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 1	0 5 0 0 0 2 8 0	1 OF 0 3

TITLE (4)

REACTOR TRIP ON LOW RCS FLOW DUE TO FAILURE OF LOOP STOP VALVE

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 0 0 0
0 5	1 6	8 7	8 7	— 0 1 1 —	0 1	1 0	1 2	8 7		0 5 0 0 0

OPERATING MODE (9)		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)
POWER LEVEL (10)	1 0 0	[ ] 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	
D. L. Benson, Station Manager	8 0 4	3 5 7 - 3 1 8 4

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	A B	I S V	A 3 9 1	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 fifteen single-space typewritten lines)* (16)

On May 16, 1987 at 0824 hours, with Unit 1 at 100% power, a low flow reactor trip occurred when "A" loop reactor coolant system (RCS) (EIIS-AB) flow decreased to 47%. Following the reactor trip, the source range channels (EIIS-DET) did not automatically reinstate. All other protection and control systems functioned properly. Operators followed appropriate plant procedures and stabilized the plant following the reactor trip. This event occurred when the "A" hot leg loop stop valve (EIIS-ISV) stem failed, permitting the disc to drop, partially blocking loop flow. A metallurgical analysis was performed which determined that the stem failure was attributed to stress corrosion cracking. The stress corrosion cracking was due to a combination of excessive backseating force and a process of thermal aging. At the next outages of sufficient duration, samples from the Unit 1 and Unit 2 stems will undergo metallurgical analysis, and the Unit 2 stems will undergo ultrasonic testing as was previously performed on Unit 1. The failure of source range channels to reinstate was due to the under compensation of the intermediate range channel NI-36. The source range channels were manually reinstated, and technicians readjusted the intermediate range compensating voltage.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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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, 1987 at 0824 hours, with Unit 1 at 100% power, a low flow reactor coolant system (EIIS-AB) (RCS) flow decreased to 47%. Following the reactor trip, the source range channels (EIIS-DET) did not automatically reinstate.

All other protection and control systems functioned properly. Operators followed appropriate plant procedures and stabilized the plant following the reactor trip.

2.0 Safety Consequences and Implications

The low flow reactor trip automatically trips the reactor to maintain sufficient margin above a DNBR of 1.3 with a loss of RCS flow. The complete loss of flow in one loop from a reactor power of 100% (2441 MWt) with three loops operating is an analyzed event. During the event, "A" loop flow was maintained at approximately 47% and total core flow remained at 84%. A confirmatory analysis performed by Nuclear Engineering concluded that DNBR was maintained above the accident analysis value of 1.3. In addition, all other safety related systems remained operable during the event, and plant parameters remained well within the bounds of the accident analysis. Therefore, this event did not constitute an unreviewed safety question, and the health and safety of the public were not affected.

3.0 Cause

The cause of this event was a failure of the "A" RCS hot leg loop stop valve (MOV-1590) (EIIS-ISV). The valve stem failed, permitting the disc to drop, partially blocking the loop flow.

The valve stem failure was attributed to stress corrosion cracking. The stem material was analyzed by a contractor laboratory and determined that it was of a metallurgical composition which was conducive to the process of thermal aging when exposed to elevated temperatures (above 550 degrees Fahrenheit). The material was 17-4 PH stainless steel heat treated to H-1100. The resulting embrittlement due to thermal aging, coupled with high stress due to excessive backseating force, reduced the resistance to stress corrosion cracking.

The failure of the source range channels to reinstate was due to the under compensation of the intermediate range channel NI-36.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

4.0 Immediate Corrective Actions

The Operators performed all appropriate emergency procedures and function restoration procedures to ensure the plant was returned to a stable condition. This included manually reinstating the source range channels.

Also, the STA 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 Actions

The unit was placed in the cold shutdown condition, and the stem of the hot leg loop valve was replaced. The stems of the other five Unit 1 loop stop valves were ultrasonically tested and found to be satisfactory.

Technicians readjusted the intermediate range compensating voltage.

6.0 Actions Taken to Prevent Recurrence

At the next outages of sufficient duration, samples of the loop stop valve stems from Unit 1 and Unit 2 will undergo metallurgical analysis. Additionally, the Unit 2 stems will undergo ultrasonic testing as was previously performed on Unit 1.

During the Unit 1 start-up following the hot leg stem replacement, the loop stop valves were not placed on their backseats. Due to leakage through the packing, the valves were later backseated to a maximum deflection of 1/16 inch according to the manufacturer's recommendations.

7.0 Similar Events

A similar failure occurred on the Unit 1 "B" loop Hot Leg Isolation Valve on December 1, 1973.

8.0 Manufacturer/Model Number

Anchor Darling/Drawing Nos. 95-11778 and 95-11779.

USNRC-DS

1987 OCT 19 A 10:02

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

October 12, 1987

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

Serial No.: 87-013A
Docket No.: 50-280
Licensee No.: DPR-32

Gentlemen:

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

REPORT NUMBER

87-011-01

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be reviewed by Safety Evaluation and Control.

Very truly yours,

Harry L Miller
for David L. Benson
Station Manager

Enclosure

cc: Dr. J. Nelson Grace
Regional Administrator
Suite 2900
101 Marietta Street, NW
Atlanta, Georgia 30323

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