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DUKE POWER

May 24, 1995

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

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/95-004

Gentlemen:

Attached is Licensee Event Report 414/95-004 concerning REACTOR TRIP DUE TO COMPONENT FAILURE AND INADVERTENT FEEDWATER ISOLATION.

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

Very truly yours,

D. L. Rehn

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Attachments

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 2	05000 414	1 OF 6

TITLE (4)
Reactor Trip Due to Component Failure and Inadvertent Feedwater Isolation

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
04	27	95	95	-- 004 --	00	05	24	95	N/A	05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
D.P. Kimball, Safety Review Group Manager	(803)831-3743

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)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

ABSTRACT

On April 27, 1995, at 1156 hours, Unit 2 was in Mode 1, Power Operation, at 100% power when a component in the "A" Reactor Trip Breaker cubicle failed to change state during surveillance testing. As a result an invalid reactor trip indication to the Steam Generator (S/G) Level Control System occurred. The S/G Level Control System responded by reducing speed of both Main Feedwater (CF) Pumps to minimum. A reactor trip occurred when S/G levels decreased to the S/G Lo-Lo Level Reactor Trip Setpoint. The Auxiliary Feedwater (CA) System responded as expected to restore S/G levels. When average reactor coolant temperature (Tave) reached 553 degrees F, Engineered Safety Features Actuation System (ESFAS) Interlock, P-12, occurred, closing the steam dump valves. This event is attributed to component failure. Corrective actions include replacing the failed component. At 2345 hours, Unit 2 was in Mode 3, Hot Standby, with Tave at 557 degrees F, and S/G levels being maintained by CF. During checkouts to determine the failed component, a P-4 reactor trip signal was generated. CF Isolation occurred as a result of P-4 with Tave less than 564 degrees F. This event is attributed to work practices in that required procedures were not used. Corrective actions included appropriate management counseling of the involved individual.

BACKGROUND

The Reactor Trip Switchgear [EIIS:SWGR] consists of two Reactor Trip Breakers [EIIS:52], RTA (Train A) and RTB (Train B), and two Reactor Trip Bypass Breakers, BYA (Train A) and BYB (Train B). The Reactor Trip Breakers provide for an interruptable source of power to maintain the Control Rods [EIIS:ROD] at their withdrawn position during unit operation. When a reactor trip signal is received, the Trip Breakers open allowing the Control Rods to gravity insert into the core, shutting down the reactor [EIIS:RCT]. The Reactor Trip Bypass Breakers are used during testing of their respective, train related Reactor Trip Breakers. Each of the four breakers is installed in its own breaker cubicle (cell). A cubicle mounted mechanical switch, "33 cell switch", provides input to a circuit that provides reactor trip indication to the Digital Feedwater Control System (DFCS). This is accomplished by energizing an electro-mechanical "X" relay [EIIS:RLY] for each Trip Breaker that operates. When the logic for reactor trip indication is satisfied, the DFCS receives this indication and responds by running the Main Feedwater [EIIS:SJ] (CF) Pumps [EIIS:P] back to minimum speed. In addition to contacts for reactor trip indication for the DFCS, each "33 cell switch" has contacts that provide input to the P-4, Reactor Tripped, circuit.

P-4, Reactor Tripped, an Engineered Safety Features Actuation System [EIIS:JE](ESFAS) Interlock [EIIS:IGL], actuates Turbine [EIIS:TRB]trip, and closes CF valves [EIIS:V] on reactor coolant average temperature (Tave) less than 564 degrees F.

The Steam Generator [EIIS:SG](S/G) Level Control [EIIS:JB](IFE) System is designed to 1) maintain a programmed level in each of the four S/Gs during steady state and 2) restore S/G levels to programmed level during transients. If the system is unable to restore levels during an event such as a loss of feedwater flow, then a reactor trip will occur on S/G water level lo-lo. On a reactor trip signal generated by reactor trip breaker status, the IFE system runs the CF Pumps back to minimum speed.

The Auxiliary Feedwater [EIIS:BA](CA) System provides an assured source of emergency feedwater to the S/Gs during plant conditions when the CF system is not available. The CA system includes two motor [EIIS:MO] driven pumps and a steam powered turbine driven pump. Autostarts for the Motor Driven CA pumps include: 1) loss of both Main Feedwater Pumps and 2) Lo-Lo level in any one S/G. Autostarts for the Turbine Driven CA Pump include Lo-Lo level in any two S/Gs.

The Steam Dump [EIIS:JI](IDE) System is designed to provide for rejection of heat from the S/Gs to the main condenser [EIIS:COND]when the Main Turbine is not connected to the Main Steam [EIIS:SB] (SM) System. This is accomplished by the plant trip controller [EIIS:PMC],which controls the condenser dump valves' modulation to maintain Tave at 557 degrees F. A "blocking" circuit is provided to prevent excessive cooldowns below 553 degrees F thus ensuring Tave remains above the minimum temperature for criticality.

The P-12, Lo-Lo Tave Interlock, is part of ESFAS. The purpose of the interlock is to block steam dump valve actuation to prevent excessive cooldown below the minimum temperature for criticality. Its setpoint is 553 degrees F on any two out of four Reactor Coolant [EIIS:AB](NC) System loops. If the logic is satisfied, the steam dump valves are failed closed until NC system temperature is above 553 degrees F.

EVENT DESCRIPTION

April 27, 1995

1155:49 hours An invalid reactor trip indication was received by the S/G Water Level Control System. Both Main Feedwater Pumps (CFPs) ran back to minimum speed.

1156:44 hours The reactor tripped on "B" S/G Lo-Lo Level. The Main Turbine tripped on Reactor Trip. Auxiliary Feedwater (CA) System Autostart signals on Lo-Lo Levels in 2 out of 4 S/G started both Motor Driven CA Pumps and the Turbine Driven CA Pump.

1157:03 hours A CF Isolation occurred as expected on Reactor Trip with Tave less than 564 degrees F.

1200 hours Tave decreased to the Engineered Safety Features Actuation Setpoint (ESFAS) Interlock, P-12 setpoint of 553 degrees F, resulting in closing of the steam dump valves to the main condenser. Tave continued to decrease until S/G levels were restored to the minimum value for CA System reset and flow throttling. The lowest Tave observed was 535 degrees F.

Following the Reactor trip, the Failure Investigation Process (FIP) was initiated to determine the component that failed, causing the CF Pumps to run back to minimum speed. The FIP Team consisted of an Engineering Team Lead, a Management Team Lead, an Execution Team Lead and IAE technicians. The FIP Team identified two components. A plan was developed to checkout the X1A relay. The X1A relay, was checked and based on the satisfactory results, the "33 cell switch" was to be checked next.

2345 hours Unit 2 was in Mode 3, with Tave at 557 degrees F, and S/G levels being maintained by CF, when a CF Isolation occurred during checkout of the "33 cell switch" in reactor trip breaker RTA cubicle. A P-4 reactor trip indication was generated. With Tave less than 564 degrees F, this P-4 signal caused a CF Isolation. Operators entered AP/2/A/5500/06, LOSS OF S/G FEED, reset CF Isolation and re-established normal feedwater.

CONCLUSION

The reactor trip event occurred as a result of an invalid reactor trip indication from a failed component in the cubicle of reactor trip breaker RTA, causing the CF Pumps to runback to minimum speed. Since the CF Pumps were not tripped by the initiating signal, no Main Turbine trip occurred or was required. The operators' efforts to restore CF flow were unsuccessful because the invalid reactor trip indication rendered manual control of the CF Pumps inoperable and there was no indication of the reason for the CF Pumps' behavior. With the unit at Rated Thermal Power, the levels in the S/Gs rapidly decreased. The reactor trip on S/G Lo-Lo Level started both CA Motor Driven Pumps on Lo-Lo Level in B S/G. When a second S/G reached Lo-Lo Level, the Turbine Driven CA Pump started. By contrast, S/G level recovery took longer than it would have for a "Both CF Pumps Tripped" event which would have immediately tripped the Main Turbine and as well as the Reactor. The reactor coolant system cooldown to 535 degrees F following the trip was a direct result of full CA flow being supplied to the S/Gs until Narrow Range S/G level indication was restored and CA flow throttling was initiated.

The CF Isolation event occurred as a result of a Train A P-4, Reactor Tripped, signal being generated when the mechanical "33 cell switch" mounted in the RTA breaker cubicle was exercised. The Execution Team Lead on the FIP Team accompanied the IAE Technicians to execute the plan for checking the electro-mechanical X1A relay. The Execution Team, following a detailed and documented plan, completed the checks of the relay without incident. Since the checks of the relay did not indicate that the relay was failed, the next step in the process was to develop a detailed and documented plan for checking the "33 cell switch". The Execution Team Lead entered the breaker cubicle to visually inspect the switch and, without an approved plan, he acted inappropriately by exercising the switch. This resulted in generating a P-4 signal. P-4 in conjunction with Tave being below 564 degrees F resulted in CF Isolation. Operators reset the CF Isolation and restored normal feedwater to the S/Gs. This event is attributed to work practices/inappropriate action in that the FIP Execution Team Lead exercised the "33 cell switch" inside the cubicle for RTA, without the required procedure being used. Based on the results of the checks performed on the mechanical "33 cell switch", the switch was replaced. The decision to replace the switch was based on the fact that it exhibited some rotational "roughness" when manually exercised. Though the electro-mechanical X1A relay did not indicate failure, the relay was replaced.

The decision to replace the relay was based on the design feature that the relay relies on spring action to release the electrical contacts once the relay's electrical coil is de-energized. The two components removed from the RTA cubicle have been sent to the manufacturer for testing. A database search shows that these two subcomponents are not NRPDS reportable. The Catawba Safety Review Group (CSRG) will evaluate the results of the failure analysis and revise this Licensee Event Report (LER) if necessary.

A review of reportable events for the 24 months prior to the Unit 2 reactor trip of April 27, 1995 revealed four reactor trips have occurred at Catawba that were attributed to component failure. LER 413/93-006, Revision 1 describes a Unit 1 reactor trip as a result of a blown control power fuse [EIIS:FU] for Intermediate Range Nuclear Instrument [EIIS:IG] Channel N35. LER 413/94-001 describes a Unit 1 reactor trip as a result of loss of condenser vacuum when a two inch heater [EIIS:HTR] drain line broke. LER 414/94-006 describes a Unit 2 reactor trip as a result of closure of a Main Steam Isolation Valve due to failure of a Cutler-Hammer relay. LER 414/95-001 describes a Unit 2 reactor trip as a result of closure of a Main Steam Isolation Valve due to failure of an optical isolator. Since the failed component in this event is different than those described above, this event is not a recurring.

A review of reportable events for the 24 months prior to the Unit 2 CF Isolation of April 27, 1995 revealed one event that was attributed to document use practices by IAE. LER 414/94-007 describes a Unit 2 reactor trip that occurred as a result of the concurrent use of two procedures by two different IAE teams which placed the Reactor Protection System in a testing configuration not previously encountered by the technicians. The steps taken to place the test circuits in a normal configuration resulted in a reactor trip. For the event that occurred on April 27, 1995, no procedure was being used during the checkout of the "33 cell switch". Since the procedure use practice circumstances are different for these two events, this event is not recurring.

CORRECTIVE ACTIONS

REACTOR TRIP
IMMEDIATE

1. Operations entered EP/2/A/5000/E-0, REACTOR TRIP OR SAFETY INJECTION, and the transition was made to EP/2/A/5000/ES-0.1, REACTOR TRIP RESPONSE, to stabilize the unit in mode 3.

SUBSEQUENT

1. PT/0/A/4150/02, TRANSIENT INVESTIGATION, was completed and based on review, component failure was identified as the root cause of the trip.

2. A Failure Investigation Process Team was assembled to develop detailed and documented plans for failed component determination.

3. The X1A relay and "33 cell switch" were replaced and tested under W/O 95030158- 01.

4. The X1A relay and "33 cell switch" which were removed have been sent to the manufacturer for failure analysis.

PLANNED

1. Electrical/Systems Engineering will evaluate the manufacturer's results and make recommendations for further component evaluation.

2. The Catawba Safety Review Group (CSRG) will evaluate the results of the failure analysis and revise this Licensee Event Report (LER) if necessary.

FEEDWATER ISOLATION
IMMEDIATE

1. Operations entered AP/2/A/5500/06, LOSS OF S/G FEED, and re-established normal feedwater.

SUBSEQUENT

1. Management has counseled the involved individual.

SAFETY ANALYSIS

The reactor trip that occurred from Rated Thermal Power on loss of feedwater flow is bounded by the analysis for Turbine Trip as described in Chapter 15 of the FSAR. All systems responded as expected for this event.

Response to the CF Isolation that occurred in Mode 3, during component checkouts, was as expected for the initiating signal being a reactor trip with low Tave as described in Chapter 10 of the FSAR for CONDENSATE AND FEEDWATER SYSTEMS.

The health and safety of the public were not affected by these events.