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)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 1 4	1 OF 0 6

TITLE (4) Feedwater Isolation Caused By Steam Generator High Level Due To A Valve Failure And 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 3	1 4	8 8	8 8	0 1 3	0 0	0 4	1 3	8 8	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)								
POWER LEVEL (10)	0 0 0	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	
O	B A	V	B 3 5 0	See Below							

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 March 14, 1988, at 1158 hours, while a flush of Auxiliary Feedwater (CA) System piping was in progress, Steam Generator (S/G) 2A Narrow Range level reached the high high level setpoint and caused a Feedwater Isolation. The Unit was in Mode 4, Hot Shutdown, at the time.

This incident has been attributed to an equipment failure. 2CA62A, CA Pump 2A to S/G 2A Isolation valve, did not fully close and caused the S/G to be overfilled. NPRDS reportability of this failure is under evaluation. This incident has also been attributed to a Personnel Error. While performing a valve alignment to support a Chemistry activity, a Control Room Supervisor mistakenly closed the incorrect valve and caused the isolation of all blowdown from the S/Gs. This isolation hindered efforts to maintain S/G level during the subsequent swell of the S/G 2A inventory. The S/G Blowdown System was returned to service and the S/Gs were returned to normal level. The Feedwater Isolation signal was reset and all valves and pumps returned to their previous alignment. A Problem Investigation Report has been initiated to further evaluate the cause of 2CA62A failure to close.

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

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

BACKGROUND:

The Auxiliary Feedwater (EIIS:BA) (CA) System ensures a sufficient feedwater supply to the Steam Generators (EIIS:SG) (S/Gs) in the event of loss of Main Feedwater (EIIS:SJ) (CF) System, to remove stored and residual core energy from the Reactor Coolant (EIIS:AB) (NC) System. Each Unit is provided with a separate CA System which is designed to be capable of cooling down the NC System to the temperature at which the Residual Heat Removal (EIIS:BP) (ND) System may be operated. There are two motor (EIIS:MO) driven CA pumps (EIIS:P) (MDCAPs) and one steam turbine (EIIS:TRB) driven CA pump (TDCAP) per Unit. Each MDCAP is capable of supplying two S/Gs while the TDCAP is capable of supplying all four S/Gs.

CA flow to each S/G is modulated by air operated control valves (EIIS:V) (CA36, CA40, CA44, CA48, CA52, CA56, CA60, and CA64). These valves fail open to predetermined positions on a CA autostart signal. The travel stops on these valves are set to optimize the system flow rates for various accident scenarios. The valves can be remotely operated from the Control Room within the preset limits after the CA autostart signal has been reset. The Catawba Final Safety Analysis Report (FSAR) accident analyses assume degraded operation of the CA System such that a total of 491 gpm of CA flow is delivered to only two intact S/Gs.

The S/G Blowdown System (EIIS:WI) (BB) is used to help maintain proper S/G secondary side chemistry. Non-volatile solids which tend to concentrate in the S/Gs are removed through continuous blowdown of fluid from the shell side of the S/Gs. The blowdown is routed to a Blowdown (EIIS:TK) Tank in the Turbine Building.

Steam in the Blowdown Tank is used to maintain Blowdown Tank pressure. To control pressure, steam is routed through a pressure control valve to the steam side of the D Feedwater Heaters (EIIS:HTR).

Water and solids in the Blowdown Tank are routed through a pump, or bypass line, through heat exchangers, filters (EIIS:FLT), demineralizers (EIIS.DM), and returned as condensate grade water to the Condenser Hotwell. Provisions for the bypass of most components and routing of blowdown to the Turbine Building sump has been provided for in the piping.

Blowdown Tank level control is accomplished through BB Pump Discharge Flow Control and BB Pump Recirc valves. As tank level decreases to 50%, the pump discharge valve starts to close and the recirc valve starts to open. At low level in the tank, the discharge valve is fully closed and the recirc valve is fully open to maintain tank level. If the Blowdown Tank level reaches a low-low setpoint, the BB pumps are tripped to prevent cavitation. After clearing the low level condition, the pumps may be manually restarted.

As tank level increases from normal operating level, the BB Pump Discharge valve is designed to fully open to regain normal level. If the level reaches the high high level setpoint, all flow from the S/Gs is stopped automatically by closure of the BB Flow Control valves. Automatic isolation of these valves will also

occur on high tank pressure. These valves are automatically reset when the abnormal tank parameter is returned to normal, and blowdown from the S/Gs to the Blowdown Tank will resume.

DESCRIPTION OF INCIDENT:

On March 9, 1988, while swapping from the S/G CF Bypass Control valves to the S/G CF Control valves, the Unit received a Feedwater Isolation signal and a subsequent Turbine and Reactor (EIIS:RCT) trip. During a resulting CA autostart, low suction pressure was sensed by CA Pump 2A suction pressure switches and an automatic swap to the assured suction source, the Nuclear Service Water (EIIS:BI) (RN) System, occurred. Introduction of the RN water from Lake Wylie and Asiatic clams from the suction piping caused fouling of the CA Flow Control valves for S/Gs 2A and 2B (see LERs 414/88-12 and 413/88-15).

Corrective measures, including the removal of clams from CA piping by flushing, ensued on March 11, 1988, after both Units entered Mode 4, Hot Shutdown.

On March 14, 1988, Operations and Performance personnel were preparing to flush the sections of CA piping from CA Pump 2A to S/Gs 2A and 2B. The CA Flow Control valves were aligned to the fully open position to maximize the flush velocity. Clam debris was to be caught in the Cavitrol cages of the flow control valves.

S/Gs 2A and 2B were drained to approximately 25% Narrow Range (NR) level to make room for the volume of water used during the flush. The S/Gs were ready for filling at 1126 hours, and the drain was secured. At 1130 hours, Chemistry personnel telephoned the Control Room and requested that the BB Demineralizers be bypassed to permit swapping the in-service demineralizer. This is normally accomplished by opening 2BB176, BB Demineralizer Bypass valve. In this case however, the Control Room Senior Reactor Operator (CRSRO) closed 2BB188, Prefilter to Demineralizer Isolation valve. This stopped all flow from the BB Tank and tank level started to rise.

At 1135:01 hours, the CRSRO realized the alignment error, repositioned 2BB188 to OPEN, and also opened 2BB176. This rapidly increased flow from the BB Tank to the Hotwell.

At 1137:50 hours, the BB tank reached the low level setpoint and flow was automatically stopped from the tank. This again caused tank level to rapidly rise which resulted in BB tank high high level interlock actuation. All flow was automatically stopped from the S/Gs to the BB tank.

At 1148:51 hours, the CRO aligned CA Pump 2A to S/G 2A and started the flush. The flush was intended to fill the S/G past the 65% high level alarm to 70% NR level and then secure flow by closing 2CA62A, CA Pump 2A to S/G 2A Isolation. The CRO would then align flow to S/G 2B and flush that CA line. The S/G levels were taken past the high level alarm to maximize run time for the flush. This process had been performed previously without difficulty and had allowed adequate margin before reaching the 78.1% Feedwater Isolation Setpoint. It was predetermined that should problems arise, the CRO would trip the pump.

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

In accordance with the plan, S/G 2A level was increased to 70% and the CRO depressed the CLOSED pushbutton for 2CA62A. The valve did not fully close.

At 1155:21 hours, the level had reached 77% and 2CA62A had not fully closed, so the CRO tripped the pump. Immediately afterwards, 2CA62A closed. Also at approximately this time, Chemistry personnel called the Control Room to inform them that the BB Demineralizers could be returned to service.

Level in S/G 2A continued to rise despite the isolation of the CA System. In an effort to isolate possible leakage from the CF to CA Systems, the CRO closed 2CA185, Tempering Flow to S/G 2A Isolation, and 2CA149, CF to CA Bypass valve. Level continued to increase.

At 1158:06 hours, 2BB176 was closed by the CRSRO. By this time, Control Room personnel realized they had lost BB flow and dispatched a Nuclear Operations Technician to investigate.

At 1158:19 hours, as S/G 2A reached the high high level setpoint (78%), Feedwater Isolation occurred as designed. CF Pump 2B automatically tripped, the remaining tempering flow valves were closed and the CF Containment Isolation Bypass valves closed, with the exception of 2CF90, S/G 2A Containment Isolation Bypass valve. S/G level continued to increase to 84% NR level.

At 1201 hours, the CRO aligned the CA System to perform the flush of piping to S/G 2B. This flush was secured at 65% level by tripping the CA pump and then closing the CA Pump 2A to S/G 2B Isolation valve.

Operations personnel at the BB Tank evaluated the situation and concluded that draining some of the BB Tank volume to the Turbine Building Sump would be of benefit. They received permission to dump BB water to the sump from Health Physics and Chemistry personnel and then opened 2BB48, BB Tank Bypass to the Turbine Building Sump. Once flow was re-established, the valve was reclosed. Flow continued to increase and the low flow alarm cleared.

At 1224:48 hours, 2CF90 indicated closed per computer indication. This indication cycled between CLOSED and NOT CLOSED repeatedly for approximately 7 minutes. At 1225:02 hours, the BB Tank high level interlock cleared and flow automatically resumed from the S/Gs to the BB Tank. After level decreased in S/G 2A to less than the 78% high level interlock setpoint, the CRO reset the Feedwater Isolation signal and realigned the affected v[illegible]. He then restarted the CF Pump.

As blowdown continued, level in both S/Gs 2A and 2B decreased to normal levels. The CRO continued to drain these S/Gs to 25% level to perform a second flush.

At approximately 1330 hours, a second flush of both CA lines was completed. At the conclusion of each of these flushes, the CA pump was manually tripped and the CA Isolation valves were closed. By 1356 hours, the flushes were completed and the S/G levels were returned to normal.

CONCLUSION:

This incident has been attributed to the failure of 2CA62A to close within the required period of time. Because the valve did not respond appropriately, the S/G was overfilled. S/G levels continued to increase after isolation of CA due to swelling of the relatively cool Feedwater as it remained in the S/G and heated up.

2CA62A is a Borg-Warner gate valve, Model Number CN 1500-1206J-219, which is positioned by a Rotork control Motor Operator Model Number 16NA1-57. A search of the NPRDS database revealed there are 16 applications of this valve with two previous failures reported. There are 44 applications of the motor operator in service with 8 reported failures. The NPRDS reportability of this failure of 2CA62A will be evaluated.

The actuator for 2CA62A was Motor Operated Valve Actuator Tested (MOVAT) on December 4, 1987, and the actuator was calibrated to deliver the proper closing force per Design Engineering specifications. Also, Performance personnel tested the valve stroke time on February 19, 1988, with acceptable results.

This incident has also been attributed to a personnel error. If the CRSRO had not inadvertently secured all blowdown flow, level in the S/G could have been maintained lower than the 78% trip setpoint and the incident would not have occurred.

This incident will be discussed with the appropriate personnel with emphasis on the use of flow diagrams before performing infrequent plant evolutions which do not require a procedure.

There have been several previous ESF Actuations due to equipment malfunctions at Catawba. This is a recurring event. However, there are no previous incidents in which an Auxiliary Feedwater valve failed to close and overfilled the S/G.

There have been several ESF actuations due to Personnel Error. This is also considered a recurring event.

CORRECTIVE ACTION:

SUBSEQUENT

(1) Operations personnel re-established Blowdown and reduced S/G levels.

(2) The CRO reset the Feedwater Isolation signal and realigned the affected valves.

(3) The CRO restarted CF Pump 2B.

(4) Problem Investigation Report 2-C88-0143 was initiated to further evaluate the cause of 2CA62A failing to close under high differential pressure conditions.

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(5) Work Request 39998 OPS was initiated by Operations to investigate the cycling indication of 2CF90 in the closed position.

PLANNED

This incident with the appropriate personnel with emphasis on consulting flow diagrams before performing infrequent plant manipulations without a procedure.

SAFETY ANALYSIS:

Level in S/G 2A reached approximately 84% before blowdown was re-established and reduced level to normal.

The slow closure of 2CA62A would have only affected maintaining pressure in S/G 2A had a pipe break occurred upstream of 2CA62A. In this case, three other S/Gs would be intact and S/G 2A would not be required (reference FSAR Accident Analysis 15.2.8). Additionally, the Unit was operating in a mode (Hot Shutdown) in which the Auxiliary Feedwater System is not required to be operable.

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

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

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April 13, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-13

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-13 concerning a feedwater isolation caused by a steam generator high level due to a valve failure and a personnel error. 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

JGT/82/sbn

Attachment

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