEE EVENT REPORT (LER)

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

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.

FACILITY NAME (1) Surry Power Station, Unit 1	DOCKET NUMBER (2) 05000 - 280	PAGE (3) 1 OF 4
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TITLE (4)
Unit 1 Manual Reactor Trip During Safeguards Actuation Logic Testing

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
05	11	94	94	006	00	06	09	94		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) 100	20.402(b)			20.405(c)			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 M. R. Kansler, Station Manager	TELEPHONE NUMBER (Include Area Code) (804) 357-3184
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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
A	JE	CL		N					

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 May 11, 1994, with Unit 1 initially at 100% power, the reactor was manually tripped at 2109 hours. The trip was initiated when one of the two sources of main feedwater was isolated and Operators were unable to recover steam generator levels. The cause of partial feedwater isolation was inadvertent actuation of a relay in the safety injection actuation circuitry during logic testing. Actuation of this relay caused the tripping of the power supply to one of the Main Feedwater (MFW) pump motors and the closure of the MFW pump's discharge isolation valve. The Reactor Protection System functioned as designed, and post-trip response was satisfactory. The reactor was placed in a safe, hot shutdown condition. The health and safety of the public were not affected. This report is being made pursuant to 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

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
Surry Power Station, Unit 1	05000 - 280	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 4
		94	- 006 -	00	

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

1.0 DESCRIPTION OF THE EVENT

On May 11, 1994, with Unit 1 initially at 100% power the reactor was manually tripped at 2109 hours. Immediately before the trip, at 2108 hours, an alarm was received in the Main Control Room indicating that circuit breaker 15C5, which supplies one of the Main Feedwater Pump B tandem drive motors [EIIS: SJ,MO], had opened. Because of the supply breaker-feedwater discharge valve interlock scheme, the associated discharge valve, 1-FW-MOV-154B [EIIS: SJ,ISV] was closing. Control Room Operators immediately began reducing turbine load and reactor power in an effort to restore stable conditions. Reduction in feedwater supply and the shrink from power reduction caused steam generator levels to continue to decrease, and the Control Room Operator manually tripped the reactor.

All control rods fully inserted into the core, and 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 (AFW) Pumps [EIIS: BA,P] 1-FW-P-2, 1-FW-P-3A and 1-FW-P-3B, started automatically and supplied feedwater to the steam generators [EIIS: AB,SG] as designed.

The main steam dumps [EIIS: SB,V] automatically opened to admit steam directly to the main condenser [EIIS: COND]. The Reactor Coolant System (RCS) [EIIS: AB] average temperature (Tave) was reduced to 547 degrees Fahrenheit and the main steam dumps closed as designed. RCS Tave decreased to a minimum of approximately 535 degrees Fahrenheit. RCS temperature subsequently stabilized at 547 degrees Fahrenheit (no-load temperature).

Intermediate Range Nuclear Instrument (IRNI) [EIIS: IG,JI] N-36 indicated off-scale low.

Control Room Operators responded to the trip in accordance with emergency and other operating procedures. Plant response was as expected.

A four-hour non-emergency report was made to the Nuclear Regulatory Commission in accordance with 10CFR50.72(b)(2)(ii) at 2348. This event is being reported pursuant to 10CFR50.73(a)(2)(iv), manual actuation of the Reactor Protection System (RPS) [EIIS: JC].

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

When the reactor was manually tripped, RPS actuations functioned as designed and all control rods fully inserted into the core. The electrical buses transferred properly, and off-site power was maintained throughout the event. The emergency diesel generators (EDG) [EIIS: EK,DG] remained operable in automatic, but were not required to start. The AFW Pumps started and supplied feedwater to the steam generators as designed. 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. No conditions adverse to safety resulted from this event, and the health and safety of the public were not affected.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Surry Power Station, Unit 1	05000 - 280	94	- 006 -	00	3 OF 4

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

3.0 CAUSE OF THE EVENT

Just before the reactor trip, Licensee Instrument and Calibration (I&C) Technicians were conducting surveillance testing in accordance with procedure 1-IPT-FT-RP-SI-001A, Train A Safeguards Actuation Logic Testing. A step in this procedure requires the measurement of electrical resistance of the relay coil F1A [EIS: JE,RLY,CL] in the actuation circuitry of Safety Injection Train A. This relay is located in the A Train Protection Rack with restricted access to the terminals. When the technician inserted the ohmmeter probes to take the measurement, he inadvertently disturbed the contacts which control the trip coil for the MFW pump motor [EIS: SJ,MO,CL(94)] as described in Section 1.0 above. The circuit breaker tripped, initiating the sequence of events which led to the manual reactor trip. The cause of the event was human error; however, the difficulty in testing the affected relay was a major contributing factor.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the trip, Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with emergency and other operating procedures. The Shift Technical Advisor monitored the critical safety function status trees to verify that unit conditions were acceptable. Plant response was as expected, and the unit was stabilized at hot shutdown.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

Intermediate Range Nuclear Instrument (IRNI) N-36 indicated off-scale low. This IRNI response has been observed following previous reactor trips and evaluated. The overcompensated response following a reactor trip is considered normal and is the expected response that may result when using the NSSS suppliers preferred methodology for setting compensating voltage. The Source Range Nuclear Instruments (SRNI) automatically reinstated as designed.

When the main steam dumps automatically opened to admit steam directly to the main condenser, the Reactor Coolant System (RCS) average temperature (Tave) was reduced to 547 degrees Fahrenheit and the main steam dumps closed as designed. The RCS cooldown minimum temperature of approximately 535 degrees Fahrenheit is expected and has been observed during previous reactor trips. RCS temperature subsequently stabilized at 547 degrees Fahrenheit (no-load temperature). This condition has been evaluated previously. Corrective actions are being taken based on the results of this evaluation. Shutdown margin calculations following a reactor trip include a conservatively safe value for RCS temperature ensuring adequate shutdown margin exists.

A post trip review was conducted and no additional corrective actions were identified.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

6.0 ACTIONS TO PREVENT RECURRENCE

Although the major cause of the transient was human error, a significant contributor was the inaccessibility of the circuit being tested. The following corrective actions are underway:

- A plant modification has been developed which will locate test points in positions where required test data may be obtained without the risk of inadvertent actuation. It is anticipated that this modification will be installed during the next refueling outage for each unit.
- An evaluation is being conducted to determine if the interval between the required surveillance testing of this and similar relays may be increased without adversely affecting nuclear safety. If such a change is permissible, the number of potential challenges to plant stability will be significantly reduced.

7.0 SIMILAR EVENTS

LER S2-90-004, Unit 2 Manual Reactor Trip Following Inadvertent Grounding During Testing.

8.0 MANUFACTURER/MODEL NUMBER

None.