POW 28-06-01-01

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)
SURRY POWER STATION, UNIT 2	0 5 0 0 0 2 8 1	1 OF 0 5

TITLE (4) Reactor Trip Due to Low Low Steam Generator Level Due to Closure of Turbine Governor Valves

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 5	1 6	8 8	8 8	0 1 0	0 0	0 6	1 5	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10): 1 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 AREA CODE

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	B A	P	I 0 7 5	YES					
X	A A	Z I	M 0 3 5	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH / DAY / YEAR

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

On May 16, 1988 at 0324 hours, with Unit 2 at 100% power, a reactor trip occurred as a result of steam generator (S/G) {EIIS-HX} low low level. At 49 seconds after the trip, safety injection {EIIS-BQ} was manually initiated in accordance with Emergency Procedure 1.0 due to pressurizer {EIIS-PZR} level decreasing to 13%. The cause of the reactor trip was rapid closure of the turbine governor valves {EIIS-SCV} which resulted in shrink of the S/G levels to the reactor trip setpoint.

Extensive testing was conducted on the turbine control circuitry and the EHC system. No deficiencies were detected. The turbine impulse pressure transmitter {EIIS-PT} was calibrated. Further monitoring of the EHC system will be performed during the unit startup.

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

1.0 Description of the Event

On May 16, 1988 at 0324 hours, with Unit 2 at 100% power, a reactor trip occurred as a result of steam generator (S/G) {EIIS-HX} low low level. At 49 seconds after the trip, safety injection {EIIS-BQ} was manually initiated in accordance with Emergency Procedure 1.0 due to pressurizer {EIIS-PZR} level decreasing to 13%. At 0333 hours, a notification of an unusual event was made due to the non-spurious Emergency Core Cooling System {EIIS-BQ} initiation.

Following the reactor trip, all safety systems functioned as designed with the following exceptions.

1. Auxiliary Feedwater (AFW) {EIIS-BA} flow to the "A" S/G was lower than expected. The flow was observed to be 227 gallons per minute (gpm) compared with 325 gpm each for "B" and "C" S/Gs.

2. Control rod {EIIS-ROD} M-10 indicated approximately 30 steps withdrawn after the trip, and the rod bottom light was not illuminated. Over the course of the event, the indication slowly decreased to zero steps and the rod bottom light illuminated.

3. A greater than expected Reactor Coolant System (RCS) {EIIS-AB} cooldown was noted. (Actual minimum temperature was 537 degrees Fahrenheit, expected temperature was 547 degrees Fahrenheit).

2.0 Safety Consequences and Implications

During this event, the reactor protection system functioned as designed to trip the reactor on low low S/G level.

The UFSAR accident analysis for loss of normal feedwater assumes a total AFW flow of 520 gpm. During this event, the available AFW flow was greater than 875 gpm.

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

In addition, all safety systems remained operable during the event and plant parameters remained well within the bounds of the accident analysis. Therefore, the health and safety of the public were not affected.

3.0 Cause

The cause of the reactor trip was rapid closure of the turbine governor valves {EIIS-SCV} which resulted in shrink of the S/G levels to the reactor trip setpoint. After an in-depth investigation, the governor valve closure is suspected to have been caused by a spurious signal from the load drop anticipation circuit in the Electro Hydraulic Control (EHC) {EIIS-TG} system.

The reduced AFW flow to the "A" S/G was due to a metallic object partially blocking the cavitating venturi {EIIS-OR} in the "A" AFW line. The metal piece was determined to be from the channel ring vane from motor driven AFW pump 2-FW-P-3B {EIIS-P}.

Control rod M-10 was verified to be inserted completely and it was determined that there was a faulty indication from the Individual Rod Position Indicator (IRPI) {EIIS-ZI} for rod M-10.

The pressurizer level setpoint for the no load RCS average temperature (Tave) of 547 degrees Fahrenheit is approximately 22%. The 10 degree drop in primary temperature that was experienced is equivalent to approximately a 10% level decrease in the pressurizer which would account for post trip pressurizer level of approximately 13%. The RCS cooldown below the no load Tave of 547 degrees Fahrenheit has been noticed on previous reactor trips. A heat balance calculation was performed and it was determined that the post trip heat removal capability, due to main feedwater flow {EIIS-JE}, AFW flow and steam dump valves {EIIS-PCV}, was more than the heat removal required to balance the core decay heat.

★U.S.GPO:1986-0-624-538/455

4.0 Immediate Corrective Action(s)

Operators followed appropriate plant procedures and quickly stabilized the unit following the reactor trip/safety injection. Also, the shift technical advisor performed the critical safety function status tree review to ensure specific plant parameters were noted and that those parameters remained within safe bounds.

5.0 Additional Corrective Action(s)

Unit 2 was placed in cold shutdown in order to inspect the AFW system. The AFW piping was visually inspected using fiber optics. Two metal pieces were located and removed. One was located upstream of the cavitating venturi in the line to "A" S/G and a second at a flow measuring orifice in the "C" S/G AFW line. An inspection of the 3 AFW pumps was conducted. It was determined that 8 small pieces were broken off from the channel ring vane tips from the "B" motor driven AFW pump (2-FW-P-3B). The two pieces that were retrieved from the AFW piping were verified to have come from 2-FW-P-3B. The pump was repaired by replacing the rotating assembly. The "A" motor driven AFW pump (2-FW-P-3A) was inspected and one chip was missing from a channel ring vane tip. A non destructive examination determined that no cracks in the channel ring vanes existed. The channel ring vanes were polished to reduce the possibility of cracking. The turbine driven AFW pump (2-FW-P-2) was disassembled and numerous channel ring vanes had chips missing from the outer vane ends. A non destructive test revealed several cracked channel ring vanes. The defective assemblies were replaced. The channel ring vanes were polished as required.

Extensive inspection of the system piping did not locate the small pieces broken from the AFW pumps. A 10CFR50.59 analysis was performed, and concluded that the presence of a number of small metallic pieces in the S/Gs would not adversely affect the S/G performance. In addition, strainers {EIIS-STR} have been installed upstream of the cavitating venturies to prevent obstruction of the venturi by foreign materials. A test was conducted and adequate AFW flow was verified prior to returning the system to service.

Extensive testing was conducted on the turbine control circuitry and the EHC system. No deficiencies were detected. The turbine impulse pressure transmitter {EIIS-PT} was calibrated. Further monitoring of the EHC system will be performed during the unit startup.

The IRPI signal conditioner module for rod M-10 was replaced and calibrated. A rod drop test demonstrated that the rod was inserting as designed, within the time requirements of the Technical Specifications.

An evaluation will be conducted to determine if changes are desirable to the steam dump circuitry to minimize the cooldown following a reactor trip.

6.0 Action(s) Taken to Prevent Recurrence

A recorder will be used to monitor the EHC controller during startup. The strainers installed in the AFW system should preclude any further obstructions of the cavitating venturies.

7.0 Similar Events

During the manual reactor trip on March 27, 1988 (Unit 2 LER 88-004), a similar decrease in AFW flow to the "A" steam generator was noted. Extensive inspections at that time did not locate any metal pieces similar to those found during the present inspection.

8.0 Manufacturer/Model Number

Auxiliary Feedwater Pumps:

2-FW-P-3A, B	Ingersoll Rand/3HMTA8
2-FW-P-2	Ingersoll Rand/4HMTA6

Individual Rod Position Indicator:

M-10	Magnetics Inc./Project K931, K932

VIRGINIA ELECTRIC AND POWER COMPANY
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

June 16, 1988

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

Serial No.: 88-026
Docket No.: 50-281
Licensee No.: DPR-37

Gentlemen:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company submitted the following Licensee Event Report (LER) for Surry Unit 2 on June 15, 1988.

REPORT NUMBER

88-010-00

This report was inadvertently issued without the inclusion of page 1 of 5. Consequently, the report is being reissued in its entirety.

Very truly yours,

David L. Benson

David L. Benson
Station Manager

Enclosure

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