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)
Degraded Auxiliary Feedwater Flow 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)
0 5	2 1	8 6	8 6	— 0 1 7 —	0 0	0 6	2 0	8 6	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 3

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)	[ ] 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)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 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)	
[ ] 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
B	B A	F C Y	F 1 3 0	NO					

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 29, 1986, at 1340 hours, personnel determined that the Auxiliary Feedwater (CA) Pump Turbine could not produce the required discharge flow. The investigation found a valve associated with the turbine had the pressure reduction cage assembly fouled with debris, preventing passage of required flow.

On May 21, 1986, at 1850 hours, it was determined that the Motor Driven CA Pump 2B could not produce the required discharge flow. The investigation found two valves associated with the pump had their pressure reduction cage assemblies also fouled with debris.

While investigating the flow problems, the system pre-operational test results were reviewed and it was determined that the flow from these pumps was degraded during that test in October, 1985, and has likely been degraded since that time. Unit 2 has moved through all modes of operation during the period of degraded CA flow.

This incident is assigned Cause Code B, Design, Manufacturing, Construction/ Installation Deficiency. The CA control valves' pressure reduction cage assemblies are sized such that the cages could foul due to the inability to pass piping corrosion products or small materials remaining in the system after flush activities which are larger than the cage opening.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(i)(B).

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

BACKGROUND

There are two separate systems which can supply feedwater to the Steam Generators. The Main Feedwater (CF) System (EIIS:SJ) is the normal supply and is in service for most modes of plant operation. The Main Feedwater System consists of two feedwater pumps which take suction from the discharge of three Condensate Booster Pumps and two C Feedwater Heater Drain Tank Pumps. The Auxiliary Feedwater (CA) System (EIIS:BA) acts as a backup supply of feedwater to the steam generators if the Main Feedwater Pumps become unavailable. There are two Motor Driven Auxiliary Feedwater Pumps and one turbine driven pump. These take their suction from either the Auxiliary Feedwater Condensate Storage Tank, the Upper Surge Tank for the Main Condensate System, or the Condenser Hotwell, with the Nuclear Service Water (RN) System (EIIS:BI) as the assured source of suction. The two Motor Driven Auxiliary Feedwater Pumps will start automatically upon the loss of both Main Feedwater Pumps.

The following valves are Fisher, Cavitrol-type, control valves which utilize a pressure reduction cage assembly to provide pressure drop across the valve:

2CA-36, CAPT Flow to S/G 2D
2CA-40, CA Pump 2B Flow to S/G 2D
2CA-44, CA Pump 2B Flow to S/G 2C
2CA-48, CAPT Flow to S/G 2C
2CA-52, CAPT Flow to S/G 2B
2CA-56, CA Pump 2A Flow to S/G 2B
2CA-60, CA Pump 2A Flow to S/G 2A
2CA-64, CAPT Flow to S/G 2A
2CA-180, CA Pump 2B Flow to Upper Surge Tank Dome Throttle
2CA-181, CAPT Discharge to Upper Surge Tank Dome Throttle
2CA-179, CA Pump 2A Discharge to UST Dome Throttle

The openings in these valves pressure reduction cage assemblies are 0.049 inches. RN is strained at the RN Pump Structure to 0.031 inches.

The Condenser Circulating Water (RC) System (EIIS:SG) may contain materials up to 0.125 inches in diameter since the Low Pressure Service Water (RL) System (EIIS:KG) is strained to that size. The RC supply to the CA System was added with the Standby Shutdown Facility Design in approximately 1979.

The RN System provides the assured suction source for the CA System, by automatic swapover logic. Should RN head pressure not be detected on the CA suction header, CA suction will automatically realign to the RC System. However, this swapover will not occur as long as RN head pressure is detected.

Procedure PT/2/A/4250/06B, CA Flow Verification, is required to be performed prior to entry into Mode 2 if the unit has been in Cold Shutdown for greater than 30 days.

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

DESCRIPTION OF INCIDENT

At 0305 hours, on April 23, 1986, the unit initially entered Mode 3. On April 29, 1986, an unacceptable flow rate to the Upper Surge Tank (UST) was obtained from the Auxiliary Feedwater Pump Turbine (CAPT) during a Performance Surveillance Test. Personnel issued a work request to investigate/repair valve 2CA-20. This valve was found to be fouled with pieces of Iron Magnetite and sand. Also, the valve's cascade piston would not respond due to rust.

Prior to completion of the repair to the valve, personnel attempted to verify proper CAPT flow to steam generator (S/G) 2B. Flow was unacceptable (less than 260 gpm) at normal S/G pressure. A work request was then initiated to investigate/repair valve 2CA-181. The cage assembly of 2CA-181 was found to be fouled with Iron Magnetite and sand. Repairs of 2CA-20 and 2CA-181 were completed on May 2, 1986, and CAPT flow to the UST was verified to be acceptable.

On May 15, 1985, Motor Driven CA Pump 2B failed to provide acceptable flow to the UST during an Operations Surveillance Test. A work request was issued to investigate/repair valve 2CA-180. Personnel then attempted to verify proper Motor Driven CA Pump 2B flow to S/Gs 2C and 2D. The flow rate obtained was unacceptable at normal S/G pressure. The cage assembly of 2CA-180 was found to be fouled with Iron Magnetite and sand. Repair to 2CA-180 was completed May 16, 1986, and Motor Driven CA Pump 2B flow to the UST was verified to be acceptable.

On May 21, 1986, at 1415 hours, Feedwater Isolation occurred due to a high-high S/G level. During the transient, personnel had difficulty maintaining CA flows. Following the transient, 3 post-trip CA flow graphs from Reactor Trips on May 17, 18, and 19, 1986, and the Auxiliary Feedwater Pre-operational Test results were reviewed. CA flow was found to be degraded to S/Gs 2C and 2D during the post-trip transients and the pre-operational test. Personnel then attempted to verify CA Train B flow to the S/Gs with flow control valves full open. The flow to S/G 2C was found to be 185 gpm, and flow to S/G 2D was found to be 210 gpm. Acceptable flow to each S/G was greater than/equal to 240 gpm. Personnel then issued work requests to investigate/repair 2CA-40 and 2CA-44. Repairs to both of these valves were completed on May 23, 1986, and Motor Driven CA Pump 2B flow to the S/Gs was verified to be acceptable (greater than 300 gpm to each S/G).

CONCLUSION

This incident is assigned Cause Code B, Design, Manufacturing, Construction/Installation Deficiency. The CA control valves' cage assemblies were not sized to assure that fouling of the cages would not occur. Although these materials have not been chemically analyzed, the material found appears to be Iron Magnetite, sand, and small diameter dirt or rock. The RN System is strained to 0.031 inches and the RL System is strained to 0.125 inches at the intakes.

The sand should pass through the cage assembly. However, larger diameter Iron Magnetite has been found to be the major contributor to the fouling. The differential pressure across the strainer cage is expected to force the Iron Magnetite through the cage so that fouling does not occur. This breakdown of the Iron Magnetite apparently has not consistently occurred.

The CA Flush Procedure was reviewed. The methods used were consistent with accepted practice. All of the Fisher Control Valves utilizing cage assemblies were removed from the system for flush and later reinstalled. Approximately 25 feet of 4 or 6 inch piping was not flushed, due to the inability to push flow through the CAPT. This section was also not flushed on Catawba Unit 1, or McGuire Units 1 and 2. McGuire CA Control Valves utilize cage assemblies which vary from 0.05 inches at the lowest flow position to 0.125 inches at the full flow position. McGuire has experienced no fouling problems with this size cage assembly.

The pre-operational test of Unit 1 CA System did not verify flow into the S/Gs. The Unit 2 CA System pre-operational test verified that CA flow was being supplied to the S/Gs. However, the specific values of CA flow were not included in the acceptance criteria for the test. The CA System Flow Verification Surveillance Test has previously been performed in Mode 4 (Hot Shutdown), with S/G pressure not at normal pressure (greater than 1000 psig). Performing these surveillances at normal S/G pressure would have indicated valve fouling problems earlier in the process. 3 Reactor trips occurred prior to discovery of the CA valve fouling problems. The degraded CA flow to S/Gs 2C and 2D was not recognized during the post-trip reviews of Reactor Trips on May 17, 18, and 19, 1986. Currently, specific CA flow is not required to be reviewed during the post-trip review. Also, a standard group of transient monitor plots was specified to be generated for the post-trip review. CA flow to S/Gs C and D was not included in this group. Personnel plan to review the post-trip format to include expected CA flow values, as well as other improvements.

On May 22, 1986, the CAPT failed to provide adequate flow to S/G 2B. Personnel adjusted the position of the CAPT throttle control valves to provide additional flow to the S/Gs. It is believed that this is indicative of further valve fouling.

During the Unit Blackout Test on May 27, 1986, CA flows to S/G's 2C and 2D were observed to be degraded (less than 300 gpm). On May 28, 1986, at 2232 hours, the unit entered Mode 4. On June 9, 1986, S/G 2D was drained to approximately 65% wide range. Motor Driven CA Pump 2B was placed in service and control valve fouling was verified. 2CA-40 was cleaned and reassembled on June 10, 1986. Also on that day, S/G 2C was drained to approximately 20% wide range. Motor Driven CA Pump 2B was placed in service and control valve fouling was verified. On June 11, 1986, 2CA-44 was cleaned and reassembled.

On June 11, 1986, at 2015 hours, a Unit 1 Reactor Trip occurred. During the transient, CA flow was observed to be slightly decreased from previous values, although still remaining acceptable. Personnel plan to monitor transient responses on both units for indications of CA flow degradation.

Personnel may utilize Motor Driven CA Pumps to provide normal feedwater during unit startup. This could potentially contribute to CA control valve fouling after the system is flow tested prior to entry into Mode 2.

Flow degradation due to control valve fouling has not occurred previously on Units 1 or 2.

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

CORRECTIVE ACTION

(1) Cage assemblies were cleaned or replaced for valves 2CA-181, 2CA-180, 2CA-40, and 2CA-44.

(2) S/G's 2C and 2D were drained and CA Pump 2B flow was observed. 2CA-40 and 2CA-44 were cleaned again.

(3) CAPT flow control valves have been throttled open so that flow remains acceptable.

(4) Procedures PT/1 & 2/A/4250/06B were revised so that CA flow will be verified at S/G pressure greater than 1000 psig, and results will be provided to the Unit Coordinator.

(5) The latest Unit 1 Reactor Trip Transient was reviewed and possible CA flow control valve fouling was identified.

(6) 2CA-20 was cleaned and returned to service.

(7) The frequency of PT/2/A/4250/06B has been increased so that proper CA Flow will be verified prior to each entry into Mode 2. After the fouling problems are corrected, the surveillance interval will be as required by Technical Specification 4.7.1.2.2.

(8) The need to utilize main feedwater in non-emergency situations rather than the Auxiliary Feedwater System has been emphasized to the operators.

(9) CA Control Valve cage assemblies will be changed to variable size openings of 0.05 inches at minimum flow to 0.125 inches at full flow.

SAFETY ANALYSIS

The Auxiliary Feedwater System is required to be operable in Modes 3, Hot Standby, 2, Startup, and 1, Power Operation. Unit 2 initially entered Mode 3 on April 23, 1986, during power escalation testing. Motor Driven CA Pump 2A was operable while the unit was in Modes 3, 2, or 1, except for 8.5 hours on May 9 and 10, 1986. At that time, CA Pump 2A was inoperable due to a failure of Diesel Generator (D/G) 2A. The pump would have operated with available off-site power. Motor Driven CA Pump 2A has consistently provided greater than 260 gpm at normal S/G pressure to S/G 2A and S/G 2B during all surveillance testing and unit transients to the present date.

The worst single failure that would affect CA System coincident with control valve fouling would be a failure of D/G 2A. In that event, Motor Driven CA Pump 2A would not respond if required. Auxiliary Feedwater flow would be provided by Motor Driven CA Pump 2B and the CAPT and flow from both would be degraded.

During the post-trip responses of May 17, 18, and 19, 1986, the minimum flow provided by Motor Driven CA Pump 2B was approximately 170 gpm to S/G 2C and 150 gpm to S/G 2D. The CAPT did not start during these Reactor trips. The minimum flow provided by the CAPT during the Unit Blackout Test on May 27, 1986, was 317 gpm to S/G 2B, and S/G 2C.

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On an autostart of the CA System with D/G 2A inoperable and CA control valves fouled to the extent previously observed, CA flow of 954 gpm would be provided to S/Gs 2B, 2C, and 2D. The CAPT could be manually realigned to equalize flows to feed S/Gs 2C and 2D. Based on previously obtained flowrates, S/Gs 2C and 2D would receive 954 gpm from Motor Driven CA Pump 2B and the CAPT. Therefore, sufficient auxiliary feedwater flow would have been available for cooldown from extended full power operation.

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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June 20, 1986

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 Licensee Event Report 414/86-17 concerning degraded Auxiliary Feedwater flow caused by blockage in the flow control valves 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

Hal B. Tucker

RWO:slb

Attachment

xc: Dr. J. Nelson Grace, Regional Administrator
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