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 2	0 5 0 0 0 4 1 4	1 OF 0 6

TITLE (4)
Reactor Trip Due To Feedwater Pump Suction Flow Pressure Switch Out Of Calibration

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)
1 1	2 0	8 6	8 6	— 0 5 1 —	0 1	0 2	1 1	8 7	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 5

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
X	S J	P S	X 9 9 9	NO					

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 November 20, 1986, at 1159:50:809 hours, the Reactor tripped on Main Turbine trip above 69% Reactor power. Due to a Feedwater (CF) Pump suction flow pressure switch being out of calibration, the CF Pump B Recirculation Control Valve opened automatically. The resulting flow/pressure transient caused the trip of all Condensate Booster Pumps. Both CF Pumps tripped on low suction pressure resulting in the Turbine trip and the subsequent Reactor trip. The unit was at 98% power at the start of the transient and the Reactor power was run back to 85% prior to the Reactor trip.

This incident is assigned Cause Code X, Other. The CF Pump suction flow pressure switch was found to be out of calibration, resulting in a flow/pressure transient when CF Pump B Recirculation Control valve automatically opened.

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

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BACKGROUND

The Condensate (CM) System (EIIS:SD) starts at the Condenser Hotwell, where flow from an equalization header splits to supply each of three Hotwell Pumps (EIIS:P). Two Hotwell Pumps are capable of handling the full range of normal operating conditions, with the third pump available for redundancy or use during abnormal occurrences. Each Hotwell Pump has a suction strainer (EIIS:STR) to protect the internals from debris. The third Hotwell Pump will automatically start on low CM Booster Pump suction pressure.

After passing through various system equipment, Condensate flow splits to supply each of three CM Booster Pumps. Under normal operating conditions two CM Booster Pumps will be in service with the other in standby. The Booster Pumps are provided to supply the total suction head requirements of the Main Feedwater (CF) (EIIS:SJ) Pumps. The standby pump will automatically start on low CF Pump suction pressure.

The Load Rejection Valve (EIIS:FCV) provides a bypass from the Hotwell Pumps discharge directly to the suction of the CM Booster Pumps. The control switch for the valve has two positions, AUTO and CLOSED. In the AUTO position, the valve will open and throttle to maintain adequate CM Booster Pump suction pressure. In the CLOSED position, the valve will be closed except during a generator load rejection or on loss of one CF Pump when above 56% load. The valve will automatically open and throttle to maintain adequate CM Booster Pump suction pressure.

The CF Pump Recirculation Control Valves provide a minimum flow path from the CF Pump discharge to the Main Condenser (EIIS:COND). The valves will open when their respective CF Pump is reset and a low CF Pump suction flow signal or a Reactor (EIIS:RCT) Trip signal is generated.

Procedure PT/2/B/4250/04B, CFPT Stop Valve Movement Test, is used to perform a daily test to verify the freedom of movement of the Low and High Pressure Stop Valves on the CF Pump Turbines during normal operation.

DESCRIPTION OF INCIDENT

On May 14, 1986, a Nuclear Station Modification (NSM) was completed to modify the CF Pump Recirculation Valves, 2CF-6 and 2CF-13, to fail open on low CF Pump suction flow and/or Reactor Trip signals.

On November 19, 1986, PT/2/B/4250/04B was performed. CFPT 2B High Pressure Stop Valve was cycled at 0841 hours. The CF Pump B suction flow pressure switch (2CMPS5944) indicated low due to being out of calibration at some point in time after this event.

On November 20, 1986, the unit was at 98% power. CM Booster Pumps B and C were in service with CM Booster Pump A in standby. Hotwell Pumps B and C were in service. Hotwell Pump A had been removed from service and red tagged at 0900 hours, for cleaning of the suction strainer.

Personnel were in the progress of conducting procedure PT/2/B/4250/04B, CFPT Stop Valve Movement Test. At 1154:22 hours, a Nuclear Equipment Operator cycled the CF Pump Turbine 2B High Pressure Stop Valve per the PT. 2CF-13, CF Pump B Recirculation Control Valve, unexpectedly opened. At 1154:27:250 hours, the CM Booster Pump Suction Header Pressure Lo alarm was received. At 1156:30 hours, the Nuclear Control Operator began Turbine (EIIS:TUR) Power/Reactor Power reduction due to decreasing Steam Generators (S/Gs) levels. At 1156:56 hours, 2CM-83, CM Load Rejection Valve, automatically opened. The CF Pump Suction Header Pressure Lo alarm (EIIS:PA) was received at 1159:10:140 hours. CM Booster Pump A started automatically at 1150:11 hours on low CF Pump suction pressure. The CF Pump Suction Header Pressure Lo alarm returned to normal at 1159:11:633 hours. At 1159:41:731 hours, the CM Booster Pump Suction Header Pressure Emergency Lo alarm was received. At 1159:42 hours, all CM Booster Pumps tripped on low suction pressure. CF Pumps B and A tripped on low suction pressure at 1159:46:173 hours and 1159:50:337 hours, respectively. At 1159:50:625 hours, the Main Turbine tripped on loss of both CF Pumps. At 1159:50:809 hours, the Reactor tripped at 85% power on Main Turbine trip above 69% Reactor power. At 1159:51 hours, the Motor Driven Auxiliary Feedwater (CA) (EIIS:BA) Pumps started on loss of both CF Pumps and CF Isolation occurred on Reactor Trip with Low Reactor Coolant System (EIIS:AB) average temperature. At 1159:52 hours, valves 2SB-3, 2SB-6, 2SB-9, 2SB-12, 2SB-15, 2SB-18, 2SB-21, 2SB-24, and 2SB-27, Main Steam bypass to Condenser Control Valves, automatically opened. 2CM-83 automatically closed at 1159:54 hours. Between 1159:59:045 hours and 1200:04:579 hours, all four S/Gs Low Low Level alarms (EIIS:LA) were received. The Turbine Driven CA Pump started automatically on low level in 2 of 4 S/Gs at 1200:08 hours. 2SB-3, 2SB-6, 2SB-12, 2SB-15, 2SB-21, and 2SB-24 also closed automatically at that time. Between 1201:13 hours and 1201:18 hours, 2SB-27, 2SB-9, and 2SB-18 automatically closed.

At 1204:28 hours, the Nuclear Control Operator secured the Turbine Driven CA Pump. At 1209:22 hours, he started CM Booster Pump B. The Nuclear Control Operator reset the CF Isolation signal at 1245:29 hours. At 1322:04 hours, the Nuclear Control Operator started CF Pump B and aligned to feed the S/Gs. Between 1329:17:313 hours and 1330:06:197 hours, S/Gs C, D, and B Low Low Level alarms returned to normal. At 1330:10 hours, the Nuclear Control Operator secured the Motor Driven CA Pumps. At 1404:06:519 hours, S/G A Low Low Level alarm returned to normal.

On November 22, 1986, a Variation Notice (VN) was completed to correct relay contact status on the control circuits for the CF Pump Recirculation Control Valves. A CF Pump trip occurred during unit restart (see LER 414/86-52). At 0855 hours, the unit entered Mode 2, Startup, and entered Mode 1, Power Operation, at 1045 hours.

CONCLUSION

The flow/pressure transient that resulted in the trip of the CM Booster Pump and the subsequent Turbine and Reactor trips was initiated by the opening of 2CF-13, CF Pump B Recirculation Control Valve. A post trip investigation revealed that pressure switch (EIIS:PS) 2CMPS5944 was out of calibration. This switch provides

CF Pump B suction flow indication to the control logic of 2CF-13. 2CF-13 should have opened when 2CMPS5944 indicated low. This did not occur. Further investigation revealed that the NSM to modify the CF Pump Recirculation Valves logic circuitry, incorrectly reversed the contacts involved with the CF Pump Tripped/Reset logic. The logic circuitry identified CF Pump B as being tripped, when it was actually reset. This prevented 2CF-13 from opening on the low suction flow signal from 2CMPS5944.

Later, when the Nuclear Equipment Operator closed CF Pump B Turbine High Pressure Stop Valve while performing procedure PT/2/B/4250/04B, the logic circuitry sensed a Reset condition on CF Pump B and 2CF-13 opened. This resulted in the flow/pressure transient and the subsequent Reactor trip. Had the NSM been correct, 2CF-13 would have opened when 2CMPS5944 indicated low resulting in the flow/pressure transient. PT/2/B/4250/04B was last performed on November 19, 1986, at 0841 hours. Had 2CMPS5944 been indicating low at that time, the transient would have occurred. Therefore, this incident is assigned Cause Code X, Other, due to pressure switch 2CMPS5944 being out of calibration.

On September 1, 1986, a CF Pump trip occurred due to pressure switch 2CMPS5944 being out of calibration (see LER 414/86-38). This resulted in 2CF-13 not opening to prevent the CF Pump trip. 2CMPS5944 was subsequently calibrated.

Personnel initiated a Work Request to investigate and repair the cause of 2CMPS5944 indicating low. Investigation revealed the pressure switch to be out of calibration. 2CMPS5944 was recalibrated, functionally verified, and the work request completed on November 22, 1986.

Pressure switch 2CMPS5944 is a Barton 289-A. A review of NPRDS indicated 34 failures out of 156 reportable applications involving the Barton 289-A. Of the 34 failures, 26 were due to being out of calibration. This failure is not reportable to NPRDS.

A VN was initiated on November 21, 1986, to correct the contact status for the Tripped/Reset logic for the CF Pump Recirculation Control Valve control circuits. The VN was completed on November 22, 1986.

During this incident, Hotwell Pump A was red tagged for cleaning of the suction strainer. Had the pump been available, it would have automatically started on low CM Booster Pump suction pressure. This may have possibly prevented the trip of all CM Booster Pumps and the subsequent Reactor trip.

During review of the Transient Monitor for this incident, various CA flow abnormalities were detected. An investigation of S/G A CA Flow Transmitter (EIIS:FT), CAFT5090, revealed that the equalization valve was open. This valve must be closed when the transmitter is in service. A review of the Work Request Data Base did not indicate any associated work requests by which the valve may have been left open. The valve was possibly inadvertently opened during the Unit 2 Generator repair outage during September/October, 1986. Work requests were initiated and completed to calibrate the CA flow transmitters on S/Gs A, C, and D. CA flow indication to S/G B was slow in responding after the start of the CA Pumps. A Work Request was initiated to correct the possible slow response of 2CA-58, CA Pump A Discharge to S/G B Isolation. Investigation revealed the valve to be operating properly and the work request was completed on November 22, 1986.

The Event Recorder times were noted to be one hour ahead of the Alarm Typer. A work request was initiated and completed to correct the problem.

There have been no previous incidents of Reactor trip due to equipment being out of calibration.

CORRECTIVE ACTION

(1) Nuclear Control Operator began Turbine Power/Reactor Power reduction.

(2) Nuclear Control Operator secured the Turbine Driven CA Pump.

(3) Nuclear Control Operator started CM Booster Pump B.

(4) Nuclear Control Operator reset the CF Isolation signal.

(5) Nuclear Control Operator secured the Motor Driven CA Pumps.

(6) A VN was initiated and completed to correct contact status on CF Pump Recirculation Control Valves.

(7) A Work Request was initiated and completed on pressure switch 2CMPS5944.

SAFETY ANALYSIS

The Reactor tripped as designed upon the trip of the Main Turbine above the P-9 setpoint (69% power). Had the anticipatory Reactor Trip on Turbine Trip failed to function (no credit is taken for the anticipatory Reactor Trip on Turbine Trip in Chapter 15 of the FSAR), the Reactor would have been tripped by the Reactor Protection System (EIIS:JC) on Low-Low Steam Generator Level as a result of the loss of the Main Feedwater Pumps. The high pressurizer (EIIS:PZR) pressure, overtemperature Delta T, and high pressurizer water level trip functions provided additional redundancy to the low-low Steam Generator Level Trip function. Following the Reactor Trip, power quickly decreased to and was maintained at decay heat levels.

Following the trip of the Main Feedwater Pumps, the Motor Driven Auxiliary Feedwater Pumps started as designed. The Turbine Driven Auxiliary Feedwater Pump started on Low-Low level in 2-out-of-4 Steam Generators following the Reactor trip. The minimum Steam Generator level encountered during the transient occurred just after the trip and was 57.5% wide range, after compensating for the difference between the calibration temperature of the wide range level instrumentation and the

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

Steam Generator operating temperature (45.1% uncompensated). The top of the tube bundle is at 58.7% wide range. This is the first trip to occur at high power levels on Catawba Unit 2. The Westinghouse site representative feels that the S/G level response was as expected. Decay heat was removed by Auxiliary Feedwater with steam relieved to the Main Condenser. The Steam Generator Power Operated Relief Valves, atmospheric Dump Valves, or Main Steam Safety Valves were not required during this transient.

Pressurizer pressure remained below the setpoint of the Pressurizer Power Operated Relief Valves (EIIS:RV) and above the safety injection setpoint. Pressurizer level remained on scale and above the setpoint for pressurizer heater cutoff and letdown isolation. Reactor coolant average temperature stabilized at approximately 543 degrees F post trip. The Primary System cooldown did not exceed the 100 degrees F per hour Technical Specification limit.

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

DUKE POWER COMPANY
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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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February 11, 1987

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Revision 1 to Licensee Event Report 414/86-51 concerning a Reactor trip due to a Feedwater pump suction flow pressure switch being out of calibration. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker /mc

Hal B. Tucker

RWO/13/sbn

Attachment

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

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