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 a Defective Procedure

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 4	1 9	8 6	8 6	— 0 2 2 —	0 0	0 5	1 9	8 6	N/A	0 5 0 0 0
										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)	50.72(b)(2)(ii)
[ ] 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	
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)		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 April 19, 1986, at 1341:33 hours, the Unit 1 Reactor tripped when all three flow transmitters on loop C of the Reactor Coolant (NC) System indicated a low flow. The low flow indication was due to the loss of pressure on the high pressure side of the transmitters. At the time of the incident, personnel were bleeding the high pressure line of the Channel 2 flow transmitter on NC loop C while returning it to service. The unit was at 100% power at the time of this incident.

This event is assigned Cause Code D, Defective Procedure, due to the procedure being used to calibrate the transmitter lacking the proper warning that the three high pressure flow transmitters share a common instrument line.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv) and 10 CFR 50.72, Section (b)(2)(ii).

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

BACKGROUND

Three flow transmitters are provided on each of the four Reactor Coolant System (EIIS:AB) loops. Each transmitter has an individual low pressure line running from the loop to the transmitter. All three transmitters for each loop receive a high pressure signal from a common instrument tap. The transmitters monitor the Reactor Coolant flow by measuring the pressure drop generated by flow through the loop. Each of these transmitters supplies an input to the protection portion of the Reactor Protection System (EIIS:JC). In order to receive a low flow signal, 2 of 3 of the Channels in 2 of 4 loops must register low fluid flow.

DESCRIPTION OF INCIDENT

On April 19, 1986, personnel were performing procedure IP/1/A/3222/15B on the Reactor Coolant Loop C, Flow Channel 2 Transmitter per a Work Request. Per the procedure, the low pressure side of the transmitter was used to calibrate the transmitter. The transmitter was placed back into service with the High and Low pressure valves open and Equalize valve closed. When the transmitter did not show a correct differential pressure, it was suspected that there was air in the high pressure line. Under the impression they were isolated on this transmitter, the personnel opened the test tee on the high pressure line. Since a common instrument line was shared by all three transmitters the high pressure was lost on the transmitters, satisfying the 2 out of 3 logic.

At 1341:33:867 hours, a Lo Flow in Reactor Coolant Loop C alarm caused the Reactor to trip. At 1341:33:931 hours, Reactor Main Trip Breaker B tripped, and at 1341:33:935 hours, Reactor Main Trip Breaker A tripped. At 1341:34:828 hours, Steam Generator (S/G) D Lo Lo Level occurred, at 1341:34:846 hours, S/G A Lo Lo Level occurred, at 1341:34:868 hours, S/G C Lo Lo Level occurred, and at 1341:34:920 hours, S/G B Lo Lo Level occurred. At 1341:34:449 hours, S/G B Lo Lo Level returned to normal, at 1341:35:483 hours, S/G D Lo Lo Level returned to normal, at 1341:35:495 hours, S/G C Lo Lo Level returned to normal, and at 1341:35:511 hours, S/G A Lo Lo Level returned to normal. At 1341:37 hours, Main Steam Bypass to condenser Control Valves, 1SB003, 1SB012, 1SB015, 1SB0121, 1SB024, and 1SB027 opened, the S/G Blowdown Containment Isolation valves closed, and Auxiliary Feedwater (CA)(EIIS:BA) Motor Driven Pump B started automatically. At 1341:40 hours, the Turbine Driven Auxiliary Feedwater Pump (CAPT) started. At 1341:47 hours, Main Steam Bypass to Condenser Control Valves closed automatically. At 1341:54 hours, Main Feedwater (CF) (EIIS:SJ) Pumps A and B tripped automatically. At 1341:58 hours, CF Isolation occurred. At 1341:07 hours, Main Condensate Booster Pumps A and B tripped automatically. At 1343:23 hours, C2 Heater Drain Pump was secured. At 1343:26 hours, Main Condensate Booster Pump C tripped automatically. At 1344:37 hours, the CAPT was secured. On April 20, 1986, at 1545 hours, prior to entering Mode 2, Startup, the Reactor tripped on a High Flux Source Range actuation of the Engineered Safeguard Features (see LER 413/86-23).

CONCLUSION

This incident is assigned Cause Code D, Defective Procedure. The procedure used to calibrate the transmitter did not contain an appropriate warning that the transmitter was not completely isolated on the high pressure side. Responsible technicians are instructed on how to vent air from transmitters. The method that was used was appropriate.

During this incident, the CA System flow to S/G B over-ranged and caused a spurious signal to the Safety Parameter Display System (SPDS). This problem was previously identified (see LER 413/86-06).

1EMF53A, Containment Post LOCA Radiation Monitor, indicated activity of greater than 3R/hr for 15 seconds during this event. This EMF has been giving spurious alarms in the past. Personnel performed a source check on the EMF to ensure its operability.

Main Steam to Condenser Dump Valves, 1SB006, 1SB009, and 1SB018, did not respond during this incident. Work Requests were previously written to investigate and repair these valves. In addition, valve 1SB027 did not fully open. A previous work request was written to investigate and repair this valve.

As in two previous Reactor Trips, LER's 413/85-67 and 413/86-06, overpressurization of the CF suction piping occurred. The pressure reached a value of about 1050 psig, which is in excess of the guidelines setforth in ANSI B31.1 and ASME code, Section 8 for piping design. Although the design pressures have been exceeded, the pressure reached during the transient was less than the system hydro test pressure of 1298 psig. Since the transient pressure has been reached only three times, there has been no adverse effect upon the piping system. A solution to this problem has been identified which could be carried out by the Control Room personnel. In this incident, the C2 Heater Drain Pump was not tripped until 2 minutes after the Reactor Trip, which was after the overpressurization occurred. If this solution is to be effective, the pump must be manually tripped as soon after the Reactor Trip as possible. A Station Problem Report is presently being reviewed so an electrical actuation to trip this pump can be incorporated at the 2nd refueling outage. The need for an immediate temporary modification to electrically trip the C2 Heater Drain Tank Pump upon a trip of both Main Feedwater Pumps is being assessed.

The CA flow to the S/G's was sporadic and was apparently caused by the operator throttling the control valves. No effect was seen from this action except CA flow fluctuations. After the Reactor Trip, all four S/G's registered a Lo-Lo Level alarm in the Alarm Typer. Immediately, the Lo Lo Level alarms for all four S/G's returned to a normal status. This indicates that a spurious spike was received on the narrow range S/G level indicators. This produced an unnecessary start of the CAPT and required additional CA throttling by the Operators.

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

Procedures IP/1/A/3222/15A, B, and C have been revised to add a precautionary statement about venting the high pressure line of a transmitter with the high pressure line valved in.

CORRECTIVE ACTION

(1) Personnel closed the test tee, re-establishing pressure level in the high pressure line.

(2) The station entered the procedure for Unit Fast Recovery, OP/1/A/6100/05.

(3) Procedures IP/1/A/3222/15A, B, and C were revised to add a precautionary statement about venting the high pressure line of a transmitter with the high pressure line valved in.

(4) An investigation will be undertaken to determine the cause of the spurious S/G Lo Lo Level indication signal.

(5) An investigation of the need for an immediate temporary modification to electrically trip the C2 Heater Drain Tank Pump upon loss of both Main Feedwater Pumps will be undertaken.

SAFETY ANALYSIS

Following the Reactor Trip due to indicated Low Flow in loop C of the Reactor Coolant System, power immediately decreased to zero. Pressurizer (PZR) pressure increased to 2246 psig, dropped to 1998 psig and stabilized at 2230 psig. PZR level dropped and stabilized at 25%. Tave decreased and stabilized at 558 degrees F. Main Feedwater isolation occurred due to low Tave. Main Steam Bypass to Condenser Valves 1SB003, 1SB012, 1SB015, 1SB021, 1SB024 and 1SB027 opened and reclosed as expected. No Power Operated Relief Valves lifted during this event as pressures remained acceptable. Pressure stabilized at approximately 1070 psig for S/Gs A, B, C, and D. The Turbine Driven (CAPT) and 1B Motor Driven Auxiliary Feedwater Pumps started as expected and were available to supply feedwater to the S/G's. Steam from the CAPT vented to the atmosphere and it was verified that no unexpected releases had been made. Primary cooldown rates for the Reactor Coolant System did not exceed 100 degrees F/hour. 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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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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May 19, 1986

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

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

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/86-22 concerning a Reactor trip due to a defective procedure. 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 / [illegible]

Hal B. Tucker

RWO:slb

Attachment

xc: Dr. J. Nelson Grace, Regional Administrator
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NRC Resident Inspector
Catawba Nuclear Station

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