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DUKE POWER

August 12, 1994

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

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/94-003

Gentlemen:

Attached is Licensee Event Report 414/94-003 concerning MANUAL REACTOR TRIP ON LOSS OF NORMAL FEEDWATER DUE TO SYSTEM PERFORMANCE.

This report is being submitted three days past the 30 day NRC requirement. Late submittal was discussed with Catawba Senior Resident, R.J. Freudenberger, who finds this date for submittal to be acceptable.

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

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

Mr. R. E. Martin
U. S. Nuclear Regulatory Commission
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Mr. R. J. Freudenberger
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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

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

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 2	05000414	1 OF 7

TITLE (4): Manual Reactor Trip on Loss of Normal Feedwater Due to System Performance

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
									N/A	05000
07	10	94	94	- 003 -	00	08	12	94	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)

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

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

On July 10, 1994, at 1347 hours with Unit 2 in Mode 1, Power Operation, at 100% power, a turbine runback occurred following manual trip of a malfunctioning Main Feedwater Pump (CFP). Following the runback, a loss of Main Feedwater (CF) occurred, caused by one of the Steam Generator's (S/G) levels reaching the "High-High" level ESFAS Interlock Setpoint. This resulted in ESFAS actions to trip the Main Turbine (MT), trip both CFPs, and isolate CF. Tripping of both CFP initiated an ESFAS signal to start both Motor Driven Auxiliary Feedwater Pumps. No automatic reactor trip signal was initiated because the unit was below the Reactor Trip on Turbine Trip Interlock Setpoint. The reactor was manually tripped at 1355 hours from 65% power. The unit was stabilized in Mode 3, Hot Standby, using "Reactor Trip Response" Emergency Operating Procedure. The Root Cause of this event is attributed to the performance of the CF System following level recovery from low S/G levels. The initial trip investigation did not reveal any anomalies in the response of the S/G Level Control System. Following restart, additional information indicated the need for continuing evaluation of the S/G Level Control System performance. The performance of the S/G Level Control System transient response is a unit availability/reliability concern and not a nuclear safety issue. Corrective actions include evaluation of secondary systems interactions and responsiveness of the S/G Level Control System including the CF Control Valves.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

BACKGROUND

The Main Turbine Control System [EIIS:JJ](ITE) is designed to control the turbine [EIIS:TRB] under all load conditions. Incorporated into this design is the capability to "runback" the turbine load during selected abnormal conditions for the secondary side of the unit. One such condition is the loss of a single Main Feedwater Pump [EIIS:P] with turbine load greater than 65%. The "runback" terminates at less than or equal to 65% turbine load based on turbine first stage pressure.

The Main Feedwater [EIIS:SJ](CF) System consists of two Main Feedwater Pumps, piping [EIIS:PSP] and valves [EIIS:V] necessary to deliver the Main Feedwater to the four Steam Generators [EIIS:SG](S/G) during normal plant operation.

The Steam Generator Level Control System [EIIS:JB](IFE) is designed to maintain a programmed level in each Steam Generator. Programmed level is a function of reactor [EIIS:RCT] power. For Unit 2 the program ramps narrow range level from 62% at 0% power to 67% at 100% power. The system controls level by adjusting Main Feedwater Pump speed to maintain the Feedwater Control Valves [EIIS:LCV] at the optimum throttling position for unit load as the valves respond to level and/or steam flow changes, thus ensuring adequate feedwater flow to maintain level. The system also functions to restore and maintain levels at acceptable values during normal transients, thus mitigating undesirable reactor and/or turbine trips. If, however, the levels in the Steam Generators reach the limits which either jeopardize the Steam Generators' ability to act as a "heat sink" due to low-low level or threaten to damage equipment due to moisture carryover as a result of high-high level, then a signal is generated to either trip the reactor via the REACTOR TRIP SYSTEM [EIIS:JD](IPE) for Steam Generator water level Low-Low or trip the Main Turbine, trip the Main Feedwater Pumps and initiate Main Feedwater Isolation via the ENGINEERED SAFETY FEATURES ACTUATION SYSTEM [EIIS:JE](ISE) for Steam Generator water level High-High.

The ENGINEERED SAFETY FEATURES ACTUATION SYSTEM senses selected plant parameters and compares them to predetermined limits. If a limit is exceeded, the system sends actuation signals to those ENGINEERED SAFETY FEATURES components whose aggregate function best serves the requirements of the condition. One such condition is Steam Generator Water Level High-High. For Unit 2 the limit is </= 77.1% of narrow range instrument span. The actuation signals generated are: Main Turbine Trip, Both Main Feedwater Pumps Trip, and Main Feedwater Isolation. Tripping of both Main Feedwater Pumps generates an actuation

signal to start both Motor [EIIS:MO] Driven Auxiliary Feedwater Pumps. The tripping of the main turbine will initiate a REACTOR TRIP SYSTEM signal if reactor power is above the Reactor Trip on Turbine Trip (P-9) Interlock [EIIS:IEL] Setpoint.

EVENT DESCRIPTION

On July 10, 1994, at 1347 hours with Unit 2 in Mode 1, Power Operation, at 100% power, an automatic turbine runback occurred following the manual trip of 2A Main Feedwater Pump. 2A Main Feedwater Pump was tripped due to high thrust bearing temperatures. The thrust bearing temperatures had been above normal and Engineering was in the process of having Operations "bias" the loading on 2A and 2B Main Feedwater Pumps when the outboard thrust bearing temperature unexpectedly increased to 210 degrees. Abnormal Procedure, AP/2/A/5500/03, LOAD REJECTION, was entered and all applicable steps completed. The unit responded as designed during the runback. The runback terminated at a reactor power level of 76%.

At 1348 hours with Unit 2 at 76% power, the Balance of Plant Operator manually reduced turbine load an additional 100 MWE, due to all four Steam Generators' feedwater control valves being fully open and the 2B Main Feedwater Pump's speed being higher than expected. Additionally, the 2B Main Feedwater Pump suction flow was greater than 20000 GPM.

At 1354 hours with Unit 2 at 65% power, the Operator at the Controls noted rapidly rising levels in all four Steam Generators. The Main Feedwater Control Valves were observed to be fully open. Steam Generator levels were at programmed level for the current reactor power and rising. 2A and 2C Steam Generators had the highest indicated levels. The Operator at the Controls took manual control of 2CF-28 (2A Steam Generator Main Feedwater Control Valve) and began throttling it closed in order to control the rising level. 2CF-28 was placed back in automatic and 2CF-46 (2C Steam Generator Main Feedwater Control Valve) was placed in manual and throttled closed in order to control the rising level.

At 1354:45 hours with Unit 2 at 65% power and 2CF-46 in manual, 2C Steam Generator Narrow Range level reached the "High-High" level (P-14) ESFAS Interlock Setpoint of $</=$ 77.1%. The following ESFAS signals were initiated by P-14; a trip of the Main Turbine, trip of both Main Feedwater Pumps, and Main Feedwater Isolation. All components responded to their respective ESFAS signals. The trip of the remaining Main Feedwater Pump satisfied the logic for the ESFAS signal to start both Motor Driven Auxiliary Feedwater Pumps. Both pumps responded to the ESFAS signal.

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

At 1355 hours with Unit 2 at 65% power, a manual reactor trip was initiated. No automatic trip signal was present or required. The unit was below the Reactor Trip on Turbine Trip Interlock (P-9) Setpoint of >/= 69%. The manual reactor trip was initiated as required by the "LOSS OF S/G FEED" Abnormal Procedure, AP/2/A/5500/06. Upon manually tripping the reactor, Emergency Procedure, EP/2/A/5000/E-0, Reactor Trip or Safety Injection, was entered, all appropriate steps performed and transition was made to EP/2/A/5000/ES-0.1, Reactor Trip Response. The Safety Parameters Display System (SPDS) was continuously monitored and EP/2/A/5000/F-0, Critical Safety Function Status Trees, was implemented. All Critical Safety Functions were fully satisfied. The unit was maintained in Mode 3, Hot Standby, until restart was commenced using OP/2/A/6100/05, UNIT FAST RECOVERY.

CONCLUSION

This event occurred when the four S/G's levels became higher than expected following recovery from loss of a single CFP with the unit initially at Rated Thermal Power. During the runback, the operating CFP speed increased and the CF Control Valves fully opened as expected in response to decreasing S/G levels. Upon level recovery, the IFE valve demand began decreasing as S/G level approached programmed level. Three of the four CF Control Valves began closing within twelve seconds of the demand change. The fourth CF Control Valve began closing twenty-five seconds after the demand change. Even the delay of twelve seconds was too long for the CF Control Valves to reduce flow quickly enough to slow the rise of S/G level. It is concluded that the aggregate response of the IFE and the CF Control Valves in automatic was too slow to prevent level increase "overshoot" during the loss of CFP transient.

The operators' actions to reduce CF flow manually had no adverse effect on the ultimate outcome. The exhibited rate of change of all four S/G levels and the values of those levels show that had the operators been able to "turn" level in the 2C S/G, then level in one or more of the other three S/G would have reached the "HIGH-HIGH LEVEL (P-14) SETPOINT", culminating in the same event sequence.

Since the Root Cause for the levels in each S/G reaching or approaching the "HIGH-HIGH LEVEL (P-14) SETPOINT" has been determined to be performance of the S/G Level Control System, the paragraphs below address areas of continuing evaluation.

1) Process instrument calibration in the S/G Level Control System; this is to determine if conservative instrumentation tolerances exhibited a "stacking" effect resulting in more than the required conservatism, such that operational margins are unnecessarily reduced.

2) S/G Level Control System signal generation; this is to ensure that the demand signals have sufficient response characteristics for transient conditions while still maintaining steady state stability.

3) CF Control Valve operation; this is to ensure timely response to transient conditions while still maintaining steady state stability.

4) CF Control Valve throttling position; this is to restore the valves to their "optimum" throttling position.

5) Reassessment of the Auxiliary Feedwater (CA) System piping modification; this is to ensure that the effects, if any, of the larger diameter piping are addressed for CF transients.

Subsequent to this event, a Work Order was issued to check the pneumatic controls for the CF Control Valves to ensure proper Instrument Air [EIIS:LD](VI) System alignment, fitting tightness and adjustment screw locknut tightness.

A review of the Operating Experience Program (OEP) database for the previous 24 months prior to this event revealed that no reactor trip Licensee Event Reports (LERs) had the same cause code. A review of LERs for events other than reactor trips revealed that for the same time frame, none had the same cause code. Therefore, this is not a recurring problem.

CORRECTIVE ACTIONS

IMMEDIATE

The reactor was manually tripped and EP/2/A/5000/E-0, REACTOR TRIP OR SAFETY INJECTION, was entered with transition to EP/2/A/5000/ES-0.1, REACTOR TRIP RESPONSE, to stabilize the unit in Mode 3, Hot Standby.

SUBSEQUENT

1) PT/0/A/4150/02, TRANSIENT INVESTIGATION, was completed and based on review of transient data retrieved on July 10, 1994, the unit was restarted using OP/2/A/6100/05, UNIT FAST RECOVERY. No items affecting unit startup were identified.

2) Work Order #94058318 01 was issued on 08/03/94 to conduct a visual inspection of the pneumatic controls for the Main Feedwater Control Valves to ensure the physical integrity of the pneumatic fittings, the proper Instrument Air System alignment, and the tightness of locking devices on the adjustment screws. No conditions were found to exist that would affect valve operation. This Work Order was completed on 08/04/94.

PLANNED

1) Electrical/Systems Equipment Engineering evaluate process instrument calibration in the S/G Level Control System to determine if conservative instrumentation tolerances exhibit a "stacking" effect such that more than required conservatism exist.

2) Electrical/Systems Equipment Engineering evaluate S/G Level Control System signal generation to ensure that the demand signals have sufficient response characteristics for transient conditions while still maintaining steady state stability.

3) Instrument and Electrical Department evaluate CF Control Valve operation to ensure timely response to transient conditions while still maintaining steady state stability.

4) Electrical/Systems Equipment Engineering evaluate the current CF Control Valve throttling position and restore the valves to their "optimum" throttling position.

5) Mechanical Systems Engineering reassess the CA piping modification to verify that any effects on CF transients have been considered and accounted for.

6) Catawba Safety Review Group will evaluate the need to supplement this report upon completion of the above five PLANNED CORRECTIVE ACTIONS.

SAFETY ANALYSIS

This event was initiated by a turbine runback as a result of a loss of a single CFP at Rated Thermal Power. The turbine trip on "HIGH-HIGH" S/G level occurred as a result of excessive feedwater flow following recovery from the runback. This event is bounded by the Safety Analysis documented in FSAR Section 15.1.2, FEEDWATER SYSTEM MALFUNCTION CAUSING AN INCREASE IN FEEDWATER FLOW.

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, 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 copies of NRC Form 366A) (17)

The operator's action to manually trip the reactor, pre-empted the occurrence of the "LOW-LOW" S/G level reactor trip assumed in the FSAR and therefore mitigated the consequences of the automatic trip.

After a review of this event, it was determined that all systems responded as designed to shut down the reactor and maintain it in a safe condition. There was no unusual release of radioactive material.

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