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)
Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	1 OF 0 4

TITLE (4): Reactor Trip Due to Isolation of Incorrect Transmitter

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

OPERATING MODE (9): 1

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)	X 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)	50.72(b)(2)(ii)
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)	
Roger W. Ouellette, Associate Engineer - Licensing	7 0 4	3 7 3 - 7 5 3 0

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

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 July 10, 1985, at 1546 hours, a reactor trip occurred on Low Low Steam Generator (S/G) Level while personnel were performing 18 month surveillance calibrations on a S/G D Main Steam pressure transmitter. After the technician in the Control Room had placed S/G D Channel II steam pressure bistable in a test condition, he relayed to another technician in the field the incorrect transmitter mark number for isolation. The incorrectly isolated transmitter was for the Main Steam pressure loop of S/G A. Consequently, the indication of zero pressure in the steam loop prompted the modulation of the S/G A Main Feedwater control valve. The S/G A water level decreased to the low low setpoint and the reactor tripped from 100% full power. Operations entered the Unit Fast Recovery Procedure. The incorrect transmitter was returned to service, the correct transmitter was isolated, and the surveillance calibration was completed.

This incident is classified as a Personnel Error. While performing the calibration procedure, the technician communicated the incorrect transmitter number to the technician in the field.

The incident is reportable per 10 CFR 50.72, Section (b)(2)(ii) and 10 CFR 50.73, Section (a)(2)(iv).

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

Procedure IP/1/A/3222/79D, Steam Pressure Loop D Channel 2, Protection Cab. 2 1SMPT5180 PT-545, is a surveillance calibration of the Main Steam pressure transmitter conducted on an eighteen month basis.

The function of the pressure transmitters on channels 1 and 2 is to control the input of feedwater into the S/G, correcting any mismatch of steam flow/feedwater flow, by modulating the Main Feedwater control valves.

During calibration, IP/1/A/3222/79D requires switching feedwater control to the channel opposite the one being calibrated. This eliminates the possibility of a steam flow/feedwater flow mismatch.

On July 10, 1985, Unit 1 was operating at 100% full power. At 1540 hours, personnel began calibration of a S/G D Main Steam pressure transmitter per IP/1/A/3222/79D. In order to carry out this calibration, a technician was located in the Control Room at the 7300 process cabinets directing the calibration procedure, and another technician was located in the Exterior Doghouse to isolate and apply inputs to the transmitter. To conduct this activity, telephone communication was established between the Control Room and the Exterior Doghouse.

The procedure for calibration was followed correctly up to and including Section 10.4.4. Upon performing Section 10.4.5, transmitter 1SMPT5180 was to be isolated for testing. Technician A instructed Technician B to isolate transmitter 1SMPT5080. This was the mark number for S/G A pressure transmitter, instead of the S/G D pressure transmitter. Technician B verified the communication by repeating back to Technician A the number 1SMPT5080. Technician A gave Technician B an affirmative verification. Technician B then isolated transmitter 1SMPT5080.

When 1SMPT5080 was isolated, the Process Control System received an indication of zero pressure in the S/G A Steam Loop. In order to correct the steam flow/main feedwater flow mismatch, Main Feedwater control valve 1CF27 began to modulate, reducing the feedwater input. Since steam was still being generated at 100% full power, the level of the water in S/G A began to decrease, and eventually reached the programmed 55% trip level (when the unit is operating at 100% full power). This caused the Reactor Trip at 1546:46. The Reactor Trip caused the Main Turbine Trip and the Motor Driven Auxiliary Feedwater (CA) Pumps to auto-start. At 1546:52 hours, the Main Feedwater Pump Turbines A and B tripped on low Tave and reactor trip. At 1547:10 hours, the Main Feedwater isolation valves 1CF33, 1CF42, 1CF51, and 1CF60 closed. Operations entered the Fast Recovery Procedure at 1600 hours. Main Feedwater Pump Turbine 1B was restarted. The Motor Driven Auxiliary Feedwater Pumps were secured. On July 12, 1985, at 0152 hours, the unit reentered Mode 1.

At 1546:47 S/G D initiated a Low Low Level Reactor Trip. S/G A was in the low low level state. At 1546:52 the Turbine Driven Auxiliary Feedwater Pump number 1 auto-started due to two-out-of-four S/G's being at low low level.

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

t 1547:15 the Motor Driven CA Pump A discharge pressure reached the low alarm level. No evidence exists showing the pump actually had low discharge. Indications are that there is a discharge pressure indicator problem in the system.

On July 10, 1985, at 1816 hours the CF side of Feedwater Heaters 1A2 and 1B2 were isolated due to the failure of relief valve 1CF71 on heater 1B2. The failure of the valve occurred after the trip, at the pipe/shell interface. The failure was due to fatigue created by the inability of the valve to reseat.

S/G B PORV 1SV13 experienced a peak pressure of 1139 psig without opening. The set point for this PORV is 1125 psig. LER 413/85-43 references this same valve not opening when subjected to 1175 psig. Upon investigation, the valve was found to have a setting of about 60 psig too high. The setpoint was reset and the valve responded within 10 psig of the setpoint. In this incident the pressure exceeded the setpoint by 14 psig. This is acceptable due to a tolerance of $\pm 3\%$ of the setpoint (as specified in the FSAR).

This event is classified as a Personnel Error. The Reactor Trip was caused by a Low Low Water Level in S/G A. The low low level occurred when the incorrect transmitter was isolated due to personnel miscommunication. Efforts to prevent recurrence of this problem include providing documentation to the technician in the field for verification of the transmitter numbers and providing better instrument identification in the field by labelling instrumentation.

The initial opening of relief valve 1CF71 was caused by a pressure spike created by the closing of the Main Feedwater isolation valves. The present system shuts the isolation valves 4 seconds after Main Feedwater pump trip. If the Main Feedwater pumps are allowed to coast down over a longer period of time before closing the isolation valves, the pressure spike would be reduced. Presently, the relief valve is being repaired.

CORRECTIVE ACTION

1) Transmitter 1SMPT5080 was returned to service, transmitter 1SMPT5180 was isolated and procedure IP/1/A/3222/79D was completed.

2) Operations started Unit Fast Recovery Procedure OP/1/A/6100/05.

3) A work request was written to investigate low discharge pressure indication on Motor Driven CA Pump A.

4) A method will be investigated to provide documentation in the field for the instruments being calibrated.

5) Better instrument identification will be provided in the field for those instruments which have trip or reliability potential. Re-labelling of the transmitter numbers will begin with those instruments related to the S/G's and continue as routine calibrations are performed.

6) The event will be discussed with all appropriate technicians.

SAFETY ANALYSIS

Following the Reactor Trip due to Low Low S/G A Level, power immediately decreased to zero. Pressurizer pressure increased to 2345 psig and Pressurizer Operated relief valves (PORV's) opened and reclosed as necessary to relieve pressure. The pressurizer level decreased to a level of 19% and stabilized at 21%. Tave stabilized at 556 degrees F within 30 minutes. At the time of the Reactor Trip a steam pressure spike was experienced. The spike diminished immediately upon lifting of S/G C PORV 1SV7. Peak pressure for 1SV7 was 1125 psig. S/G B pressure reached 1139 psig. Lifting of PORV 1SV13 did not occur. 1139 psig is within +3% of the setpoint and is therefore acceptable. S/G A and D (PORV) did not open due to pressure never reaching the setpoint. The pressure never dropped below the 1000 psig post trip level. Adequate core heat removal was available through the S/G's in the post trip mode.

The health and safety of the public were not affected by this incident.

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VICE PRESIDENT
NUCLEAR PRODUCTION

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August 9, 1985

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

Subject: Catawba Nuclear St on, Unit 1
Docket No. 50-413

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/85-45 concerning a reactor trip due to the isolation of an incorrect transmitter. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker/-RO

Hal B. Tucker

RWO:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
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Catawba Nuclear Station

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