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 3

TITLE (4)

Reactor Trip Due to Steam Generator Low-Low Level

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 5	8 5	8 5	— 0 2 5 —	0 0	0 5	1 5	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 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, Assistant 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 April 15, 1985 at approximately 0522 hours, the Unit 1 Reactor tripped due to a low-low level signal from Steam Generator (S/G) A. The low-low level was caused by insufficient main feedwater flow to S/G A. Unit 1 was in Mode 1 at 90% reactor power at this time.

During Power Escalation Testing, the Main Steam flow transmitters were not calibrated as required by Periodic Test PT/1/B/4150/16, Unit Load Steady State. This resulted in Main Steam flow being approximately 10% higher than Control Room indication. Prior to the Reactor trip, the Operator had been experiencing vibration problems with Main Feedwater Pump Turbine A. In an effort to maintain S/G levels, the Nuclear Control Operator placed first Pump B and then feedwater regulating valves 1CF27 and 1CF28 in MANUAL. The Operator attempted to maintain a decreasing S/G A level by increasing feedwater flow to a value approximately 4% greater than Main Steam flow from S/G A. Since the Operator was not aware of the error in Control Room indication of Main Steam flow, feedwater flow to S/G A was actually about 7% lower than Main Steam flow. This caused S/G A level to decrease to the low-low level setpoint of 50% which resulted in a reactor trip. This incident is classified as an Administrative Error, since adequate administrative controls did not exist to ensure that the transmitters were calibrated after testing was completed at each power plateau, or that the appropriate Operations Personnel were made aware of the specific amount of mismatch.

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The Main Feedwater (CF) System supplies feedwater to the Steam Generators (S/G's) at the temperature, pressure and flow required to maintain proper S/G levels. Feedwater flow to the S/G's A, B, C and D is regulated by valves 1CF28, 1CF37, 1CF46 and 1CF59 which can be manually or automatically controlled. In AUTO, CF flow, Main Steam (SM) flow, and S/G water level are used as controlling parameters for the Regulating Valves. Comparisons between CF and SM flow, and CF flow and S/G level are used to throttle the valves as required to provide adequate feedwater flow to the S/G's. In MANUAL, the valves are throttled from the Control Room.

Main Steam flow from each S/G is monitored by two transmitters which provide signals to the Process Control and Protection Cabinets. From these cabinets, signals are sent to the Feedwater Control System, the computer and the Control Room SM Flow Indicators to provide input for CF Regulator Valve control.

PT/1/B/4150/16, Unit Load Steady State, is performed after Initial Fuel Load and after each refueling at various Reactor Power Plateaus, by the Performance Group. One purpose of this procedure is to perform a cross-check verification of CF and SM Flow in each S/G. This is accomplished by measuring S/G CF Flow, SM Flow, S/G Blowdown (BB) Flow, and Auxiliary Feedwater (CA) Flow so that a SM-CF mismatch can be calculated. If the calculated value is >2%, I&E is notified so that SM Flow indication can be calibrated as necessary.

From January 19, 1985 to April 12, 1985, PT/1/B/4150/16 was performed at various Reactor Power Plateaus to determine the Feedwater-Main Steam flow mismatch. Per the Procedure, Performance notified I&E of the test results so that SM Flow Indication could be calibrated. However, an I&E Staff Engineer and the Performance Reactor Group agreed to wait, until PT/1/B/4150/16 had been completed for 100% Full Power, to calibrate the SM Flow Indicators. Therefore, even though Feedwater-Main Steam mismatches greater than 2% were found during testing, the SM Flow Transmitters remained out of calibration during Power Escalation Testing.

On April 15, 1985 at 0353 hours, CF Pump Turbine A Low Pressure (L/P) Bearing Vibration Alarm was received in the Control Room, and would not clear. In response to this, the Nuclear Control Operator (NCO) placed CF Pump B in MANUAL and increased the load on that pump. This resulted in a load decrease on CF Pump A. As this did not clear the Vibration Alarm, the Operator decreased Reactor Power from 94% to 90% Full Power at 0455 hours. At 0458 hours the Vibration Alarm cleared and the NCO returned CF Pump B to AUTO.

At 0515 hours, S/G A and B Level Deviation Alarms were received in the Control Room. S/G A and B Narrow Range Levels reached approximately 70% (the programmed S/G level is 64% at 90% Full Power). The NCO then placed Feedwater Regulating Valves 1CF28 and 1CF 37 in MANUAL and then throttled these valves to decrease CF flow. When S/G A and B levels reached 66%, the NCO placed these valves back in AUTO.

After these valves were returned to AUTO, S/G B level stabilized. However, S/G A level continued decreasing. The NCO, assuming valve 1CF28 was not regulating flow properly, returned it to MANUAL at 0516 hours. The valve was

throttled until CF flow was 4% greater than SM flow, as indicated by Control Room instrumentation. However, since the mismatch in Feedwater-Main Steam flow for S/G A was greater than 11% at 90% Full Power, the NCO unknowingly was limiting Feedwater flow approximately 7% below Main Steam flow. Therefore, S/G A level continued to decrease and the NCO fearing that opening of 1CF28 any further would cause enough shrink in S/G A to reach the Low-Low Level Setpoint of 50%, did not act to increase Feedwater flow.

At approximately 0523 hours, the Unit 1 Reactor tripped due to S/G A low-low level. This resulted in a Turbine Trip and CF isolation. The low-low S/G level initiated an auto-start of both Motor-Driven CA Pumps. However, before S/G levels could be restored by these pumps, S/G's B, C and D Low-Low Level Reactor Trips occurred. This initiated an auto-start of the Turbine Driven CA Pump. The Condenser Dump Valves also opened to allow SM discharge to the condenser.

The increase in S/G A and B levels was possibly initiated when CF Pump B load was increased and Reactor Power decreased. After levels were reduced by the NCO and the CF Regulating Valves returned to AUTO, the NCO thought that the continued decrease in S/G A level was a result of the inability of 1CF28 to maintain S/G A level. However, after the trip occurred, I&E checked the calibration and stroke time of the CF Regulating Valves and determined that the response of 1CF28 was satisfactory.

This incident is classified as an Administrative Error, since adequate administrative controls did not exist to ensure that the transmitters were calibrated after testing was completed at each Power Plateau, or that the appropriate personnel in Operations were notified of the specific amount of mismatch in Feedwater-Main Steam flow.

CORRECTIVE ACTION

1) The calibration and stroke time for Regulating Valves 1CF28, 1CF37, 1CF46 and 1CF59 were checked. The Subsequent Corrective Action identified that 1CF28 was able to adequately maintain S/G A level. All SM Flow Transmitters have been calibrated.

2) A procedure change will be made to PT/1/B/4150/16 to require notification of Operations of SM-CF Flow mismatches. Also this change will be incorporated into the Unit 2 Unit Load Steady State Procedure.

SAFETY ANALYSIS

Following the Reactor Trip, Reactor Coolant (NC) System pressure remained above the setpoint for Automatic Safety Injection Actuation and below the setpoint for opening the Power Operated Relief Valves. NC System Temperature Cool Down Rates were within the Technical Specification limit of 100F/hr. Pressurizer level remained above 17% and below 60% and recovered to approximately 26%. S/G levels were re-established to approximately 38% Narrow Range within approximately 30 minutes. All CA Pumps started as designed to restore S/G levels and to ensure adequate decay heat removal. The Reactor tripped with no anomalies. The health and safety of the public were unaffected by this incident.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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May 15, 1985

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

Subject: Catawba Nuclear Station
Docket No. 50-413

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/85-25 concerning a Reactor Trip from 90% Reactor Power due to low-low level in Steam Generator A. 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

cc: Dr. J. Nelson Grace, Regional Administrator
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