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 9

TITLE (4)
Reactor Trip on Loss of Main Feedwater Pump Due to Design Deficiency

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

OPERATING MODE (9): 1

POWER LEVEL (10): 0 6 2

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)

[ ] 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 November 20, 1985, at 0031 hours, the Reactor tripped automatically on Low Low Steam Generator (S/G) C level after the loss of Main Feedwater (CF). One CF Pump was in operation at the time. While attempting to place the non-operating pump into service, an Operator manually opened the CF suction valve for the non-operating pump, satisfying the third requirement for the Windmill Protection Logic which tripped the Condensate Booster Pump and the Hotwell Pumps. The operating CF Pump subsequently tripped due to the loss of suction. This resulted in the Low Low S/G C Level. The unit was at 62% power at the time of this incident.

This incident is classified as a Design Deficiency. The CF System Header Pressure Indicator wiring was reversed on the instrumentation drawing. This falsely satisfied one of the three Windmill Protection Interlocks.

A Temporary Station Modification was written and implemented to rewire the CF Pump Header Pressure Switch and correct the Windmill Protection Logic.

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BACKGROUND

The Windmill Protection Logic for the Main Feedwater (CF) (EIIS:SJ) Pumps exists to eliminate the possibility of the suction pressure on the CF Pumps from driving the pumps due to low discharge header pressure. The system prevents overspeed and overheating of the pump shaft bearings. Three conditions must be met prior to initiation of the protection logic. First, the CF discharge header pressure must be less than 900 psig. Second, the bearing lube oil pressure must be inadequate. Third, the CF pump suction valve must be open. The CF Pump Windmill Protection is a Non-Safety Related component, therefore no pre-operational inspection is required for the wiring of the CF header pressure indicator for the Windmill Protection Logic.

Prior to this incident, work was being done on CF Pump 1A per a Temporary Station Modification (TSM). The bearing lube oil pressure was lowered by throttling valve 1CF11, (Thrust/Journal/Drive Bearing, Lube Oil Inlet), which in turn lowered the bearing temperature. The lube oil pressure gauge was used as a guide to lower the bearing lube oil pressure to maximum pressure of 9.5 psig on CF Pump 1A. The previous maximum pressure was 12 psig. Since an error of 1.5 psig existed in the pressure gauge, the maximum pressure that the lube oil could actually achieve on CF Pump 1A was 8.0 psig. The pressure is considered to be inadequate for pump operation when it falls below 5.5 psig. The alarm cannot be reset until the pressure reaches 8.2 psig.

An interlock exists between the lube oil pressure status and the CF pump suction valves. If the lube oil pressure drops below 5.5 psig and the Low Lube Oil Pressure alarm is actuated, the suction valves cannot be opened from the Control Room.

DESCRIPTION OF INCIDENT

On November 19, 1985, at 0306:46 hours, the CF Pump 1A Low Oil Pressure Trip annunciator was actuated due to the lube oil pump being shutdown. The interlock between CF Pump 1A Suction Valve 1CM137 and the Low Lube Oil Pressure Trip annunciator was activated through the Windmill Protection Logic.

On November 20, 1985, at approximately 2330 hours, procedure OP/1/A/6250/01, Condensate and Feedwater System, was begun in order to start CF Pump 1A. The Operator at the Controls (OATC) attempted to open valve 1CM137. When the valve would not respond, a Nuclear Equipment Operator (NEO) was dispatched to manually unseat the valve. After the NEO returned, the OATC once again attempted to open valve 1CM137. Again the Windmill Protection Logic

prevented the opening of the valve. The NEO was sent back to manually open the valve in order to start CF Pump 1A. The OATC did not realize at this point that the Low Lube Oil Pressure Trip annunciator was activated and that this was the reason 1CM137 could not be opened from the Control Room.

At approximately 0030 hours, the NEO opened valve 1CM137. This satisfied the third condition for the CF Pump Windmill Protection Logic. The condition of Low CF Pump discharge header pressure was met due to the wiring on the indicator being reversed when the component was installed per Design drawings.

At 0030:20 hours, the Windmill Protection Logic tripped Hotwell Pumps 1B and 1C automatically. The Condensate (CM) (EIIS:SD) Booster Pumps 1A and 1B tripped automatically. At 0030:22 hours, CM Booster Pumps 1A and 1B tripped automatically. At 0030:23 hours, Hotwell Pump 1A tripped automatically. This eliminated sufficient suction flow to CF Pump 1B. At 0030:27 hours, CF Pump 1B tripped automatically on Emergency Low Suction Pressure. The Main Turbine then tripped automatically due to the loss of all CF Pumps. Motor Driven Auxiliary Feedwater (CA)(EIIS:BA) Pump 1A started automatically due to loss of all CF Pumps. At 0030:30 hours, Motor Driven CA Pump 1B started automatically and Main Steam Bypass to Condenser Valve 1SB012 opened. At 0030:34 hours, Main Steam Bypass to Condenser Valve 1SB003 opened.

CF Pump 1B was attempted to be restored to service per procedure AP/1/A/5500/06, (Loss of S/G Feedwater). Reactor Coolant (NC) System (EIIS:AB) average temperature (Tave) increased to approximately 593 degrees F, thereby decreasing Reactor power from 61% to 53% full power due to the negative moderator temperature feedback. At 0030:38 hours, Pressurizer (PZR) Power Operated Relief Valve (PORV) 1NC34A lifted. At 0030:43 hours, Steam Generator (S/G) C PORV 1SV7 lifted. At 0030:45 hours, S/G D PORV 1SV1 lifted and S/G B ASME Code Safety Valve Number 1, 1SV14, lifted. At 0030:50 hours, S/G B PORV, 1SV13, lifted. At 0030:52 hours PZR PORV 1NC34A lifted a second time. At 0030:53 hours, S/G B ASME Code Safety Valve, 1SV15, lifted. At 0030:55 hours, PZR PORV 1NC34A reseated. At 0031:37 hours, the Reactor tripped on S/G C Low Low Level. The S/G C Low Low level signal cleared immediately after the reactor tripped. At 0031:45 hours, valve 1SA5, (S/G 1C Main Steam to CA Pump Turbine), opened automatically, aligning steam to the Turbine Driven CA Pump (CAPT). At 0031:49 hours, the PZR Backup Heaters automatically energized, and valves 1SB003, 1SB012, and 1SB021 closed. At 0031:50 hours, S/G C went into Low Low Level alarm. At 0031:51 hours, S/G A went into Low Low Level alarm. This caused valve 1SA002, (S/G B Main Steam to CAPT), to open automatically aligning steam with the CAPT. At this

point the OATC began manipulating the CA throttle valves. S/G's B and C levels increased to approximately 55%. This caused rapid cooling and shrinking of the Reactor Coolant System. At 0031:54 hours, S/G B ASME Code Safety Valve 1SV14 reseated and valves 1SB15 and 1SB24 closed. At 0031:57 hours, S/G B ASME Code Safety Valve 1SV15 reseated. At 0032:10 hours, S/G B PORV 1SV13 reseated and valve 1SB009 closed. At 0032:11 hours, S/G D PORV 1SV1 reseated and at 0032:14 hours, S/G C PORV 1SV7 reseated.

At 0032:17 hours, Feedwater Isolation occurred due to the Reactor trip and a low NC average temperature (Tave) (less than 564 degrees F) occurring simultaneously. The NC System was continuing to cooldown rapidly and the PZR level was dropping. The PZR Backup Heaters de-energized at 0037:51 hours, Letdown Isolation occurred due to the PZR level going below 17%. At 0044:43 hours, valve 1NV252A, (NV Pumps Suction from Refueling Water Storage Tank (FWST)), was opened in order to align the FWST with the Chemical and Volume Control (NV) (EIIS:CB) Pump and restore the PZR level. At 0045:45 hours, valve 1NV252A was closed. At 0055:12 hours, Letdown was restored.

On November 20, 1985, at 2047 hours, Reactor startup was begun and the unit entered Mode 2, Startup. At 2127 hours, the unit entered Mode 1, Power Operation.

TRANSIENT ANALYSIS

Prior to the reactor trip, the loss of the Condensate Booster Pumps and the Hotwell Pumps caused a sharp decrease in CF Pump 1B suction pressure, from 800 psig to the low suction pressure trip setpoint of 275 psig. Steam generator feedwater was restored within the required time by the auto-start of the motor-driven CA pumps. The CAPT auto-started with the reactor trip.

Steam generator levels decreased as feedwater flow was momentarily lost and continued to decrease after the start of the CA Pumps. Level reached the low-low level reactor trip setpoint (approximately 29%) in S/G 1C. The other levels were following very close. Post-trip feeding of the S/G's caused slight overfilling of B and C, reaching close to 55% creating an excessive cooldown rate.

Steam pressure control began following the Main Turbine trip (on loss of CF Pumps) with the actuation of the condenser dumps. Three condenser dump valves did not actuate, which presumably caused the S/G PORV's and two code safeties to open. The PORV's of B, C, and D steamlines actuated 40-50 psig above setpoint value. Banks 1 and 2 of B steamline code safeties actuated within their setpoint

tolerances. Steamline C experienced the highest pressure of the S/G's at a value of 1194 psig. S/G A PORV did not actuate even though steam pressure reached 1185 psig, 60 psig above the setpoint. Averaging S/G's, pressure increased approximately 135 psig before the reactor trip.

Upon loss of feedwater flow, (Tave) increased sharply from 578 F to 593 F just prior to the reactor trip. The increase in Tave caused a corresponding reduction in reactor power from 61% to 53% full power. Following the reactor trip, Tave rapidly dropped to about 563 F and then continued to decrease. This rapid cooldown was due to overfilling of the steam generators. Tave reached a minimum of 520 F and began increasing as CA flow was reduced by securing the turbine-driven pump.

Pressurizer level responded as expected by increasing during the Tave increase due to reactor coolant swell. Maximum level reached was 62.7% just prior to the reactor trip. Level then decreased rapidly to approximately 37%. The rate of decrease slowed down but continued and 17% was reached, resulting in letdown isolation and the trip of all pressurizer heaters. This occurred about six minutes after the trip. Pressurizer level continued to decrease to a minimum of 13.8%. The FWST was manually aligned to the charging pump suction in order to restore pressurizer level. Letdown flow was established when level reached 30%, 24 minutes after the trip.

Rising pressurizer level caused pressurizer pressure to increase to within the relief range of the pressurizer power-operated relief valves. One of the three reliefs opened at 2328 psig and again at 2316 psig. The openings occurred 13 seconds apart and were six seconds and three seconds in duration, respectively. After the second opening, pressure began to approach normal just prior to the trip. After the trip, pressure decreased steadily with decreasing level to approximately 2000 psig. Pressure continued to decrease to a minimum of 1899 psig. Pressure began to increase when the CAPT was secured 16 minutes after the trip. Pressure increased again when the pressurizer heaters were placed on. Letdown flow was established when pressure reached approximately 2050 psig.

All four reactor coolant pumps continued to run throughout the transient. Flowrate changes were due only to density changes in the reactor coolant caused by changing Tave.

Reactivity was controlled by the reactor trip. Residual heat was removed by CA to the condenser. Adequate core cooling was maintained at all times. Safety system availability was not

affected by this event. No emergency core cooling systems were required and Emergency power was not actuated.

CONCLUSION

This event is classified as a Design Deficiency. The CF Pump header pressure indicator wiring was reversed on the Design drawings and thus reversed when the instrumentation was installed. This caused the indicator to incorrectly provide a low header pressure signal and satisfy one of the three requirements for CF Pump Windmill Protection Logic.

A contributing cause is classified as Personnel Error. The Low Oil Pressure Trip alarm for CF Pump 1A bearing lube oil was actuated approximately 21 hours prior to the start of this event due to the Lube Oil Pump being shutdown. There is no indication that the alarm was cleared during the 21 hours prior to the event. Investigation shows that it would have been impossible to clear the alarm due to the inability of the oil pressure to reach a level of 8.2 psig even if the lube oil pump is running. Also, the interlock between the Low Lube Oil Pressure Trip and valve 1CM137 was activated and was prohibiting operation from the Control Room. Therefore, when the OATC tried to electrically open the valve after unseating it and it would not open, an investigation should have been conducted to find out why the valve would not open.

An additional contributing cause is classified as Cause Category X, Other. The pressure gauge used as a guide for setting the lube oil pressure had an error of 1.5 psig. When the value of the maximum pressure was lowered by throttling valve 1CF11, to 9.5 psig, the actual pressure was 8.0 psig. This prevented the Low Lube Oil Pressure trip alarm from resetting once pressure in the bearings was obtained.

The S/G B ASME Code Safety Relief Valves coordinated with the S/G B PORV in order to relieve pressure from S/G B. Four factors attributed to this. First, three of the Main Steam Bypass to Condenser Valves did not open limiting steam flow to the condenser. Second, the Atmospheric Dump Valves do not arm until 65% power. Therefore, during this incident they were not armed and did not release steam pressure. A Station Problem Report (SPR) was submitted to modify the loss of load interlock from 50% of full load to 29% of full load (see LER 413/85-43). This SPR has not yet been implemented. Third, the Rod Control System (EIIS:AA) was in the manual position due to the unit being in the process of increasing power. Since the control was in manual, the unit did not go into an automatic power runback in order to lower the reactor power. Therefore, steam continued to be generated at 53% power. Fourth, the S/G PORV's vary in setting from the Control Room (see LER 413/85-45) due to the nature of the equipment.

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

Procedure OP/1/A/6250/06, (Main Steam) directs S/G PORV's manual loaders to be set at 1125 psig. All of the valves responded within required tolerance of their setpoints.

During the transient the CF Pump suction pressure dropped to an approximate level of 300 psig and then increased to a level of 1020 psig immediately following the Main Feedwater trip. The pressure at the 1020 psig level was maintained for approximately 30 seconds. The pressure then decreased and leveled out at approximately 200 psig.

The opening of valve 1SA05 supplying steam to the CAPT generated a false start of the pump. LER 413/85-17 identified this problem.

Valve 1SP33, CFPT B High Pressure Drains, did not respond during this event. A Work Request was written on September 30, 1985 which requested an investigation and repair of the valve due to the valve failing to respond.

Also during this incident, the setpoint temperature of the Overtemperature Delta T Trips did not fluctuate during the transient. The setpoint is calculated from values derived from several unit conditions. The setpoint should have fluctuated, along with plant conditions during the transient. A Work Request was written and circuitry cards were replaced to correct the problem.

Design deficiencies which have resulted in incorrect control wiring do not appear to be a recurring problem.

CORRECTIVE ACTION

Subsequent

(1) Procedure OP/1/A/6100/05, (Unit Fast Recovery) was initiated.

(2) A Work Request was completed to investigate and repair the CF Pump 1A Lube Oil Switch malfunction. The maximum lube oil pressure was raised to allow resetting of the Low Lube Oil Pressure Trip annunciator.

(3) A TSM was completed to correct the wiring on the CF Pump header pressure indicator.

(4) A Work Request was completed to investigate and repair the Overtemperature Delta T Trip setpoint for NC D loop.

(5) A Work request was completed to recalibrate the CF Pump 1A Lube Oil Pressure gauges.

Planned

(1) Work Requests will be completed to investigate and repair the Main Steam Bypass to condenser valves 1SB018, 1SB027, and 1SB006, respectively, failure to open during this transient.

(2) An SPR is to be written to correct the wiring deficiency on the CF Pump Header Pressure Windmill Protection Logic current switch.

(3) The wiring of the CF Pump Header Pressure Windmill Protection Logic switch on Unit 2 is to be investigated and corrected if necessary.

(4) An Exempt Change was written to change Atmospheric Dump Value arming from 65% load to less than or equal to 50% load.

(5) The Necessary action on the pressure changes experienced during the transient on the suction side of the CF Pumps will be identified.

(6) An Update will be provided to all Operators reminding them of the need to review the annunciators prior to the startup of unit equipment.

(7) A Work Request will be initiated to investigate and repair valve 1SP33.

SAFETY ANALYSIS

Following the Reactor Trip due to Low Low S/G C level, power immediately decreased to zero. PZR pressure increased to 2328 psig and PORV 1NC23A lifted twice in order to relieve PZR pressure. The PZR level decreased to 17% which de-energized the PZR heaters and caused Letdown Isolation. The FWST and the Centrifugal Charging Pump were aligned to provide makeup to the NC System. PZR level was restored and stabilized at approximately 22%. Tave increased to 592 F, and then dropped below 530 F before stabilizing at 563 F. The rapid decrease in temperature was due to the overfeeding of S/G B and C. This resulted in Feedwater Isolation. Main Steam Bypass to Condenser Valves 1SB003, 1SB009, 1SB012, 1SB015, 1SB021, and 1SB024 opened. S/G B PORV 1SB13, S/G C PORV 1SV7 and S/G D 1SV1 lifted during this incident. In addition,

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

S/G B ASME Code Safety Valves 1SV14 and 1SV15 lifted. Since the pressure for S/G A PORV did not reach the setpoint tolerance, S/G A PORV may not have been required to lift. Pressure stabilized at approximately 850 psig for S/G A, B, C, and D. 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.

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

December 20, 1985

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 413/85-67 concerning a Reactor trip on loss of Main Feedwater pump due to a design deficiency. 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/mm

Hal B. Tucker

RWO:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

American Nuclear Insurers
c/o Dottie Sherman, ANI Library
The Exchange, Suite 245
270 Farmington Avenue
Farmington, CT 06032

M&M Nuclear Consultants
1221 Avenue of the Americas
New York, New York 10020

IE22
1/1

cc: INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

NRC Resident Inspector
Catawba Nuclear Station