NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

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 DURING AUXILIARY FEED PUMP TESTING

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 6	1 9	8 4	8 4	0 1 6	0 0	0 7	1 7	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 1 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)	☑ 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)	
R. F. SAUNDERS, 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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B A	I S V	L 2 0 0	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 June 19, 1984, with the unit just less than 10% power, a reactor trip resulted when 2 of 4 nuclear power channels, NI 44 and NI 41 exceeded 10% power with the turbine unlatched. A primary plant cooldown of approximately .8^{0}F/min. and a primary dilution of 58 ppm contributed to the power increase.

Following the trip, all control and protection systems functioned as expected. Main steam was isolated and the turbine stop valves were closed to limit primary plant cooldown.

Precautions will be added to station procedures (OP-1.4 and PT-15.1C) to prevent testing the steam driven auxiliary feedwater pump near the P-10 setpoint without the main turbine being latched.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
SURRY POWER STATION, UNIT 1	0 5 0 0 0 2 8 0	8 4	— 0 1 6 —	0 0	0 2 OF 0 3

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

1. Description of the Event

On June 19, 1984, with the unit just less than 10% power, a reactor trip resulted when 2 of 4 nuclear power channels, NI 44 and NI 41 exceeded 10% power with the turbine unlatched.

Prior to exceeding 10% power, the following occurred in chronological order:

1) A primary dilution of 58 ppm started 2 hours prior to the trip and ended one half hour prior to the trip.

2) Turbine stop valve leakoff isolation valves were open 7 minutes prior to the reactor trip.

3) Periodic testing had been completed on the turbine driven auxiliary feed pump (1-FW-P-2) 6 minutes prior to the reactor trip. During the testing, isolation valve MOV-FW-151D to "B" steam generator failed to fully close, allowing colder auxiliary feedwater to enter the generator. The flow of auxiliary feedwater coupled with the steam required to run the steam driven feed pump, started a primary cooldown of approximately .8°F/min.

4) The instrument drawer for NI 42 was closed about the time of this event.

Following the trip, a review of rod position traces and interviews with the control room operators confirmed no rod movement took place. All control and protection systems functioned as expected. Main steam was isolated and the turbine stop valve leakoff valves were closed to limit primary plant cooldown. Operators followed appropriate plant procedures and quickly stabilized the plant following the trip.

2. Safety Consequences and Implications

This event resulted in a spurious reactor trip. All safety systems functioned as designed. Also, this event could not happen at a higher power level because the turbine would have been latched. Therefore, an unreviewed safety question is not created and the health and safety of the public were not affected.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
SURRY POWER STATION, UNIT 1	0 5 0 0 0 2 8 0	8 4	— 0 1 6	— 0 0	0 3 OF 0 3

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

3. Cause

The automatic reactor trip was initiated from exceeding 10% power on 2 of 4 nuclear power channels, NI 44 and NI 41, with the turbine unlatched.

A primary plant cooldown of approximately .8^{0}F/min. and a primary dilution of 58 ppm contributed to the power increase.

Based upon a detailed review of the Post Trip data and repeated attempts to create an electronic spike, the closing of the N-42 instrument drawer has been ruled out as a possible cause.

4. Immediate Corrective Action

Operators performed all appropriate emergency procedures and function restoration procedures to ensure the plant was returned to a stable condition. This included starting a second charging pump, isolating main steam and securing the turbine stop valve leakoff isolation valves.

Also, the STA performed the critical safety status tree reviews to ensure specific plant parameters were noted and appropriate procedures were used to maintain those parameters within safe bounds.

5. Additional Corrective Actions

A broken gear pin was found to have caused the malfunction of MOV-FW-151D. The gear box to MOV-FW-151D was replaced and the valve was returned to service.

6. Action Taken to Prevent Recurrence

Precautions will be added to station procedures (OP-1.4 and PT-15.1C) to prevent testing the steam driven auxiliary feedwater pump near the P-10 setpoint without the main turbine being latched.

7. Generic Implications

None.