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 4	1 OF 0 8

TITLE (4) Reactor Trip Followed By An Auxiliary Feedwater Suction SWAP To The Nuclear Service Water System Due To Equipment Failures

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

POWER LEVEL (10): 0 1 5

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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 March 9, 1988, at approximately 1825 hours, Control Room Operators (CROs) were swapping from the Steam Generator (S/G) Main Feedwater (CF) Bypass valve to the S/G CF Control valve for S/G 2B, during Unit startup following End-of-Cycle 1 refueling outage. When the CRO placed S/G CF Control valve in AUTO, the valve opened unexpectedly. The CROs immediately took manual control of the valve and the CF Pump Turbine (PT) to stabilize feedflow to S/G 2B. Levels in S/Gs C and D continued to rise. At 1825:36:317 hours, S/G 2D Hi Hi Level occurred tripping the Turbine and CFPT, and initiating Feedwater Isolation. Level in S/G 2A was decreasing and at 1825:44:179 hours, Reactor trip occurred from 15% power due to low low level in S/G 2A. After the initial trip recovery, the CROs discovered that Motor Driven Auxiliary Feedwater (CA) Pump Train A had swapped suction to the Nuclear Service Water (RN) System. The Unit was returned to Mode 2, Startup, on March 13, 1988, at 1501 hours, and Mode 1, Power Operation, at 2035 hours.

This incident has been attributed to an equipment failure. The unexpected opening of the S/G CF Control valve was attributed to a defective printed circuit card and a defective controller driver card. Duke Power personnel replaced the defective cards. Three CA suction pressure switches were discovered to be slightly out of calibration in the high direction. At the time of the transient, Upper Surge Tank (UST) level was thought to be 95% full, however, later the level indication chart recorder was discovered to have been broken at the time. The actual level is believed to have been approximately 65%. Performance transient testing showed the CA suction swap to be the result of operating with the CA Condensate Storage Tank isolated while not maintaining adequate level in the UST. 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 Main Feedwater (EIIS:SJ) (CF) System supplies feedwater to the Steam Generators (EIIS:SG) (S/Gs) at the temperature, pressure, and flow required to maintain proper S/G water levels commensurate with Reactor (EIIS:RCT) power output and Turbine (EIIS:TRB) steam requirements. The CF System contains two 50% capacity variable speed Turbine Driven CF Pumps (EIIS:P) (CFPTs). The CFPT speed may be manually or automatically controlled. The CF Pumps discharge through two stages of high pressure CF heaters. Then the feedwater divides into four CF lines, each supplying one of the four S/Gs. Each of the four CF lines contains a CF Control valve (EIIS:V), a CF Bypass Control valve, a CF Bypass to Auxiliary Feedwater (EIIS:BA) (CA) Nozzle (EIIS:NZL) Isolation valve, two CF Check valves, and a CF Containment Isolation valve. The CF Isolation valves function to terminate flow in either direction following a feedwater Isolation signal and also function to prevent or allow admission of CF to the S/G CF nozzles during various modes of operation. The CF Bypass Control valves are utilized to control CF flow to the S/G CA nozzles up to approximately 15% load, after which the CF Control valves are utilized to control flow to the S/G CF nozzles when the CF Containment Isolation valves are opened. Prior to transferring from the CA to the CF nozzle, the CF Bypass Control valves are slowly throttled closed (to between 45% and 75% open or as required) while opening the CF Control valve to approximately 6%. The nozzle transfer is accomplished by closing the CF Containment Isolation Bypass valve and then manually closing the CF Control valve while simultaneously opening the CF Containment Isolation valve. The CFPT speed should be adjusted to maintain 1280 to 1300 psig during nozzle transferring to improve CF Bypass Control valve response. The CF Control and Bypass Control valves may be manually or automatically controlled.

The CA System assures sufficient feedwater supply to the S/Gs in the event of loss of the CF System, to remove primary coolant stored and residual core energy. If condensate grade supplies are not available, then provisions are provided to align the CA System with the Nuclear Service Water (EIIS:BI) (RN) System for a long term heat sink source for the S/Gs.

The automatic switchover to RN takes place only if the CA System has been automatically initiated by a CA start signal and the CA pump suction pressure is low. Low pressure in one of three Train A or B pressure switches will alarm the OAC. Two out of three low pressure indication in the Train A or B pressure switches will align the associated RN Train to its associated Train A or B motor driven pump, if that pump has received a CA pump start signal and a time delay relay has timed out. Either Train A or B RN suction sources can align to the Turbine Driven CA Pump if all of the following has occurred:

(1) The CAPT has received an auto start signal.

(2) The pump turbine steam supply valve SA2 or SA5 is open.

(3) The pump turbine trip and throttle valve is open.

(4) Two out of three low suction pressure indication from the Train A or B pressure switches and their time delay relay has timed out.

DESCRIPTION OF INCIDENT:

On March 9, 1988, at 1245 hours, the Main Turbine/Generator was placed on line following End-of-Cycle 1 refueling outage. When Reactor power reached approximately 20%, Control Room Operators (CROs) began swapping from the S/G CF Bypass Control valves to the S/G CF Control valves for each S/G at approximately 1800 hours. CF Pump Turbine (CFPT) 2A was in automatic control at the time. At approximately 1825 hours, the Operator at the Controls (OATC) began the swap for the final S/G. When he placed 2CF37, S/G 2B CF Control valve, in AUTO the valve unexpectedly opened. Levels in S/G C and D decreased. CFPT 2A began to oscillate in response. The OATC took manual control of 2CF37 and throttled feed flow to S/G 2B. The Balance of Plant Operator (BOP) took manual control of CFPT 2A. Levels began to rise in S/G 2C and 2D. At 1825:36:317 hours, Main Turbine trip and Feedwater Isolation occurred due to S/G 2D Hi Hi Level alarm (P-14). At 1825:36:365 hours, CFPT 2A Protective Trip alarm occurred. At 1825:37 hours, both Motor Driven CA Pumps started automatically due to loss of both CFPTs, and S/G Blowdown Isolation was initiated automatically. At 1825:37:429 hours, S/G 2D Hi Hi level Turbine Trip/CF Isolation alarm returned to normal.

At 1825:40 hours, CA Pumps Train A and B Loss of Normal Suction Lo alarm occurred. At 1825:44 hours, 2RN250A, RN Header A to CA Pump Suction Isolation valve, and 2CA15A, CA Pump 2A suction from RN System valve, began opening automatically.

Level continued to decrease in S/G 2A and at 1825:44:179 hours, S/G 2A Lo Lo Level Reactor Trip alarm was received in the Control Room. At 1825:44:247 hours, the Reactor Trip Breakers opened and at 1825:47 hours, the CRO initiated a manual Reactor trip signal.

At 1825:49 hours, 2CA15A had fully opened and at 1825:58 hours, 2RN250A had fully opened which completed the CA suction swap to RN.

At 1826:05:833 hours, S/G 2B Lo Lo Level Reactor Trip alarm was received in the Control Room, and the Turbine Driven CA Pump automatically started due to low low level in two out of four S/Gs.

At 1826:31 hours, CA Pumps Train A and B Loss of Normal Suction alarm returned to normal.

At 1827:20 hours, the CRO secured the Turbine Driven CA Pump.

At 1827:55 hours, the CRO discovered 2RN250A open and manually closed it.

At 1902:23:923 hours, S/G 2B Lo Lo Level Reactor Trip alarm reset as S/G level was being restored.

At 1907:34 hours, the on-coming shift discovered 2CA15A open. CRO manually closed it.

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At 1911:16:637 hours, S/G 2A Lo Lo Level Reactor Trip alarm cleared. At 1939:40 hours, the CRO reset the S/G Blowdown Isolation signal and realigned the affected valves. At approximately 2155 hours, the CRO secured both Motor Driven CA Pumps.

Following the Reactor Trip and recovery, Unit 2 was shutdown to Mode 4, Hot Shutdown, and Unit 1 was shutdown to Mode 4 due to Asiatic clams in the RN lines supplying the CA System (see LER 413/88-15). Unit 2 was returned to Mode 2, Startup, on March 18, 1988, at 1501 hours, and entered Mode 1, Power operation at 2035 hours.

CONCLUSION:

This incident has been attributed to an equipment failure. The unexpected opening of 2CF37, when placed in AUTO, was the initiating event for the transient. There was no identifiable steam flow/feed flow mismatch and S/G level was above the setpoint. Operator action could not adequately compensate for the changes in S/G levels and CFPT speed.

When the BOP took manual control of CFPT 2A, his attention was diverted from monitoring of S/Gs C and D levels. At the point the BOP noticed the rapid rate of level increase, he took action to correct the level but the trip occurred before Operator and valve response could prevent it. Prior to the initiation of the valve swap, the Shift Supervisor recalls that S/Gs A, C, and D were in AUTO. Although transient monitor plots show levels to be unstable and control response to be sluggish. S/G level controls have been a problem in the past and modifications were made during this refueling outage to assist in evaluation of current problems and redesign of the level control system.

Following the incident, Operations personnel originated Work Request 39789 OPS to investigate/repair the cause of 2CF37 fully opening when placed in AUTO. Duke Power Instrumentation and Electrical (IAE) personnel discovered and replaced a defective printed circuit card and controller driver card (Model 283A16G03). Both cards are manufactured by Westinghouse. Following the maintenance, the valve worked (Model 2838A30G01) properly when placed in AUTO.

The printed circuit card failures are not reportable to NPRDS because they are used for Reactor control and not Reactor protection.

The CA Train A suction swap to the RN System occurred due to low CA suction which was a result of operating with the CA Condensate Storage Tank (CACST) isolated from Unit 2 while not maintaining adequate level in the Upper Surge Tank (EIIS:TK) (UST). The CACST had been isolated due to leakage. At the time of the transient, UST level was thought to be 95% full. The level indication chart recorder was later discovered to have been broken at the time and indicating a false trace at 95%. The actual level is believed to have been approximately 65% full based upon an Operator Aid Computer (OAC) alarm.

Following the swap, IAE personnel performed calibration checks on the Train A CA suction pressure switches 2CAPT5220, 2CAPS5221, and 2CAPS5222 and on the Train B CA suction pressure switches 2CAPS5230, 2CAPS5231, and 2CAPS5232. The

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calibration checks revealed that two Train A and one Train B pressure switches were out of calibration in the high direction slightly. The auto-swap time delays were verified to be within calibration. The circuitry was verified to be functioning properly. The auto-swap impulse lines were checked for obstructions. The Train B impulse line was found to be partially obstructed at the root valve. The line was cleared and the root valve internals were replaced. Additionally, the CA Pump suction check valves were disassembled and inspected to verify proper functioning.

The swap of Train A only indicates that at least two of the Train A pressure switches actuated, and the absence of a swap on Train B indicates that only one of the Train B switches actuated. This is supported by the calibration checks performed on the pressure switches. The lack of Turbine Driven CA Pump suction swap is attributed to one of the Train A pressure switches having cleared at the time of the Turbine Driven CA Pump start.

Following the calibration and inspection, Performance developed and conducted a transient test during Mode 3, Hot Standby. The transient test was set up with the following conditions:

-UST level to be approximately 60-80% full level (approximate to time of trip)

-S/G pressure between 900-1000 psig

Following the Performance transient test, a loss of CF Pump initiated an auto-start signal for the Motor Driven CA Pumps, which again resulted in an auto-swap to the RN System for Train A. Results of the test show the CA suction swap to be a result of operating with the CACST isolated while not maintaining adequate Upper Surge Tank level.

Prior to Unit startup and as a result of the CA to RN suction swap, 2CA60 and 2CA56, CA Flow Control valves, were discovered to be clogged with Asiatic clams (see LER 413/88-15). The clams were removed from the CA and RN Systems prior to Unit startup.

During this incident, several other components were discovered to exhibit inadequate performance. Performance originated Work Request 6330 PRF to investigate/repair faulty OAC indication on 2SB18, Main Steam Bypass to Condenser Control valve. Performance originated Work Request 6331 PRF to investigate/repair the failure of S/G 2B CF Containment Isolation Valve Closed annunciator to light. Performance originated Work Request 6332 PRF to investigate/repair digital point D2802 not functioning (no indication of Turbine Trip given). Performance originated Work Request 6333 PRF to investigate/repair improper response of 2SB15, Main Steam Bypass to Condenser Control valve.

There have been 3 previous Incident Investigation Reports involving Engineered Safety Features actuations due to card failures (see LERs 414/87-19, 413/86-25, and 414/87-27). This incident is considered to be recurring.

CORRECTIVE ACTION:

IMMEDIATE

CROs took manual control of CFPT 2A and 2CF37.

SUBSEQUENT

(1) CROs stabilized the Unit in Mode 3, Hot Standby, following Reactor Trip.

(2) CROs closed 2RN-250A and 2CA-15A.

(3) Work Request 39789 OPS was originated and 2CF37 was repaired.

(4) IAE performed calibration check on all Train A and Train B pressure switches, time delay relays, and verified circuitry to function properly.

(5) Auto-swap pressure switch impulse lines checked for obstructions.

(6) CA Pump Suction Check valves were disassembled and inspected.

(7) Performance developed and conducted transient test.

(8) Future operation of CA System will be to align the CA Pumps to the CACST to increase available suction header pressure.

(9) Design Engineering is investigating the possibility of decreasing CA suction low pressure setpoint and increasing the switchover time delay.

SAFETY ANALYSIS:

Following the S/G 2D high high level (P-14) and consequent decrease in S/G level, the Reactor automatically tripped on S/G 2A low-low level. Prior to Reactor trip, the P-14 initiated an automatic Turbine trip, a Main Feedwater Isolation, and automatic trip of the CF pumps. Both Motor Driven CA pumps autostarted on loss of both CF pumps, and the Turbine Driven CA pump autostarted 29 seconds later upon low-low level in two-out-of-four S/Gs. The redundant steam supply valves for the Turbine Driven CA pump, SA2 and SA5, opened within 3 and 6 seconds, respectively, of the Solid State Protection System (SSPS) autostart signal. The Reactor trip breakers opened within 68 milliseconds of the Reactor Trip signal and all of the control rods fell to the bottom of the core, reducing Reactor power to decay heat level. The Operators initiated manual Reactor trip within 4 seconds of the automatic trip.

Reactor coolant temperature decreased to a minimum of 537 degrees F 19 minutes post-trip, and stabilized at 539 degrees F 30 minutes post-trip, 18 degrees F from the no-load target of 557 degrees F. Pressurizer pressure decreased to a minimum of 2175 psig 4 minutes post-trip, and stabilized at the no-load target of 2235 psig 30 minutes post-trip. Pressurizer level decreased to a minimum of 18%

10 minutes post-trip, and reached 31% within 30 minutes post-trip, 6% from the no-load target of 25%. Steam pressure increased to a maximum of 1095 psig upon P-14 (and subsequent Turbine trip), decreased to a minimum of 922 psig, and then stabilized at 935 psig 30 minutes post-trip. S/G 2B, 2C, and 2D narrow range level indication remained on scale at all times following Reactor trip, and S/G 2A narrow range indication dropped off scale for approximately 5 minutes, but recovered to a value of 14% within 30 minutes post-trip (zero percent narrow range is well above the U-Tube bundle for the Unit 2 D-5 S/Gs).

At approximately 19% pressurizer leve , Operations swapped from the 75 gpm letdown orifice to the 45 gpm letdown orifice, and swapped Centrifugal Charging pump suction from the Volume Control Tank (VCT) to the Refueling Water Storage Tank. This action ensured recovery of VCT level and avoided a low pressurizer level letdown isolation (at 17% pressurizer level). Bank 1 steam dump to condenser valves opened to dump steam immediately post-trip. Due to a Loss of normal suction indication, Motor Driven CA Pump 2A suction swapped from the Upper Surge Tank to the assurred source of Auxiliary Feedwater, the RN System. When flow through the RN header increased to supply suction for CA Pump 2A, Asiatic clams were dislodged in the system and eventually clogged the cage assemblies of 2CA60 and 2CA56, S/G A and B Control valves, respectively.

Prior to Turbine Driven CA pump start, Motor Driven CA Pump 2A was supplying approximately 330 gpm each to S/Gs A and B. After the Turbine Driven CA pump start, CA flow to S/G B sharply increased to a maximum of 620 gpm, and then immediately degraded to 420 gpm due to the fouling by the clams. Approximately 2 minutes after the Motor Driven CA pump start, Operations throttled CA flow to S/G B to zero gpm. CA flow to S/G A was throttled to approximately 100 gpm within 2 minutes post-trip. Within 8 minutes post-trip, at which point the Operators had opened the S/G A CA Control valve to the full open position, the maximum attainable Auxiliary Feedwater flow to S/G A was 280 gpm. Upon reinitiating CA flow to S/G B, the maximum attainable flow rate through the clogged control valve was 120 gpm.

The Auxiliary Feedwater flow available to S/Gs C and D was 618 gpm and 314 gpm, respectively (the higher flow for S/G C was due to the operation of the Turbine Driven CA pump, which discharges to S/Gs B and C). The throttling of CA flow by the Operators several minutes post-trip was appropriate. Post-trip Reactor Coolant temperature trends with steam pressure, and steam pressure varies based on the presence of the quenching effect of the relatively cool auxiliary feedwater at the upper S/G nozzles. At the point the Operators throttled CA flow, the Reactor coolant temperature was trending below the no-load target of 557 degrees F. The Reactor Trip Emergency Procedure, EP/2/A/5000/01, requires that the wide range level indication in only one S/G be > 47% before allowing the operator to throttle CA flow. This condition was obviously met, and the Operators were successful at throttling CA flow to S/Gs B, C, and D (and thereby avoiding excessive cooldown), while also recovering S/G A level.

This event resembles and is bounded by the Turbine Trip transient as discussed in Section 15.2.3 of the Catawba FSAR. The CA flow available and maintained to S/Gs A and B for the first two minutes post-trip (prior to the Operators throttling CA flow) was within the design basis for all FSAR Chapter 15 transients, including Turbine Trip. Also, the flow to S/Gs C and D was within the design basis for all FSAR Chapter 15 scenarios. For Turbine Trip, no credit is assumed for auxiliary feedwater flow since a stabilized plant condition will be reached before CA initiation is assumed to occur. Furthermore, Motor Driven CA Pump 2B and the Turbine Driven CA pump remained operable and full flow from these pumps was available during this event if needed. Core decay heat was minimal due to the fact that the trip occurred at 21% Reactor power, at core Beginning Of Life during power escalation from the End of Cycle 1 refueling outage. Adequate core decay heat removal was available and maintained at all times. The Reactor coolant was 96 degrees F subcooled at the point of minimum NC System pressure.

The cooldown limits of 100 degrees F per hour for the Reactor Coolant System and 200 degrees F per hour for the pressurizer were not exceeded. Integrity of the fuel cladding, Reactor Coolant System, and Containment structure was maintained at all times.

Additional analysis for the degraded CA flow is provided in LER 413/88-15.

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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VICE PRESIDENT
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April 8, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-12

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-12 concerning a Reactor trip followed by an Auxiliary Feedwater Suction swap to the Nuclear Service Water System due to equipment failures. 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/10021/sbn

Attachment

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