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)

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

TITLE (4)

Manual Reactor Trip in Response to Main Feedwater Regulating Valve Failure

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	DOCKET NUMBER 05000 --
11	26	1998	1998 –	014 --	00	12	16	1998	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)	81%		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 or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
E. S. Grecheck, Site Vice President	(757) 365-2000

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
B	SJ	V	Bailey Controls Division	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 15 single-spaced typewritten lines) (16)

On November 26, 1998, at 21:08, the Unit 1 steam generator (SG) "B" main feedwater regulating valve, 1-FW-FCV-1488, failed closed. A control room annunciator alarmed indicating a difference between the SG "B" feedwater and steam flow parameters. In response, the control room operator placed 1-FW-FCV-1488 in the manual control mode and attempted to open the valve. 1-FW-FCV-1488 remained closed, however, and the "B" SG level continued to decrease. To avert an automatic reactor trip due to low SG level coincident with a steam/feedwater flow mismatch, the control room operator initiated a manual reactor trip. The Unit 1 reactor tripped from 81% power and was followed by an automatic turbine trip, as designed. 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 event was caused by the dislocation of a retaining clip in the positioner pilot valve, which caused 1-FW-FCV-1488 to fail closed. 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 November 26, 1998 at 23:35. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv).

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

1.0 DESCRIPTION OF THE EVENT

On November 26, 1998, at 21:08, the Unit 1 steam generator (SG) [EIIS-AB,SG] "B" main feedwater regulating valve [EIIS-SJ,FCV], 1-FW-FCV-1488, failed closed. A control room annunciator [EIIS-IB] alarmed indicating a difference between the SG "B" feedwater and steam flow parameters. In response, the control room operator placed 1-FW-FCV-1488 in the manual control mode and attempted to open the valve. 1-FW-FCV-1488 remained closed, however, and the "B" SG level continued to decrease. To avert an automatic reactor trip due to low SG level coincident with a steam/feedwater flow mismatch, the control room operator initiated a manual reactor trip [EIIS-JC]. The Unit 1 reactor tripped from 81% power and was followed by an automatic turbine trip [EIIS-TA,TRB], as designed.

The auxiliary feedwater pumps [EIIS-EIIS-BA-P] started on low-low SG water level and provided flow to the SGs. The Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) [EIIS-JE] armed and initiated, as designed. The main steam dump valves [EIIS-SB,TCV] automatically opened to admit steam to the main condenser.

The RCS reached a minimum temperature of approximately 538°F and subsequently stabilized at 547°F. The reactivity shutdown margin was calculated following the RCS cooldown to ensure that Technical Specification and administrative shutdown margin limits were satisfied.

The following discrepancy was noted during the post-trip response:

- Moisture separator reheater control valve, 1-MS-FCV-104D [EIIS-FCV], failed to fully close.

The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on November 26, 1998 at 23:35. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the 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.

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3.0 CAUSE

A Category 1 Root Cause Evaluation (RCE) was initiated on November 27, 1998, to determine the cause of this event and to recommend corrective actions. The RCE has preliminarily concluded that a retaining clip in the positioner pilot valve became dislocated, which caused 1-FW-FCV-1488 to fail closed. The RCE is continuing to investigate the cause of the retaining clip dislocation.

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.

The Shift Technical Advisor monitored the critical safety function status trees to ensure that plant parameters remained acceptable.

5.0 ADDITIONAL CORRECTIVE ACTIONS

1-MS-FCV-104D was examined and found to be held in a partially open position as a result of the stem binding. The stem was cleaned and the valve was tested satisfactorily.

The pilot valve assembly for 1-FW-FCV-1488 and the retaining clip in the positioner were replaced. 1-FW-FCV-1488 was subsequently tested satisfactorily.

The RCE team is evaluating unit conditions and systems response contributing to the RCS cooldown following the reactor trip.

6.0 ACTIONS TO PREVENT RECURRENCE

Approved RCE recommendations, designed to prevent the recurrence of a similar event, will be implemented.

7.0 SIMILAR EVENTS

LER 50-281/1990-003-00 Manual Reactor Trip Due to Failure of "A" Main Feedwater Regulating Valve

LER 50-281/1991-011-00 High Steam Generator Level Due to Main Feedwater Regulating Valve Oscillations Results in ESF Actuations and Reactor Trip

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8.0 MANUFACTURER/MODEL NUMBER

Bailey Controls Division
AV1 Series Positioner

9.0 ADDITIONAL INFORMATION

Unit 1 was returned to service on November 27, 1998.

Unit 2 was operating at 100% power and was not affected by this event.