

Duke Power Company
Catawba Nuclear Station
P.O. Box 256
Clover, S.C. 29710

(803) 831-3000



DUKE POWER

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Washington, D. C. 20555

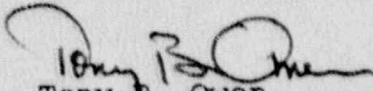
Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/90-23

Gentlemen:

Attached is Licensee Event Report 413/90-23 concerning ESF ACTUATION WHILE IN MODE 5 DUE TO DEFECTIVE PROCEDURE.

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

Very truly yours,


Tony B. Owen
Station Manager

keb\LER-NRC.TBO

cc: Mr. S. D. Ebnetter
Regional Administrator, Region II
U. S. Nuclear Regulator Commission
101 Marietta Street, NW, Suite 2900
Atlanta, GA 30323

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

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, GA 30339

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

Mr. K. Jabbour
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Mr. W. T. Orders
NRC Resident Inspector
Catawba Nuclear Station

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PDR ADOCK 05000413
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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Catawba Nuclear Station, Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 4 1 3	PAGE (3) 1 OF 0 5
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TITLE (4) ESF Actuation While In Mode 5 Due To Defective Procedure

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	3	0 9	0 9	0 2	3	0 0	0 4	2 6	N/A	0 5 0 0 0				
OPERATING MODE (9) 5			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)						73.71(b)					
POWER LEVEL (10) 0			20.402(b)			20.405(c)			X 50.73(a)(2)(iv)			73.71(c)		
			20.405(a)(1)(i)			50.36(e)(1)			50.73(a)(2)(v)					
			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)(ii)			50.73(a)(2)(viii)(A)					
			20.405(a)(1)(iv)			50.73(a)(2)(iii)			50.73(a)(2)(viii)(B)					
			20.405(a)(1)(v)			50.73(a)(2)(iii)			50.73(a)(2)(ix)					

LICENSEE CONTACT FOR THIS LER (12)

NAME R.M. Glover, Compliance Manager	TELEPHONE NUMBER AREA CODE 8 0 3 8 3 1 - 3 2 3 6
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COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC

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 30, 1990, at approximately 0744 hours, with Unit 1 in Mode 5, Cold Shutdown, an unexpected Reactor Trip occurred during the performance of PT/1/A/4350/02E, CA, CF, and Turbine Interlocks Periodic Test. A Turbine Trip on Reactor Trip followed, as well as a Feedwater Isolation signal due to Reactor Trip with Low Tave. The appropriate Feedwater (CF) and Auxiliary (CA) Feedwater valves closed as expected on the Feedwater Isolation signal. Affected CF and CA valves were realigned, and the procedure section was successfully performed following a procedure change. This incident is attributed to a defective procedure, because the Prerequisite System Conditions required that Reactor Trip Breakers 1RTA and 1RTB be Racked In-Closed. These breakers should have been Racked In-Open, to enable them to be cycled so that the desired Feedwater Isolation Signal could be reset. PT/1/A/4350/02E was revised to require that these breakers be Racked In-Open. The corresponding Unit 2 procedure, PT/2/A/4350/02E, will also be revised to ensure that these breakers are Racked In-Open.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/88

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

BACKGROUND

The Feedwater [EIIS:SJ] (CF) System provides water to the Steam Generators [EIIS:HX] (S/G) in conjunction with the Condensate [EIIS:SD] (CM) System. Each of the four CF lines contains a feedwater control valve [EIIS:V] (CF 28, 37, 46, 55), a feedwater control bypass valve (CF 30, 39, 48, 57), and a feedwater isolation valve (CF 33, 42, 51, 60). All of these valves close on a Feedwater Isolation Signal to terminate flow to the S/Gs following Hi-Hi S/G Level, Reactor Trip with Low Tave, Safety Injection, or manual actuation. Feedwater Isolation occurs on Hi-Hi S/G Level in 2/4 S/Gs on Unit 1 to ensure that CF will not overflow the S/Gs, and challenge the Main Steam [EIIS:RA] (SM) System piping [EIIS:PSP] which is not designed to take the weight of this water. Feedwater Isolation occurs on Reactor Trip with Low Tave to ensure that Reactor Coolant [EIIS:AB] (NC) System overcooling will not occur due to CF flow to the S/Gs. Feedwater Isolation on Reactor Trip with Low Tave will occur when the Reactor Trip Breakers [EIIS:BRK] are open, and 2/4 NC Loops Tave are less than 564 degrees F. In Mode 5, Cold Shutdown, average NC temperature is required to be less than or equal to 200 degrees F.

The Auxiliary Feedwater [EIIS:BA] (CA) System provides feedwater to the S/Gs in the event of a loss of normal feedwater supply, to remove primary coolant, and core residual energy. Two motor [EIIS:MO] driven and one steam turbine [EIIS:TRB] driven pumps are provided per Unit. The motor driven CA pumps autostart on loss of both CF pumps, 2/4 low-low level alarms in any one S/G, Safety Injection, or Blackout. The steam turbine driven pump will autostart on 2/4 low low level alarms in any two S/Gs or on Blackout. S/G CF Bypass to CA Nozzle Valves (CA 149, 150, 151 and 152) allow for the CF pumps to feed the S/Gs through the CA nozzles. S/G CA Nozzle Tempering Flow Isolation Valves (CA 185, 186, 187 and 188) are also provided. All eight of the above valves close on a Feedwater Isolation signal.

A Reactor Trip provided to protect against a sudden loss of heat sink is on 2 of 4 channels of narrow range S/G level being below the low-low level setpoint in one of 4 S/Gs.

Procedures PT/1,2/A/4350/02E, CA, CF and Turbine Interlocks Periodic Test are performed periodically to verify the proper response of various Feedwater and Auxiliary Feedwater components to signals such as Low-Low Steam Generator Level, Blackout, and Hi-Hi Steam Generator Level. Section 12.7 of PT/1/A/4350/02E, Feedwater Isolation Due to S/G Hi-Hi Level, verifies that Feedwater Isolation occurs in response to an initiated Hi-Hi Steam Generator Level Signal. In addition, Safety-Related Feedwater Isolation Valves are response time tested. The Prerequisite System Conditions in PT/1/A/4350/02E for Section 12.7 require that the CA Pumps 1A and 1B breakers [EIIS:BRK] are Racked Out-Open, and that leads be lifted to prevent autostart of the Turbine Driven CA Pump on 2/4 Low-Low Level in 2/4 S/Gs. The Prerequisite System Conditions also required that Reactor Trip Breakers 1RTA and 1RTB be Racked In-Closed. The appropriate CF and CA valves to be automatically closed by the Hi-Hi Steam Generator Level

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION

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

signal are aligned to the open position. In Section 12.7, Instrumentation and Electrical (IAE) personnel place Process Control Cabinet switches in test and place jumpers to ensure that no S/G Hi-Hi Level signals are already present. These actions generate Lo-Lo S/G Level Reactor Trip signals as channels are set-up for the test. A response time test (RTT) box switch [EIIS:XIS] is then installed, with the switch in the OFF position, across one channel, after which a Hi-Hi S/G Level test signal is injected into another channel. The RTT box is also connected to both the Turbine Trip and Main Feedwater Pump [EIIS:P] Trip relay [EIIS:RLY] (K621) in the Solid State Protection System (SSPS) Cabinet, and the response time test patch panel. With one channel already actuated, the desired Feedwater Isolation signal is generated by placing the response time test box switch to the opposite position, satisfying the 2/4 logic for Feedwater Isolation on Hi-Hi Steam Generator Level. After appropriate feedwater valve closure times, and relay K621 response time, are recorded and the RTT box switch is returned to the OFF position, the Feedwater Isolation signal is reset by closing, and reopening, Reactor Trip Breakers 1RTA and 1RTB. Both trains of Feedwater Isolation Logic, A and B, are tested in Section 12.7. After the test setup is removed from Process Control Cabinets by IAE personnel, Reactor Trip Breakers 1RTA and 1RTB are returned to their initial positions.

EVENT DESCRIPTION

On March 30, 1990, with Unit 1 in Mode 5, Performance personnel were performing PT/1/A/4350/02E, CA, CF, and Turbine Interlocks Periodic Test. Reactor Trip Breakers 1RTA and 1RTB were Racked In-Closed per Prerequisite System Conditions steps 8.16.12.1 and 8.16.12.2. Section 12.7, Feedwater Isolation Due to S/G Hi-Hi Level, was to be performed. Following the completion of prerequisites required for Section 12.7, IAE Technicians proceeded to perform step 12.7.5, which is composed of several substeps, to set-up the S/G Level channels in the Unit 1 Process Control Cabinet for the test. As the switches were placed in the test position per step 12.7.5, Low-Low S/G Level Reactor Trip signals were generated, as expected. However, a Reactor Trip on Low-Low S/G Level occurred, at approximately 0744 hours, opening Reactor Trip Breakers 1RTA and 1RTB. A Turbine Trip on Reactor Trip, as well as a Feedwater Isolation on Reactor Trip with Low Tave, followed. The appropriate CF and CA valves closed, as expected. None of the CA pumps autostarted, since they were isolated per the procedure. Since Unit 1 was in Mode 5, with rods fully inserted, no transients normally associated with a Reactor Trip at power occurred. The test was immediately terminated, and affected CF and CA valves were subsequently realigned. Steps 8.16.12.1 and 8.16.12.2 of PT/1/A/4350/02E were revised by a procedure change, on March 30, requiring Reactor Trip Breakers 1RTA and 1RTB to be Racked In-Open. Section 12.7 was subsequently re-performed, successfully verifying the proper Feedwater Isolation response to a S/G Hi-Hi Level signal.

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

CONCLUSION

This incident is attributed to a defective procedure for erroneous information, because Prerequisite System Condition steps 8.16.12.1 and 8.16.12.2 required that Reactor Trip Breakers 1RTA and 1RTB be Racked In-Closed, rather than Racked In-Open. This enabled the completion of Reactor Trip logic when the S/G Low-Low Level signals were generated in step 12.7.5.

When Section 12.7 was performed in the past, the Reactor Trip Breakers were Racked In-Open. The Prerequisite System Conditions had only specified that they be Racked In, and did not specify whether they should be closed or open. In March, 1990, PT/1/A/4350/02E was retyped. One of the enhancements to this retype was to specify Reactor Trip Breaker positions when racked in, however, the positions specified in steps 8.16.12.1 and 8.16.12.2 were closed rather than open. This Performance procedure retype received a cross-disciplinary review by Operations, and a review by Maintenance Engineering Services.

The subsequent corrective actions were to terminate the test, realign affected CA and CF valves, revise the procedure, and re-perform the test. The corresponding Unit 2 procedure will be appropriately revised to ensure that 2RTA and 2RTB are Racked In-Open prior to performing Section 12.7.

Two previous Licensee Event Reports have resulted from inadequate procedural precautions in the CA, CF, and Turbine Interlocks Periodic Test. LER 414/88-05 documents a case in which a CA autostart occurred during PT/2/A/4350/02E. Although the cause of the autostart could not be identified, the procedural precautions were deficient in that they did not verify that the CA autostart logic was reset prior to removing the CA autostart defeat. Corrective actions were directed toward revising the procedures to ensure that the autostart logic was reset. LER 414/89-013 describes an incident in which a Feedwater Isolation on S/G Hi-Hi Level occurred during the performance of PT/2/A/4350/02E. The same section, Section 12.7, was being performed in this previous incident. However, it was performed in Mode 4, Hot Shutdown, with heatup in progress, so that when the CF valves were opened, S/G levels rose too rapidly for Performance personnel to actuate the RTT box switch before Hi-Hi level setpoints were reached. Corrective actions were directed toward ensuring that PT/1,2/A/4350/02E system/equipment lineups would not affect S/G levels, and toward ensuring that outage schedules would be revised to reflect revisions to PT/1,2/A/4350/02E. The corrective actions for those two previous LERs should prevent future, similar recurrences. While both of the incidents involved deficient procedural precautions, they were due to not being prepared for plant conditions at the time the procedure sections were performed. The current incident differs from these previous events in that it was due to an error in procedure preparation as opposed to not considering plant conditions. Therefore, this is not considered to be a recurring event. There have been no LERs in the past 24 months resulting from errors in procedure retypes.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/86

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

CORRECTIVE ACTION

SUBSEQUENT

- 1) PT/1/A/4350/02E was terminated.
- 2) Control Room Operators realigned affected CA and CF valves.
- 3) PT/1/A/4350/02E was revised, and Section 12.7 was successfully performed.

PLANNED

- 1) PT/2/A/4350/02E will be revised to ensure that, prior to performing Section 12.7, Reactor Trip Breakers 2RTA and 2RTB are Racked In-Open.
- 2) This incident will be reviewed by all Exempt Performance Personnel.
- 3) Revisions to Performance procedures will be side barred or some mechanism will be used which easily flags what was changed. If not flagged, the entire procedure needs to be reviewed as a major rewrite.

SAFETY ANALYSIS

Feedwater Isolation response was as expected. Section 12.7 was performed in Mode 5, in which Technical Specifications require that means of decay heat removal be available. Although the CA pumps were disabled, Residual Heat Removal [EIIIS:BP] (ND) Pump 1B was operable and available to remove heat if necessary.

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