

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

February 3, 1992

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

Serial No.: 92-088
Docket Nos.: 50-280
License Nos.: DPR-32

Gentlemen:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report for Unit 1.

REPORT NUMBER

92-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be reviewed by the Corporate Management Safety Review Committee.

Very truly yours,



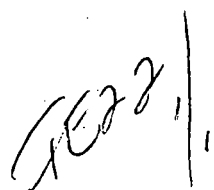
M. R. Kansler
Station Manager

Enclosure

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

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LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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.

FACILITY NAME (1) **Surry Power Station, Unit 1** DOCKET NUMBER (2) **0 5 0 0 0 2 8 0** PAGE (3) **1 OF 0 5**

TITLE (4)
Dropped Rod Due To Personnel Error Followed By A Required Manual Reactor Trip

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 1	0 2	9 2	9 2	0 0 1	0 0	0 2	0 3	9 2			0 5 0 0 0
<p>THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)</p>											

OPERATING MODE (8) N	20.402(b)	20.405(c)	<input checked="" type="checkbox"/>	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0 5 6	20.405(a)(1)(i)	50.38(c)(1)		50.73(a)(2)(v)	73.71(c)
	20.405(a)(1)(ii)	50.38(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)	<input checked="" type="checkbox"/>		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 **8 0 4 3 6 5 2 0 0 1**

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

CAUSE	SYSTEM	COMPONENT	MANUF. TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUF. TURER	REPORTABLE TO NPRDS
X	I G	J X	P 3 2 3	Y	X	T G	J X	L 0 4 5	Y
X	S B	B C V	F 1 3 0	Y	X	A A	R O D	W 3 5 1	Y

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE). NO

EXPECTED SUBMISSION DATE (15)

MONTH	DAY	YEAR

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

On January 2, 1992 at 1649 hours, with Unit 1 at 56% reactor power, troubleshooting was in progress for "B" shutdown bank control rod E-5 to determine why E-5 had dropped into the core at 0754 that morning during the performance of biweekly control rod freedom of movement testing. As "D" control bank was manually stepped out by the operator to control delta flux, a second rod, "D" control bank control rod H-2 dropped. Control Rod H-2 was verified to be in the core and the reactor was manually tripped in accordance with station abnormal procedures. This event occurred as the result of personnel error in that the troubleshooting guide prepared by Electrical Maintenance for rod E-5 did not identify shared circuitry between rod E-5 and H-2 which would result in H-2 being dropped if control rods were stepped during troubleshooting. Following the reactor trip, "A" main feed pump tripped when its recirculation valve failed to open due to a failed solenoid, intermediate range nuclear instrumentation indication was erratic due to high voltage power supply problems, steam generator atmospheric power operated relief valve for "A" steam generator responded poorly while the valves for "B" and "C" steam generators did not respond as expected, and the turbine generator electro-hydraulic control system indications were erratic. A four hour non-emergency report was made to the Nuclear Regulatory Commission in accordance with 10CFR50.72.

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

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

1.0 DESCRIPTION OF THE EVENT

On January 2, 1992 at 1649 hours, with Unit 1 at 56% reactor power, troubleshooting was in progress for "B" shutdown bank control rod E-5 [EIS-AA,ROD]. E-5 had dropped into the core at 0754 that morning during the performance of the monthly control rod freedom of movement testing. As the reactor operator manually adjusted control rods to control delta flux variations associated with the dropped rod, a second rod ("D" control bank control rod H-2) dropped. The operator promptly verified control rod H-2 to be in the core by observing a skewed core power distribution, and the reactor was manually tripped in accordance with station abnormal procedure 1-AP-1.00, "Rod Control System Malfunction." All rods were verified to be on the bottom following the trip, operators performed the appropriate station procedures, and the Shift Technical Advisor (STA) monitored the critical safety function status trees to ensure that plant parameters remained within safe bounds.

Due to steam generator (S/G) shrink during the transient [EIS-SB,SG], the level in the three steam generators decreased to less than 13%, which resulted in the start of the three auxiliary feed water pumps [EIS-BA,P] and the arming of ATWS Mitigation System Actuation Circuit (AMSAC). At 1650, the "A" main feed pump (MFP) [EIS-SJ,P] tripped when its recirculation valve failed to open. Also at 1650, AMSAC timed out and tripped. The AMSAC trip signal opened the control rod drive motor generator set supply breakers [EIS-AA,MG] and provided redundant turbine trip and auxiliary feedwater actuation signals as designed. Reactor coolant system [EIS-AB] average temperature (RCS T_{ave}) decreased below 543°F at 1651. The pressurizer heater breakers [EIS-AB,PZR] opened and letdown isolated at 1653 due to the low pressurizer level resulting from the cooldown. RCS T_{ave} stabilized at 532°F, eight minutes following the manual reactor trip. At 1659, with reactor power in the intermediate range, intermediate range channels N-35 and N-36 [EIS-IG] exhibited erratic behavior with the start up rate meter oscillating from the low to the high end of scale. At 1700 both source range nuclear instruments energized and indicated a stable shutdown was in progress.

At 1701, after turbine generator speed [EIS-TL,TG] had been observed to be 1100 RPM and decreasing, a main turbine vibration annunciator was received, the operators observed an erroneous main turbine speed indication of 2350 RPM at the main control board, and the electro-hydraulic control system [EIS-TG] indications were observed to be erratic. In response to the indication, main steam trip valves [EIS-SB,ISV] were closed and an operator was dispatched to verify turbine

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speed. After verifying the four turbine stop valves [EIS-SB,ISV] were fully closed, and that turbine speed was decreasing, preparations were initiated to reopen the main steam trip valves. The response of the "A" steam generator PORV following closure of the main steam trip valves was poor and the operator was unsuccessful in opening the "B" and "C" steam generator PORVs.

A four hour non-emergency report was made to the Nuclear Regulatory Commission in accordance with 10CFR50.72. This event is being reported pursuant to 10CFR50.73(a)(2)(i) and (iv).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event consisted of a manual reactor trip initiated immediately following the full insertion of a second control rod into the core at power. Plant parameters responded as expected following the trip. Although the "B" and "C" steam generator PORVs failed to open, no safety significance was involved since the plant is analyzed for a trip from full power with no steam generator PORVs operable. The safety analysis relies on the steam generator code safety valves for maintaining the primary heat sink and the steam generator safety valve setpoints were not challenged during the event. This event was within the bounds of the accident analysis, thus the health and safety of the public were not affected.

3.0 CAUSE OF THE EVENT

This event occurred as the result of personnel error in the preparation of the troubleshooting guide for rod E-5. An existing procedure which is normally used during refueling was referenced to prepare the troubleshooting instructions. The instructions directed that the fuses for control rod drive mechanism E-5 be removed (for personnel protection) to measure the resistance of the mechanism coils. Because the electricians preparing the troubleshooting guide did not fully understand the operation of the rod control system and did not consult station drawings, they were unaware that one of the fuses removed was common to the moveable coils for control rods E-5 and H-2. Consequently, when the control rods were manually stepped to control delta flux, the moveable coil for H-2 did not energize, the stationary coil for H-2 de-energized, and control rod H-2 dropped into the core.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control rod H-2 was verified to be in the core by observing a skewed core power distribution, and the reactor was manually tripped in

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accordance with station abnormal procedure 1-AP-1.00. Operators performed the appropriate procedures and the unit was brought to a stable hot shutdown condition. The STA monitored the critical safety function status trees to ensure that plant parameters remained within safe bounds.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

The coil stack for control rod drive mechanism E-5 was determined to contain a defect that was corrected by bringing the plant to cold shutdown and replacing the coil stack. The "A" MFP recirculation valve failed due to sticking solenoid operated valve, SOV-150A, which was subsequently replaced. The Intermediate Range Nuclear Instrument indication problems were determined to be power supply related and the high voltage power supplies were replaced. The erroneous turbine speed indication and problems encountered with the turbine Electro-Hydraulic Control System were caused by a capacitor failure in a +15 volt power supply which was replaced. The air supply regulators for the three steam generator PORVs were replaced. Component Failure Evaluations were initiated for the intermediate range nuclear instrument power supplies and the PORV air regulators.

6.0 ACTIONS TO PREVENT RECURRENCE

This event has been reviewed by personnel in the Electrical, Instrument and Controls, and Operations Departments through inclusion in required reading. This event will be reviewed in Electrical, Instrument and Controls, and management continuing training programs as well, re-emphasizing the need to exercise caution when preparing and approving troubleshooting instructions, particularly those instructions associated with equipment that is energized and inservice. As an enhancement, operator training will be revised to discuss the ramifications of stepping control rods while rod control troubleshooting is in progress.

Both the existing Instrument and Controls procedure for troubleshooting the rod control system and the abnormal procedure which governs operation with a misaligned control rod will be changed to alert the operator to the ramifications of stepping control rods while rod control troubleshooting is in progress. A new procedure will be developed for Electrical Maintenance troubleshooting of an energized and inservice rod control system which will also alert the operator to the ramifications of stepping control rods with troubleshooting in progress.

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Finally, enhanced training on the Rod Control System will be provided for technicians responsible for the system.

7.0 SIMILAR EVENTS

None.

8.0 ADDITIONAL INFORMATION

Power Designs Inc., Model UPMD-X54W (IR High Voltage Power Supply)

Fisher, Model 67AFR/224 (PORV Air Regulator)

Lambda, Model LMEE15-Y-3820-1 (+15 Volt EHC Power Supply)

Westinghouse Electric Corporation, Model L-106A (Control Rod Drive Mechanism)