NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): [illegible]ba Nuclear Station, Unit 2

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

PAGE (3): 1 OF 0 6

TITLE (4): Feedwater Isolation During Unit Shutdown Due To A Personnel Error

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 4	2 4	8 8	8 8	0 1 7	0 0	0 5	2 4	8 8	N/A	0 5 0 0 0
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OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 8

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	TELEPHONE NUMBER (AREA CODE)	
Julio G. Torre, Associate Engineer - Licensing	7 0 4	3 7 3 - 8 0 2 9

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 fifteen single-space typewritten lines) (16)

On April 24, 1988, at 1322:13 hours, a Main Feedwater (CF) Isolation occurred due to a Hi-Hi Level Turbine Trip (P-14) signal from Steam Generator (S/G) 2B. The P-14 signal was generated when S/G narrow range levels increased in response to Operations personnel actions to avoid a Low-Low S/G Level Reactor Trip. The low-low level setpoint was imminent in all four S/Gs due to decreasing Reactor Coolant (NC) System temperature. This downward NC temperature transient was caused by manual Control Rod insertions which were too fast for a stable S/G level response. The Unit was in Mode 2, Startup, at the time of this incident, and was in the process of being shut down in order to investigate and repair a Main Turbine Generator Exciter ground indication.

This incident is attributed to a personnel error. The Control Room Operator did not adequately interpret plant response before continuing Control Rod insertions.

This incident has been reviewed with the individuals involved. The health and safety of the public were unaffected by this event.

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BACKGROUND:

The Main Feedwater (EIIS:SJ) (CF) System supplies feedwater to the four Steam Generators (EIIS:SG) (S/Gs) through the CF nozzles (using either main or bypass feedwater control valves (EIIS:V)) and through the Auxiliary Feedwater (CA) nozzles (using the CF to CA Bypass valves). Using the Controlling Procedure for Unit Shutdown (OP/2/A/6100/02), Operations transfers CF flow from the CF nozzles to the CA nozzles and swaps from the main to the bypass feedwater control valves for flow control. This swap occurs at 15% load during a Unit shutdown. Upon a Feedwater Isolation signal generated by a Hi-Hi S/G Level (78% narrow range level), both Main and Bypass Feedwater Control valves, CF Bypass to CA Nozzle valves, and S/G Reverse Purge valves will all close, and the operating CF Pump(s) (EIIS:P) will trip.

The Auxiliary Feedwater (EIIS:BA) (CA) System is capable of supplying feedwater to the four S/Gs in the event of a loss of CF flow, to the extent that an adequate S/G water inventory can be maintained until a normal cooldown of the Reactor Coolant (EIIS:AB) (NC) System by the Residual Heat Removal (EIIS:BP) (ND) System can be performed. CA flow to the S/Gs is directed through the CA nozzles (using the CA flow control valves). Below 15% load during a Unit Shutdown, the CF to CA nozzle bypass valves are open and the feedwater bypass tempering valves which supply tempering flow to the CA nozzles at full power operation are closed. Upon a loss of both CF Pumps, the CA flow control valves from CA Pumps 2A and 2B open, and the Motor Driven CA Pumps 2A and 2B start. This CA actuation will in turn cause Steam Generator Blowdown (EIIS:W1) (BB) System and Nuclear Sampling (NM) System Containment Isolation valves to automatically close.

DESCRIPTION OF INCIDENT:

On April 23, 1988, at 0657 hours, the Unit 2 Generator or Exciter Field Ground annunciator was received in the Control Room. The Unit was in Mode 1, Power Operation, at 60% power at the time. At approximately 2015 hours, Operations personnel began to reduce power from 60% to 10%, due to an apparent Main Turbine Generator Exciter ground. On April 24, 1988, at 0537 hours, Operations personnel tripped the Main Turbine off line per Transmission Department request. At 0755 hours, the Control Room Operator at the Controls (OATC) began to reduce Reactor power to approximately 7.5% using boration and Control Rod (EIIS:ROD) insertion.

At 1310 hours, the Unit Supervisor instructed the OATC to resume power reduction to Mode 2, Startup, and subsequently to shut down the Reactor (after which the Unit Supervisor left the Control Room). The OATC then walked over to the Shift Supervisor (SS), who was talking to a group in another area of the Control Room, and informed him that the Unit Supervisor had told him to shut down the Unit to Mode 3, Hot Standby, and that he (the OATC) was making preparations to do so. The SS said "Okay", and that he would "be over there in a minute or two". The SS had intended this statement to mean that he would come over and discuss the shutdown with the OATC. The OATC interpreted this statement as permission to begin the shutdown. Having received no special instructions from the Unit Supervisor, the OATC knew of no reason why he should not proceed.

At approximately 1312 hours, the OATC inserted the Bank D Control Rods 5 steps and monitored the following plant parameters: Reactor Power, NC System operating loop temperature (TAVG), and S/G levels. Seeing only slight changes in these parameters, the OATC inserted the Control Rods 10 more steps. Following this insertion, the Unit entered Mode 2, Startup (Reactor Power decreased to 5%). Noting no apparent significant changes in plant parameters, the OATC inserted the Control Rods an additional 10 steps. Immediately following this insertion, S/G Low Level Deviation annunciators were received for all four S/Gs. TAVG had dropped enough to cause S/G narrow range levels to shrink to approximately 45%.

At approximately 1315 hours, the SS returned to the Control Room horseshoe area and observed a cooldown of the NC System already in progress: decreasing TAVG, NC Pressurizer level, and S/G levels. In order to prevent what he considered to be an imminent Reactor Trip on Low Low S/G Levels, the SS instructed the OATC to pull rods 5 steps and the Unit 2 Control Room Balance of Plant (BOP) Operator to increase CF Pump Turbine (CFPT) 2B speed to maximum in order to feed the S/Gs. During and immediately following the performance of these actions, Reactor power decreased to less than 2%, TAVG decreased to less than 547 degrees F, and NC Pressurizer (EIIS:PRZ) level decreased to 16.8% (resulting in a Letdown Isolation at 1318:32 hours). Charging flow was at this time being provided by the Reciprocating Charging Pump. Seeing that the increased feedwater flow had contributed to an increased NC System cooldown rate and that S/G narrow range levels were increasing rapidly, the SS instructed the BOP Operator to reduce CFPT 2B to near minimum speed. At 1319 hours, the BOP Operator reduced CFPT 2B speed, and CF flow decreased. At this time, S/G wide range levels had begun to decrease (at 72%) and narrow range levels had begun to slow their rate of increase (at 67%), and the SS believed that S/G Hi-Hi Level Turbine Trip (P-14) setpoints would not be reached. By 1321 hours, TAVG had recovered back to 551 degrees F (the Technical Specification lower limit for TAVG when the Reactor is critical). During the next 90 seconds, the increased S/G water inventory continued to heat up (due to the increasing TAVG) and the subsequent swell pushed narrow range levels in all four S/Gs over 78%, the P-14 setpoint. Between 1322:13 and 1322:26 hours, P-14 signals were generated in all four S/Gs.

Plant response was as designed, with the P-14 generated Feedwater Isolation signal closing the appropriate valves. CFPT 2B tripped, and CA Pumps 2A and 2B started and supplied flow through automatically opened CA flow control valves to all four S/Gs. BB and NM Containment Isolation valves closed automatically on CA actuation. Between 1322:49 and 1323:01 hours, all four S/G P-14 signals had cleared due to decreasing S/G narrow range levels. At 1324 hours, narrow range levels again began to rapidly increase (due to CA flows) and between 1324:24 and 1325:40 hours, P-14 signals were received for all four S/Gs. As levels were reaching the P-14 setpoints, the OATC closed the CA flow control valves. Between 1327:53 and 1328:27 hours, S/G levels decreased to less than the P-14 setpoint and continued to rapidly decrease. Responding to this rapid decrease, the OATC reopened the CA flow control valves (between 1329:22 and 1329:39 hours). By 1330 hours, the BOP Operator had restored normal letdown and at approximately 1331 hours, the OATC throttled the CA flow control valves in anticipation of another S/G level swell. Between 1334:00 and 1337:35 hours, P-14 signals were received (for the third time) in all four S/Gs. CF was still secured, therefore no challenges to the plant were brought about by the second or third CF Isolations. Between 1340:14 and 1354:40 hours, S/G levels decreased below the P-14 setpoints, and by 1400 hours the OATC was stabilizing S/G levels with CA. Over the next hour, S/G levels were controlled with CA as Control Rods were inserted. By 1450 hours, the Unit had been shut down to Mode 3, Hot Standby.

CONCLUSION:

This event is attributed to a personnel error. The OATC did not monitor plant parameters long enough between Control Rod insertions. Although the power reduction performed by the OATC would have resulted in the same downward TAVG transient on Unit 1, the S/G narrow range level controls on the Unit 1, D-3 type S/Gs would not have responded as sensitively to the TAVG change as the Unit 2, D-5 type S/Gs did. The OATC had received training on D-5 type narrow range level sensitivity and shutting down a Reactor, but had not received simulator training on the D-5 type S/G Reactor shutdown at this time. Since January 1988, Operator training has included shutdowns for both types of S/Gs at the Catawba Training Center. The OATC had performed a shutdown once before (on the Unit 1, D-3 type S/G).

Miscommunication occurred between the Control Room Operator and the Shift Supervisor, after which the Control Room Operator believed that he should continue rod insertion without the Shift Supervisor's presence. Also, the Unit Supervisor instructed the Control Room Operator to continue the shutdown, but did not warn him of potential S/G level swings due to NC temperature changes. The SS was involved in a conversation in another area of the Control Room, and was not observing the impending shutdown. The SS stated that had he not been preoccupied, he would have cautioned the OATC on the effects on Unit 2, D-5 type S/G levels during shutdown. Nevertheless, the Control Room Operator was qualified to individually perform the shutdown, based on his training and experience, without the presence of the SS.

Although two more P-14s were received following the incident, the CF Pumps were tripped, the CF valves were still closed, and the S/Gs were already being fed by CA. In reacting to the transient following the Control Rod insertions, the main concerns of the SS were to prevent a Reactor Trip on Low-Low S/G Levels, and to prevent TAVG from decreasing to and remaining below 551 degrees F (per Technical Specification 3.1.1.4). Operations personnel were aware of the P-14 risk in preventing these events. The increased feedwater flow resulting from increasing CFPT 2B output contributed to the P-14, although narrow range S/G levels were still falling when speed was increased. It is the opinion of the SS that the P-14 could not have been prevented following such a downward power transient on D-5 type S/G. The OATC and BOP Operator performed the proper actions once the initial transient occurred, although three P-14 signals were received in all four S/Gs. The three P-14 signals would less likely have occurred if CFPT 2B speed had been more gradually increased. In addition, the Letdown Isolation would have been less likely if NC Pressurizer level would have been raised prior to Control Rod insertions, and if Centrifugal Charging Pump A had been in service, providing a higher charging flow than the Reciprocating Charging Pump could provide. The Reciprocating Charging Pump was in operation at this time in order to ensure that

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

it could supply adequate charging flow on a regular basis as well as the other two charging pumps.

2BB-61B, S/G C Blowdown Containment Isolation Valve, closed but did not indicate closed upon the BB Isolation signal. A work request was written to investigate and repair the indication for 2BB-61B.

There have been numerous P-14 actuations due to D-5 type S/G level control problems. A Station Problem Report has been written, and actions are being taken to revise or replace the current S/G level control system. There have also been several P-14 actuations due to personnel errors: See LER 414/87-22 (P-14 due to 2CF37, Main CF Control Valve, failing open while being worked on by IAE personnel), LER 414/87-24 (P-14 due to accidental overfill during temperature reduction), LER 414/87-30 (S/G C P-14 due to personnel calibrating Channel 4 while Channel 2 was tripped), LER 413/87-46 (S/G A P-14 due to personnel removing jumpers from P-14 circuitry while S/G was filled for hydro), LER 413/86-13 (P-14 due to accidental overfill due to BB System isolation during clam flush testing), and LER 414/86-30 (S/G B P-14 due to Control Room Operator not recognizing the need to increase CFPT speed prior to adding water to the NC System). Of these six P-14 actuations, two were due to an accidental S/G overfill: LER 414/87-24 and LER 413/87-46). In both of these cases, the S/Gs were intentionally filled to a certain level, after which a level swell occurred due to a heatup of the added water. The previous corrective actions for these errors (discussing the problems with personnel involved) could not have prevented this incident, since they applied to preventing P-14 actuations, and were not associated with rod insertions. Operations personnel were fully aware that a P-14 actuation could result from their actions to prevent a Reactor Trip (resulting from Low Low S/G levels) or Technical Specification violation.

CORRECTIVE ACTION:

SUBSEQUENT

(1) Operations personnel restored NC System Letdown.

(2) Operations personnel brought S/G levels under control using Auxiliary Feedwater.

(3) The incident was discussed with the individuals involved.

(4) Work Request 6426 PRF was initiated to investigate and repair 2BB-61B, S/G C Blowdown Containment Isolation valve.

PLANNED

(1) The S/G overfill events will be discussed during Operator Requalification Training and practices to limit recurrence will be suggested. Also, situations that could lead to overfill events will be reviewed.

(2) Simulator training will be provided to all Reactor Operators on response to events that lead to S/G overfill. This training will be directed at the Unit 2 D-5 S/G model.

SAFETY ANALYSIS:

The CF, CA, BB, and NM Systems responded properly to the P-14 signal. The CF reverse purge valves and CF to CA nozzle valves closed and CFPT 2B tripped as designed for pump protection. The CA system responded properly, with CA Pumps A and B supplying normal flows to the four S/Gs. Letdown was isolated as expected, when Pressurizer level decreased to 16.8%. TAVG remained below 551 degrees F (its minimum value was 546 degrees F) for three minutes, with no adverse effects.

This incident was well within the accident described in Section 15.2.7 of the Final Safety Analysis Report (FSAR), which describes the required plant responses following a loss of normal feedwater flow. The CA System was fully capable of providing S/G heat transfer capability from the NC System and returning the plant to a stable condition.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv).

The health and safety of the public were unaffected by this incident.

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May 24, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-17

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-17 concerning a Feedwater Isolation during unit shutdown due to inadequate interpretation of plant response. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

JGT/25/sbn

Attachment

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