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DUKE POWER

April 15, 1992

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

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/92-004

Gentlemen:

Attached is Licensee Event Report 414/92-004 concerning AUXILIARY FEEDWATER SYSTEM VALVE INOPERABILITY DUE TO POSSIBLE INAPPROPRIATE ACTION, AND/OR POSSIBLE EQUIPMENT MALFUNCTION.

The original discovery date (March 16, 1992) for this incident indicated a report date of April 15. The date of discovery was subsequently determined to be actually two days earlier (March 14, 1992) than first noted, indicating a report date of April 13. This Licensee Event Report is submitted two days late.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

W. R. McCollum
Station Manager, Jr.

/djp

Attachment

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): CATAWBA NUCLEAR STATION, UNIT 1

DOCKET NUMBER (2): 0 5 0 0 0 4 1 4

PAGE (3): 1 OF 0 8

TITLE (4): AUXILIARY FEEDWATER SYSTEM VALVE INOPERABILITY DUE TO POSSIBLE INAPPROPRIATE ACTION, AND/OR POSSIBLE EQUIPMENT MALFUNCTION

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 3	1 4	9 2	9 2	— 0 0 4 —	0 0	0 4	1 5	9 2	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)	[ ] 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)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 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: ROBERT FUTRELL, COMPLIANCE MANAGER

TELEPHONE NUMBER: AREA CODE 8 0 3 — 8 3 1 - 3 6 6 5

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	B A	X Q V	R 3 7 8	Y					

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 March 14, 1992, at 1400 hours, with Unit 1 in Mode 1, Power Operation, at 100% Power, Instrumentation and Electrical (IAE) personnel discovered that limit switch contacts (30 and 31) were in the incorrect position (open) on 1CA46B, Auxiliary Feedwater (CA) System Motor Driven Pump B Discharge to Steam Generator (S/G) 1C Isolation Valve. This problem was discovered during testing following modification of the CA System Flow Optimization and Runout Protection circuits. The incorrect position of the contacts defeated the 1CA46B automatic closing capability to secure flow to S/G C in the event of a feedwater line break, which could have placed the unit in a condition that was outside the design basis of the plant. The time in which the problem actually occurred was undetermined. This incident is attributed to Possible Inappropriate Action and/or Possible Equipment Malfunction. Corrective actions included declaring 1CA46B inoperable, performing corrective maintenance, successful testing of 1CA46B, performing a review of valve setup procedure for adequacy, verifying the proper set-up of other similar CA System valves, and returning 1CA46B to service. Planned corrective actions include procedure enhancements, and the development of test procedure(s) to ensure CA System valve(s) operation, and a review of system functions.

BACKGROUND

The Auxiliary Feedwater [EIIS:BA] (CA) System provides an assured source of emergency feedwater to the steam generators [EIIS:HX] (S/G) during plant conditions where the Main Feedwater [EIIS:SJ] (CF) System is not available. Loss of CF is defined as rapid reductions in S/G water levels, a turbine [EIIS:TRB] trip, and CA actuation by the protection system logic. During these conditions the CA System is relied upon to remove energy from the primary coolant due to residual core energy. Following a reactor [EIIS:VSL] (Rx) trip from power operation, power quickly declines to decay heat levels. If CA is not immediately available and no operator action is taken to establish CA flow, secondary water levels would continue to decrease as pressure is released from the S/G through the S/G relief valves [EIIS:V].

The CA System for each unit includes two motor driven pumps [EIIS:P] (MDP A & B), powered by separate and redundant safety related power supplies, and a steam powered turbine driven pump (TDP). Normally, MDP A is aligned to S/Gs A and B, and MDP B is aligned to S/Gs C and D. The TDP is normally aligned to S/Gs B and C.

The CA MDPs auto-start signal is initiated from any one of the following conditions;

1) safety injection
2) loss of emergency bus power
3) low-low level in any S/G
4) both main feedwater pumps tripped
5) start signal from the Anticipated Transients Without Scram (ATWS) mitigation system actuation circuitry (AMSAC)

The CA TDP is automatically started on Loss Of Emergency Bus Power and Low-Low Level in any two S/Gs.

The preferred sources of water for the CA System are the CA Condensate Storage Tank, the upper surge tank, and the condenser hot well. The Nuclear Service Water [EIIS:BI] (RN) System is the assured source of water for the CA System.

A CF line rupture is of particular concern in that the unit not only loses feedwater flow to the S/Gs, but also results in reduced CA flow to the remaining intact S/Gs since the CA System would deliver more flow to the faulted loop because of lower backpressure. CA flow to the faulted loop may be partially ineffective in removing core heat because the water could escape through the break without being converted to steam. The CA System design automatically [illegible]s the flow to the faulted loop to ensure that sufficient flow is delivered to at least two intact S/Gs. The CA System shall provide a minimum flow of 490 GPM to two intact S/Gs following a feedwater line break. For certain combinations of a faulted S/G and one inoperable CA motor

[EIIS:MO] driven pump, the operable MDP and the TDP are required to operate at the same time to provide the minimum flow assumed in the design analysis.

The flow optimization circuit (FOC) for the CA System is used to close motor operated isolation valves on the supply lines to S/G B or C when a motor driven pump has been started manually or automatically, the turbine driven pump is operating, and the motor driven pump on the opposite train has failed to start or the discharge pressure is below 200 PSI. This ensures that the minimum CA flow will be established if the operating motor driven pump is aligned to the failed S/G. For example, if motor driven pump B and the turbine driven pump are running, and motor driven pump A is not running 60 seconds after the CA start signal, the CA MDP B Discharge Isolation Valve, CA-46B, will automatically close and isolate S/G C from MDP B.

Automatic Runout Protection (RP) is provided to automatically close the MDP supply lines to S/Gs B and C on a high pump discharge flow setpoint of 780 Gallons Per Minutes (GPM) increasing flow indicating excessive pump runout.

An Anti Hammer Circuit, located on the Rotork Actuators Add-On-Pack (AOP), is provided on 1CA46B to prevent valve hammering and is designed as follows;

1) a limit switch is wired in series with the maintained control signal(s) to remove the signal(s) at a selected point in valve travel and,

2) a starter seal-in contact is wired in parallel with the series control signal/limit switch circuit to keep the starter contractor energized until the torque switch contact opens the circuit to stop the valve motor.

EVENT DESCRIPTION

On February 26, 1990, valve 1CA46B, CA Pump 1B Discharge to S/G 1C Isolation, was replaced with a new valve and actuator, including the Add On Pack (AOP). The new valve had a larger actuator and a better body design to ensure that the valve is capable of closing during all design conditions.

On March 7, 1990, an Instrumentation and Electrical (IAE) Technician adjusted the AOP switches per IP/0/A/3820/09, Removal, Replacement and Field Set-Up of Rotork Actuators.

On March 19, 1990, 1CA46B was successfully functionally tested. However, the functional test did not include testing the automatic closing circuitry associated with the Flow Optimization circuitry and Runout Protection circuitry.

On March 2, 1992, Design Engineering (DE) discovered a design deficiency with the FOC for the CA System. The FOC had been found to be vulnerable to a single failure which was thought to affect the ability of the CA System to provide adequate heat removal during a feedwater line break event.

On March 14,1992, at 1256 hours, CA Pump 1B was declared inoperable during the performance of TT/1/A/9200/65, Retest For Nuclear Station Modification (NSM) CN-11287; CA System Flow Optimization and Runout Protection. CA Pump 1B was placed in the Technical Specification Action Item Log (TSAIL). 1CA58A, CA Pump 1A Discharge to S/G 1B, was successfully tested per TT/1/A/9200/65.

On March 14, 1992, at 1400 hours, two IAE Specialists discovered 1CA46B AOP contacts (30,31) in the incorrect position (open) with the valve open during the performance of TT/1/A/9200/65. Problems with a pressure transmitter led the Specialists to investigate this circuit. This condition defeated the automatic closing capability of 1CA46B but did not affect the manual control of the valve from the Control Room.

At 1440 hours, Work Request (WR) 13266IAE (92017111 01) was written to investigate and repair 1CA46B, and this WR was added to the TSAIL with the entry for the B CA Pump to track the WR and the inoperability of 1CA46B.

On March 15, 1992, at 0151 hours, WR 92017111 01 was completed, and at 0205 hours, CA Pump 1B was removed from the TSAIL due to completion of TT/1/A/9200/65 & WR 92017111 01 restoring the operability of 1CA46B.

On March 16, 1992, at approximately 0700 hours, a Nuclear Production Engineer (NPE) in Component Engineering (CE) issued WR's 92017157 01 and 92017172 01 to check the set-up of 2CA46B and 2CA58A, respectively.

On March 17, 1992, at 1030 hours, WR's 92017157 01 and 92017172 01 were completed, and no discrepancies were found.

CONCLUSION

This incident is attributed to root causes of Possible Inappropriate Action and/or Possible Equipment Malfunction and is reportable per Section (a)(2)(ii)(B) of 10CFR50.73. The unit could have operated under conditions outside the current design basis due to a failure of 1CA46B to perform as designed as part of the FOC. Specifically, the feedwater line break analysis assumes CA flow to the faulted S/G is isolated after 60 seconds in the feedwater line break event. Flow optimization circuitry is provided to automatically accomplish this function. Due to the incorrect position of contacts 30 and 31, 1CA46B would not have automatically isolated flow to S/G C in the event that S/G C was the faulted S/G. Operator action to identify and isolate the faulted S/G, as directed by existing emergency procedures, was not affected and would have occurred within 15 minutes. This analysis demonstrated acceptable results under these conditions.

The IAE Specialists involved in the corrective maintenance and the responsible Component Engineer were unable to conclusively determine if the AOP was improperly adjusted upon set-up in 1990, (Possible Inappropriate Action) or if the AOP subsequently malfunctioned (Possible Equipment Malfunction).

When the AOP was initially set-up on March 7, 1990, a retest of the automatic close function was not performed for the (FOC) and the Runout Protection (RP) circuit for 1CA46B. This retest was not required because the FOC and RP circuits were not affected by the valve, actuator, and AOP replacements. Implementation Procedure TN/1/A/1186/00/06A for NSM CN-11186, Rev. 0, Work Unit 06 Replacement of Valve 1CA46B, specified all of the test/retest requirements, of which all were successfully completed to verify proper valve operation.

The IAE Specialist that signed step 10.5.6 of IP/0/A/3820/09 during the initial set-up to adjust the AOP switches and verify proper operation did not remember the particular work, but stated that the proper operation was verified per the procedure.

Between March 7, 1990, and the incident discovery on March 14, 1992, there were no documented maintenance activities performed on 1CA46B that could have affected the status of the AOP contacts 30 and 31.

CE performs an Annual Review of Rotork Actuator Work Requests to summarize the mode of failure identified during corrective maintenance activities on safety related valves. The 1990 review indicates the irregular occurrence of 3 AOP equipment failures discovered during corrective maintenance activities. In 1991, zero AOP's required corrective maintenance.

When corrective maintenance was performed per WR 92017111 01, the AOP switch was adjusted to close the contacts (30 & 31) for proper operation, and the proper operation of the AOP and the valve was verified per performance test TT/1/A/9200/65. The exact cause of the malfunction was not determined during corrective maintenance.

The IAE Specialists that performed the corrective maintenance did not conduct a root cause failure analysis prior to corrective maintenance initiation in an attempt to determine the cause of the AOP malfunction. Therefore, evidence to determine the root cause failure was lost during the corrective maintenance.

IAE Management is the process of providing Root Cause Failure Analysis Training to IAE personnel performing corrective maintenance. This Root Cause

Failure Analysis Training will assist in future root cause failure determinations.

Procedure enhancements to IP/O/A/3820/09 will be completed to ensure proper AOP set-up/operation.

As a result of this incident, Systems Engineering (SE) will develop procedure(s) to test the proper operation of the Unit 1/2 CA System FOC/RP valves. On a broader basis, DE will review other safety systems that impact Chapter 15 analyses to assure that all appropriate system functions that require periodic testing are identified. SE will develop any additional testing procedures. DE will complete this review in parallel with the Catawba Design Basis Documents effort.

The possible failure of the AOP is not reportable per Nuclear Plant Reliability Data System (NPRDS) requirements. However, due to the possible failure of the AOP preventing the valve operator from functioning, the valve operator is reportable per NPRDS requirements.

A review of the Operating Experience Program Data Base indicated that there have been no previous events in the previous 24 months attributed to Possible Inappropriate Action, and/or Possible Equipment Malfunction involving the FOC and RP circuits. Therefore, this event is not considered to be a recurring problem per Nuclear Safety Assurance Guidelines.

This Licensee Event Report (LER) is submitted two days late due to the original discovery date (March 16, 1992) for this incident indicated a report date of April 15, 1992. The date of discovery was subsequently determined to be actually two days earlier (March 14, 1992) than first noted. The event date was never definitely determined. However, the event date is known to be between March 7, 1990, the date of the new AOP switch adjustments, and the discovery date of March 14, 1992.

CORRECTIVE ACTION

SUBSEQUENT

1. WR 9201/111 01 was written to repair 1CA46B.
2. 1CA46B was declared inoperable by TSAIL entry of WR 13266IAE (92017111 01).
3. WR 92017111 01 was completed.
4. TT/1/A/9200/65 was successfully completed.

7. WR 92017172 01 was written to check the set-up of 2CA58A.

8. WR's 92017157 01 and 92017172 01 were completed.

9. A review of IP/O/A/3820/09 was performed to ensure that the procedure was adequate.

PLANNED

1. Implement procedure enhancements to IP/O/A/3820/09 to ensure proper AOP set-up/operation.

2. Establish integrated tests to verify proper operation of the Unit 1/2 CA System FOC valves.

3. Complete a review of safety systems that impact the Chapter 15 analyses to assure that all appropriate system functions are included in a periodic test program.

SAFETY ANALYSIS

The flow optimization circuit automatically isolates a faulted S/G from the operating MDP and the TDP in an event where the opposite MDP is not functioning. This circuit ensures that a minimum required CA flow is delivered to two intact S/Gs from the operable MDP and the TDP. The FSAR Chapter 15 Feedwater line break accident analysis takes credit for heat removal by the two intact S/Gs only. Under this analysis, 490 GPM must be delivered to two intact S/Gs within 60 seconds. No credit is taken for CA flow to the faulted S/G.

Under the re-analysis (performed using NRC approved methodology), 190 GPM is required within 60 seconds and is adequate for heat removal for as long as 15 minutes. In this case, operator action is assumed per emergency procedure guidance to isolate the faulted S/G from the operating MDP. With the operator taking action within 15 minutes to isolate the faulted S/G, CA would provide 839 GPM to two intact S/Gs. This analysis demonstrated acceptable results.

In actuality, a feedwater line break event would result in decreased S/G water level due to blowdown through the break. With the CA System actuated, CA flow to the faulted S/G would remove heat in amounts that result in an overcooling event. The amount of heat thus removed in a feedwater line break is less than the amount that is removed during the most significant overcooling event which is a steam line break (the limiting overcooling event). With the CA System feeding the faulted S/G, the feedwater line break is an overcooling event. However, once this flow is terminated, the undercooling phase of the transient begins.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
CATAWBA NUCLEAR STATION, UNIT 1	0 5 0 0 0 4 1 4	9 2	— 0 0 4	— 0 0	0 8 OF 0 8

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

is a steam line break (the limiting overcooling event). With the CA System feeding the faulted S/G, the feedwater line break is an overcooling event. However, once this flow is terminated, the undercooling phase of the transient begins.

Thus, the significance of the loss of automatic close function discovered on 1CA46B is very low. The re-analysis ultimately concluded that the consequences of a feedwater line break without automatic flow optimization or runout protection response were acceptable and consistent with the current design basis analysis. The CA pumps would have received the appropriate signals to start and perform their intended safety function. Adequate decay heat removal would have been maintained under the postulated scenario. The health and safety of the public were not affected by this event.