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DUKE POWER

February 28, 1992

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

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/91-013, Revision 1

Gentlemen:

Attached is Licensee Event Report 414/91-013, Revision 1, concerning REACTOR TRIP DUE TO NC PUMP TRIP CAUSED BY EQUIPMENT FAILURE.

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

/lhe

Attachment

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, GA 30323

R. E. Martin
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

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

M & M Nuclear Insurers
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New York, NY 10020

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

IE22

February 27, 1992

To: W. R. McCollum

Subject: LER 414/91-013, PIR 1-C91-0272, Revision 1; Reactor Trip Due To NC Pump Trip Caused By Equipment Failure

The following revisions have been made to LER 414/91-013.

1) <u>LER Form Page 4</u> (Conclusion)

   "SCRs will be tested during preventive maintenance relay testing." will be changed to "SCRs will be tested on a one time basis to ensure that leakage current is acceptable." The SCR manufacturer recommends one time only testing because further testing could damage the SCRs. This action also applies to corrective action established for LER 414/91-008.

2) <u>LER Form Page 4</u>

   Delete the following from Conclusion, paragraph 5: "This report will be revised, if needed, based on the outcome of this failure analysis. Part 21 reportability will be re-evaluated, if needed."

   Add the following to the Conclusion, paragraph 5: "Following the analysis of the failed SCRs, Motorola has determined that the defect is not generic to these devices."

   Add the following paragraph to Conclusion following paragraph 5: All spare SCRs (Motorola P/N SCR 1379H) were returned to ABB for testing and replacement. PD personnel will replace all questionable SCRs with SCRs manufactured after 1982.

3) <u>LER Form Page 5</u>

   Change Planned Corrective Action 1 to read: "All protective relay SCRs will be tested on a one time basis to ensure that leakage current is acceptable."

   Delete Planned Corrective Action 2. Motorola has determined that the defect is not generic.

STRose

S. T. Rose
Chairman, Safety Review Group

STR/lhe

cc: LER/91-013 File

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 WTH 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 3

PAGE (3): 1 OF 0 5

TITLE (4): Reactor Trip Due To Reactor Coolant Pump Trip Caused By Equipment Failure

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 6	2 0	9 1	9 1	0 1 3	0 1	0 2	2 7	9 2	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 7 1

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)	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)	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: R. C. 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
F	A B	B R K	G 1 8 2	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 June 20, 1991, at 0823 hours, with Unit 1 in Mode 1, Power Operation, at 71% power following a refueling outage, a Reactor trip occurred on the "Low Flow P8 Permissive Trip" due to the trip of Reactor Coolant (NC) Pump 1A. The NC Pump 1A safety breaker opened as a result of a defective Silicon Controlled Rectifier (SCR), which tripped the overcurrent relay for the breaker, thereby tripping the pump. Plant response was as expected, with the Main Turbine tripping on Reactor trip, feedwater isolation, and the Auxiliary Feedwater System starting and supplying water to the Steam Generators. Emergency procedures were entered, and appropriate notifications were made. The Unit was brought back on line by 0545 hours on June 21. This incident is attributed to an Equipment Failure; the SCR for the NC Pump 1A safety breaker 50/51 XYZ overcurrent relay was found to be defective, and was replaced. All other SCRs associated with the NC Pump supply and safety breakers have been checked. Inspections of SCRs in other critical applications have been performed.

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Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	9 1	— 0 1 3 —	0 1	0 2 OF 0 5

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

BACKGROUND

The Reactor Coolant [EIIS:AB] (NC) System utilizes four loops to transport heat from the Reactor to four Steam Generators [EIIS:HX] (S/Gs). An NC Pump [EIIS:P] in each loop is powered by 6900 V Switchgear [EIIS:SWGR] (located in the Turbine [EIIS:TRB] Building), via a 6900 V feeder breaker [EIIS:BRK] (1TA-3, 1TB-3, 1TC-3, and 1TD-3) in series with a safety breaker. Each safety breaker is located in its respective 6900 V NC Pump Switchgear, in either the 560 or 577 foot elevation Auxiliary Building [EIIS:NF] Unit 1 Electrical Penetration [EIIS:PEN] room. Each safety breaker is protected by a ground fault relay [EIIS:RLY] (50G) and an overcurrent relay (50/51). Actuation of either relay trips the safety breaker.

The overcurrent relay (ITE relay Type ITE51L) contains both an instantaneous and a time delay circuit, each of which utilizes a Silicon Controlled Rectifier (SCR). The SCR acts as a solid state type contact in the relay trip logic. The overcurrent relay was manufactured by ITE Imperial Corporation, which is now ABB Power T & D Company, and the SCR is manufactured by Motorola.

The P-8 permissive is in place when 2 out of 4 power range channels are greater than 48%. This permissive unblocks the 1 out of 4 loops loss of NC flow Reactor trip interlock.

EVENT DESCRIPTION

On June 20, 1991, Unit 1 was in Mode 1, Power Operation, at 71% power following a refueling outage. Power escalation was in progress. At 0823:11 hours, a Reactor trip occurred due to the "Low Flow P8 Permissive Trip", caused by the automatic trip of NC Pump 1A. Plant response was as expected. The Main Turbine tripped on Reactor trip. Operations personnel entered Emergency Procedures to respond to the trip. At 0823:23 hours, Feedwater Isolation occurred on Reactor trip with Low Tavg. (Low Tavg was due to low decay heat, as a result of a new core). Both motor [EIIS:MO] driven Auxiliary Feedwater [EIIS:BA] (CA) pumps [EIIS:P] automatically started, and supplied water to the Steam Generators. Neither the Pressurizer nor any of the Steam Generator Power Operated Relief Valves (PORVs) opened. None of the Pressurizer or Steam Generator code safety valves lifted. Banks 1 and 2 of the condenser steam dump valves actuated. A manual response required was the closing of valve SP34 (Steam Supply to the Feedwater Pump Turbines) to isolate steam drains which were causing excessive NC System cooldown. Letdown isolation occurred at 0828:40 hours as a result of Pressurizer level decreasing to approximately 17%, due to Low Tavg. (Low Tavg resulted from low decay heat, as a result of a new core.) Letdown was restored using the appropriate Abnormal Procedure. CA flows to Steam Generators C and D were slightly below the acceptance band, and Performance performed a subsequent flow balance. NC cooldown to 538 degrees F occurred, as a result of the lack of decay heat, loss of NC Pump 1A, CA flow to the Steam Generators, and the Auxiliary Steam header not being cross-tied to Unit 2. After CA was reset, and the Auxiliary Steam [EIIS:SA] (AS) header tied to Unit 2, Tavg began to increase. Other measures taken to increase Tavg included tripping Feedwater [EIIS:SJ] (CF) Pump 1B and isolating the atmospheric steam dumps. Appropriate notifications of the trip were made. Following the completion of all Emergency and Abnormal Procedures required, OP/1/A/6100/05, Unit Fast Recovery, was entered. CF flow to the Steam Generators was restored, and CA was shutdown and returned to standby. Operations personnel initiated work request 558160PS to investigate/repair the NC Pump trip.

At 1200 hours, on June 20, Power Delivery (PD) personnel identified the cause of the pump trip to be a degraded SCR associated with the NC Pump 1A safety breaker 50/51 XYZ overcurrent relay. The SCR in the time delay circuit of the overcurrent relay was degraded, resulting in the NC Pump 1A safety breaker tripping open. The degraded SCR was replaced. The SCRs for all other NC Pump overcurrent and ground fault relays, including both Units' supply and safety breakers, were also checked by PD personnel. A defective SCR in an NC Pump 1B 50G relay was found and replaced. The other relays, and associated SCRs, checked satisfactory.

By 0225 hours, on June 21, the Unit was returned to Mode 1. By 0545 hours, on June 21, the Unit was brought back on line.

SCRs for relays in critical applications were checked by PD personnel. These applications included Units 1 and 2 ETA and ETB (4160 Essential Switchgear) normal incoming breakers, Units 1 and 2 Chemical and Volume Control [EIIS:CB] (NV) Pump motors A and B breakers, and Units 1 and 2 hotwell/condensate booster pump motors A, B, and C breakers. These inspections were complete by 1810 hours, on June 21. They were performed under work requests 269-278TRL.

CONCLUSION

This incident is attributed to an equipment failure. The SCR associated with the time delay circuit of the NC Pump 1A safety breaker 50/51 XYZ overcurrent relay was defective. Corrective actions included replacement of the SCR, inspections of other SCRs associated with NC Pump supply and safety breakers, and inspections of SCRs in other critical applications.

On May 29, 1991, the Unit 2 Reactor tripped on low flow due to a degraded SCR (see LER 414/91-008). This SCR was associated with a ground fault relay for 6900 volt switchgear feeder breaker 2TB-3 (for NC Pump 2B). Breaker 2TB-3 opened, causing NC Pump 2B to trip. Therefore, although there have not been any other incidents at Catawba during the past two years before these two events due to degraded SCRs, this is considered to be a recurring problem. A planned corrective action in LER 414/91-008 was for SCR testing to be added to

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

existing preventive maintenance activities. SCRs wi[ll] be tested on a one time basis to ensure that leakage current is acceptable.

Following the May 29 incident, a planned corrective action was made for PD personnel to test a sample group (30-40) of switchgear protective relays. Following the June 20 incident, the scope of this testing was greatly increased. As of July 15, 1991, 7 of 153 inservice relays tested have been found to have degraded SCRs (including those causing the two Reactor trips). Also, 2 out of 100 in stock have been found to be defective. Approximately 700 relays containing SCRs will be tested, some of which can only be tested off-line. On-line testing of relays for defective SCRs, where possible, is in progress. SCR failures will be trended to determine if a pattern is present. All switchgear protection relay SCRs will be tested by the end of the next three refueling outages on each Unit (one-third are tested each outage).

Industry-wide, the vendor has indicated a failure rate of less than 0.1% over the past 10 years. At Catawba, 11 SCR failures have been identified, previous to the May 29, 1991 incident. Previous relay test methods led to detection of failed (i.e. open) SCRs. The method now being used to test SCRs can detect degradation less severe than total SCR failure, which could result in tripping a relay. This testing should enable detection of degraded SCRs prior to failure.

The SCR failure is reportable to the Nuclear Reliability Database System (NPRDS). Following the analysis of the failed SCRs, Motorola has determined that the defect is not generic to these devices.

All spare SCRs (Motorola P/N SCR 1379H) were returned to ABB for testing and replacement. PD personnel will replace all questionable SCRs with SCRs manufactured after 1982.

CORRECTIVE ACTIONS

SUBSEQUENT

1) Operations personnel entered Emergency Procedures to respond to the trip.

2) Operations personnel took appropriate actions to respond to the NC cooldown.

3) Operations personnel entered the appropriate Abnormal Procedure to respond to the letdown isolation.

4) PD personnel investigated and replaced the defective SCR under work request 55816O... Other NC Pump supply and safety breaker associated SCRs were checked.

NRC FORM 366A (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) TEXT CONTINUATION

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

5) PD personnel inspected SCRs in other critical applications. The scope of testing for defective SCRs was increased.

6) The defective SCRs were sent to Motorola for failure analysis.

PLANNED

1) All protective relay SCRs will be tested on a one time basis to ensure that leakage current is acceptable.

SAFETY ANALYSIS

Plant response to the Reactor trip from 71% power was as expected. The Main Turbine tripped on Reactor trip, and feedwater isolation occurred on Reactor trip with Low Tavg. The motor driven CA pumps automatically started and supplied water to the Steam Generators. Flows to Steam Generators C and D were slightly below the acceptance band, and Performance performed a subsequent flow balance. None of the Pressurizer or Steam Generator PORVs opened, and none of the Pressurizer or Steam Generator code safety valves lifted. Banks 1 and 2 of the condenser steam dump valves actuated. Manual responses included manual Reactor trip, throttling CA flow, and closing valve SP34 (to isolate steam drains causing NC cooldown). Letdown was reestablished using the appropriate Abnormal Procedure following the letdown isolation. Prior to the transient, NC temperature was at 577 degrees F. NC System temperature cooled down to 538 degrees F during the transient as a result of the lack of decay heat, loss of NC Pump 1A, CA flow to the Steam Generators, and the Auxiliary Steam header not being cross-tied to Unit 2. NC cooldown did not exceed 100 degrees F in one hour. Heat removal was provided via the Steam Generators being fed by CA flow.

If the NC Pump trip had occurred at 100% power, plant safety equipment was available to maintain critical parameters within their required values.

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