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

TITLE (4)
Manual Reactor Trip Following "A" Main Steam Trip Valve Closure

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 1	1 3	8 4	8 4	0 0 1	0 0	0 2	1 3	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N
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)	[√] 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	
J. L. Wilson - 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	H C V	L 2 0 0	Y					
X	I G	R I	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[√] NO				

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

On January 13, a manual reactor trip was initiated upon closure of "A" main steam trip valve. Unwarranted closure of "A" main steam trip valve was due to a hairline crack in an air line pipe nipple on the valve's actuator. The reduced air pressure in the actuator allowed the disc to be deflected and closed by main steam. It is suspected that the nipple failure was due to mechanical fatigue.

The leaking nipple was replaced and the trip valve Instrument Air Lines for all main steam trip valves were inspected for leaks. Testing was performed to verify acceptable closure time for the main steam trip valves and prove the safeguards signal to the valves.

NRC Form 366A
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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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SURRY POWER STATION	0 5 0 0 0 2 8 1	8 4	— 0 0 1	— 0	) 3

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

1. Description of the Event

On January 13, 1984, with unit 2 at 100% power, the control room operator observed the annunciator for steam line isolation trip valve closure and the closed valve position indicator for "A" main steam trip valve (EIIS No. ISV). Upon receiving these indications, a manual reactor trip was initiated in anticipation of an automatic reactor protection system actuation.

Following the trip, all control and protection systems functioned as expected with the exception of the following:

1) Auxiliary feed MOV-FW-251F (EIIS No. HCV) would not manually remain closed following an automatic start on the auxiliary feed pump.

2) Vessel leakoff temperature increased. (EIIS No. RPV).

3) The source range (EIIS No.RI) would not automatically re-instate.

Operators followed appropriate plant procedures and quickly stabilized the plant following the trip.

2. Safety Consequences and Implications

The purpose of the main steam trip valve is to close immediately in the case of a rupture in the main steam line between the valve and the turbine, thus preventing rapid flashing and blowdown of the shell side of the steam generator. During this event, the unit was not experiencing a main steam line rupture. In addition, all other safety related systems remained operable during the event and plant parameters remained within the bounds of the accident analysis. Therefore, this event did not constitute an unreviewed safety question nor affect the health and safety of the public.

3. Cause

The manual reactor trip was initiated due to closure of "A" main steam trip valve. Unwarranted closure of "A" main steam trip valve was due to a hairline crack in an air line pipe nipple on the valve's actuator. The reduced air pressure in the actuator allowed the disc to be deflected and closed by main steam. It is suspected that the nipple failure was due to mechanical fatigue.

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

Secondary cause failures are:

1) Failure of auxiliary feed MOV-FW-251F has been attributed to an automatic control relay malfunction.

2) The increasing vessel leakoff temperature has been attributed to a leaking reactor vessel head inner seal.

3) The failure of the automatic re-instatement of the source range channels NI 31 and 32 was due to undercompensation of the Intermediate Range Channel NI-36.

4. Immediate Corrective Action

Operators performed all appropriate Emergency Procedures and functio restoration procedures to ensure the plant was returned to a stable condition. This included isolating the inner head seal leak off and manually reirstating the source range channels.

Also, the STA performed the 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

The leaking nipple was replaced and the trip valve Instrument Air Lines for all main steam trip valves were inspected for leaks. Testing was performed to verify acceptable closure time for the main steam trip valves and prove the safeguards signal to the valves. A minor adjustment was performed on the Auxiliary Feedwater automatic control relay (Agastat) and the valve was cycled satisfactorily. NI-36 was correctly compensated following the trip.

6. Action Taken to Prevent Recurrence

The automatic control relay for the auxiliary feed valve will be replaced when a spare relay is received. In addition, the inner reactor vessel head seal is scheduled for replacement during the next refueling outage. Maintenance and testing procedures for the intermediate range will be investigated.

7. Generic Implications

None.