10CFR50.73

Virginia Electric And Power Company
Surry Power Station
5570 Hog Island Road
Surry, Virginia 23883

November 21, 2000

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

Serial No.: 00-598
SPS: JSA
Docket No.: 50-280
License No.: DPR-32

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 1.

Report No. 50-280/2000-004-00

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,

JBSowers for
R. H. Blount II
Site Vice President

Enclosure

IE22

Commitments contained in this letter:

1. A category 1 root cause evaluation (RCE) has been initiated for this event. Upon completion, the approved recommendations from the RCE will be implemented through the corrective action program.

cc: U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SURRY POWER STATION , Unit 1	05000 - 280	1 OF 4

TITLE (4)

Reactor Trip on Lo Lo Steam Generator Level Due to Human Error

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 NAME	DOCUMENT NUMBER 05000-
10	24	00	2000	-- 004 --	00	11	21	00	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9): N

POWER LEVEL (10): 100 %

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. H. Blount II, Site Vice President	(757) 365-2001

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE	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 October 24, 2000, at 07:41, with Unit 1 at 100% power, control room annunciators alarmed indicating a low low steam generator (SG) level and an automatic reactor trip of Unit 1 occurred. Immediately prior to the trip, the main steam dump valves were observed to have received an open demand, and a loss of turbine load was exhibited. The event was caused by human error due to work being performed on the wrong unit. Work was being conducted on the Unit 1 Electrical Hydraulic Controls (EHC) System instead of the intended Unit 2 EHC System. Approved Root Cause Evaluation recommendations, designed to prevent the recurrence of a similar event, will be implemented. The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on October 24, 2000 at 10:30. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system.

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

1.0 DESCRIPTION OF THE EVENT

On October 24, 2000, at 07:41, with Unit 1 at 100% reactor power, control room annunciators [EIIS-IB] alarmed indicating a low low steam generator (SG) level condition in SG "C" [EIIS-AB,SG] and an automatic reactor trip [EIIS-JC] occurred as designed. Immediately prior to the trip, the main steam dump valves [EIIS-SB,V] were observed to have received an open demand, and a loss of turbine load was exhibited. The loss of turbine load was caused by human error due to work being performed on the wrong unit. It was determined that work was being conducted on the Unit 1 Electrical Hydraulic Controls (EHC) System [EIIS-TG]. The work was intended to be done on the Unit 2 EHC System.

All control rods [EIIS-AA] inserted into the core as designed. Individual Rod Position Indicator (IRPI) B8 rod bottom light was not received due to a burned out light bulb. The light bulb was replaced and indicated properly. The IRPI indication for this rod trended with all other IRPI indications, and indicated <20 steps. The shutdown margin for Unit 1 was determined to be satisfactory. Both motor driven auxiliary feedwater pumps [EIIS-BA-P] and turbine driven auxiliary feedwater pump automatically initiated on low low steam generator level to provide flow to the SGs as designed. The Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) armed and initiated upon receipt of low steam generator levels. Primary reactor coolant system (RCS) temperature decreased to approximately 543 degrees following the reactor trip.

No primary safety or power operated relief valves were actuated during the event. As a result of the loss of turbine load, the secondary power operated relief valves actuated during the transient.

All electrical busses [EIIS-EB,BU] transferred properly following the trip. Unit 1 was stabilized at Hot Shutdown with RCS temperature maintained at approximately 547 degrees.

Unit 2 was in a refueling outage and was not affected by this event. Unit 2 remained stable at Cold Shut Down (CSD) with RCS temperature at approximately 100^{o}F and the RCS depressurized to atmospheric pressure with containment Type A test conditions being established.

The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on October 24, 2000 at 10:30. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no safety consequences or implications. Appropriate operator actions were taken in accordance with emergency operating procedures to ensure the performance of system automatic actions and to respond to abnormal conditions. The unit was quickly brought to a stable, no-load condition. Therefore, the health and safety of the public were not affected at any time during this event.

3.0 CAUSE

A Category 1 Root Cause Evaluation (RCE) was initiated on October 24, 2000, to determine the cause of this event and to recommend corrective actions. The event was caused by human error due to work being performed on the wrong unit. A mechanic was instructed to replace relief valve 2-EH-RV-1 on the Unit 2 EHC System. The mechanic mistakenly started replacing the Unit 1 EHC relief valve 1-EH-RV-1. Upon loosening the bolts to the relief valve, the mechanic observed spraying EHC fluid and dumping of fluid through the relief valve to the EHC reservoir. Although the mechanic stopped loosening the bolts, and began re-tightening, all turbine stop valves went closed causing a Unit 1 automatic trip on steam generator low-low level.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the reactor trip, control room operators acted promptly to place the unit in a safe, shutdown condition in accordance with emergency and other operating procedures.

5.0 ADDITIONAL CORRECTIVE ACTIONS

None

6.0 ACTIONS TO PREVENT RECURRENCE

A RCE has been initiated and, upon completion, the approved recommendations from the RCE will be implemented through the corrective action program.

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

7.0 SIMILAR EVENTS

None

8.0 MANUFACTURER/MODEL NUMBER

N/A

9.0 ADDITIONAL INFORMATION

Unit 1 was returned to service on October 25, 2000.

Unit 2 was at Cold Shut Down and was not affected by this event.