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May 16, 1991

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

Subject: Catawba Nuclear Station
Docket No. 50-414
LER 414/91-06

Gentlemen:

Attached is Licensee Event Report 414/91-06, concerning HI-HI STEAM GENERATOR LEVEL (P-14) RESULTING IN A TURBINE TRIP AND AUXILIARY FEEDWATER START DUE TO INAPPROPRIATE ACTION.

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

Very truly yours,

J.W. Hampton

J. W. Hampton
Station Manager

ken:LER-NRC.JWH

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8-31-88

LICENSEE EVENT REPORT (LER)

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TITLE (4) HI HI STEAM GENERATOR LEVEL (P-14) RESULTING IN A TURBINE TRIP AND AUXILIARY FEEDWATER START DUE TO INAPPROPRIATE ACTION

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	1 6	9 1	0 1	0 0 6	0 0	0 5	1 6	9 1	N/A	0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1

POWER LEVEL (10): 2 5

	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
C. L. Hartzell, Compliance Manager	8 0 3 8 3 1 - 3 6 6 5

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
F	S J	V	B 3 5 0	N					

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☒ NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On April 16, 1991, at approximately 1452 hours, with Unit 2 in Mode 1, Power Operation, at approximately 25% Reactor Power, a Hi Hi Steam Generator (S/G) (P-14) signal was received due to '2C' Main Feedwater Control valve, 2CF46, not modulating to meet S/G 'C' demand. The P-14 signal tripped the Main Turbine and the Train 'B' Main Feedwater (CF) pump, resulting in an Auxiliary Feedwater (CA) autostart. The appropriate CF and CA valves started to close, but reopened as designed when the P-14 signal cleared. The Operator at the Controls (OATC) manually initiated a feedwater isolation and inserted rods until power decreased below Power Range indication. An investigation of the event revealed that the Main Feedwater By-pass Check Valves (2CF-166, 167, 168, 169) failed to seat properly. This allowed some of the CA flow to enter the S/Gs through the lower feedwater nozzles. This incident is attributed to an Inappropriate Action by the Instrument and Electrical (IAE) technician. Corrective actions include procedure revisions.

NRC Form 366 (9-83)

BACKGROUND

The Main Feedwater [EIIS:SJ] (CF) System supplies feedwater to the four Steam Generators [EIIS:HX] (S/Gs) at the temperature, pressure, and flow required to maintain proper S/G water levels commensurate with Reactor power output and Turbine [EIIS:TRB] steam requirements. The CF System contains two 50% capacity variable speed Turbine Driven CF Pumps [EIIS:P] (CFPTs). Each of the four CF lines contains a Feedwater Control valve [EIIS:V] (CF-28, 37, 46, 55), a CF Bypass Control valve (CF-30, 39, 48, 57), two CF Check valves, and a Feedwater Isolation valve (CF-33, 42, 51, 60). The CF Isolation valves function to terminate CF flow in either direction following a CF Isolation signal and also function to prevent or allow admission of feedwater to the S/Gs CF nozzles during various modes of operation. The CF Control valves are normally automatically controlled by the S/G Level Control system to maintain proper S/G levels.

The Auxiliary Feedwater [EIIS:BA] (CA) System assures sufficient feedwater to supply the S/Gs for decay heat removal in the event of loss of normal feedwater. The CA System is designed to automatically start two motor [EIIS:MO] driven CA pumps on the loss of both CF pumps, fully open the Flow Control Valves (2CA36, 40, 44, 48, 52, 56, 60, 64) to supply flow to the S/Gs, and isolate the Steam Generator Blowdown [EIIS:WI] (BB) System and the Nuclear Sampling [EIIS:KN] (NM) System valves associated with the four Steam Generators.

The CA System consists of two full capacity motor driven pumps that are capable of supplying feedwater to two steam generators. Initiating conditions are any one or combination of the following: 2 of 4 low-low level alarms in any 1 of 4 S/Gs, initiation of ATWS Mitigation and Actuation System Circuit (AMSAC), loss of all CF pumps, initiation of a safety injection (SS), or loss of offsite power. In addition, a turbine driven pump is normally aligned to supply feedwater to two S/Gs. The turbine driven pump will start automatically and provide the minimum required flow against a S/G pressure corresponding to the set pressure plus 3% accumulation of the lowest set main steam safety valve. This pump will start automatically on any one or both of the following conditions: 2 of 4 low-low level alarms in any 2 of 4 S/Gs or loss of offsite power.

An interlock is provided to ensure that CF flow is isolated to a S/G when an unacceptable Hi Hi S/G level setpoint is exceeded. The interlock (P-14 signal) provides a feedwater isolation signal tripping the CF pumps and closing the CF Control Valves (2CF28, 37, 46, 55), CF Bypass Control Valves (2CF30, 39, 48, 57), CF Containment Isolation Valves (2CF33, 42, 51, 60), CF Containment [EIIS:NH] Isolation Bypass Valves (2CF87, 88, 89, 90), the CF Tempering Flow Isolation Valves (2CA185, 186, 187, 188), and the CF Bypass to CA Nozzle Valves (2CA149, 150, 151, 152). In addition, upon a Main Feedwater isolation, two motor driven CA pumps automatically start as described above due to tripping of both CF pumps.

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

The Feedwater Isolation signal is provided to initiate isolation of each S/G to rapidly terminate feedwater flow and steam blowdown inside Containment following a Main Steam [EIIS:SB] (SM) System or Feedwater line break in Containment, to prevent loss of S/G water inventory due to a pipe [EIIS:PSP] rupture outside Containment, and to prevent overfilling the S/Gs if for some reason the normal means of controlling S/G level malfunctions. The CF Control Valves, CF Control Bypass Valves, and CF Isolation valves will all close on a Feedwater Isolation. CF Isolation is actuated by any one of the following:

1) Reactor Trip coincident with Low Tave,
2) P-14,
3) Safety Injection, or
4) Manual actuation

EVENT DESCRIPTION

On April 15, 1991, work request (WR) 482660PS was written to investigate and repair 2CF-46, S/G '2C' CF Control Valve, due to the demand position indicating low. On April 16, 1991, at approximately 1054 hours, with Unit 2 in Mode 1, Power Operation, at 100% Reactor Power Level, the Control Room Operator (CRO) started reducing Rx Power. At approximately 1452 hours, with Rx Power at about 25%, 2CF-46 S/G '2C' CF Control Valve, failed to modulate closed as the S/G feedwater demand decreased, thus causing the level in the S/G to increase. Hi Hi S/G level channels 3 and 4 went into alarm, resulting in CF Isolation (P-14) signal from S/G '2C'. CF pump 2B and the Unit 2 Main Turbine tripped at 14:53:01 hours; S/G 'C' Hi Hi Level went out of alarm approximately one second later. Again at 14:53:02 hours, all four Channels (1, 2, 3, 4) S/G 'C' Hi Hi Level went into alarm, resulting in a second P-14 signal. All four channels of the S/G 'C' Hi Hi Level signal subsequently cleared.

At 14:[illegible] hours, the required containment isolation valves closed on the BB and NM Systems in response to the CA autostart signal. Containment isolation valves (2CF-33, 42, 51, 60) went to 10% closed, CA motor driven pumps 'A' and 'B' started, and CF tempering flow isolation valves (2CA-185, 186, 187, 188) closed. At 14:53:20 hours, Bank 1 Condenser Steam Dump [EIIS:SO] (SB) System valves (2SB-9, 18, 27) opened; Bank 2 Condenser Steam Dump valves (2SB-6, 15, 24) did not open. Between 14:53 and 14:55 hours, CF Containment Isolation valves (2CF-33, 42, 51, 60) opened. At 14:55:29 hours, S/G 'A' power operated relief valve 2SV-19 opened, then closed 18 seconds later.

At approximately 14:55 hours, the CRO entered AP/2/A/5500/02, Turbine Generator Trip, and performed all required actions. At the same time the CRO noticed less than normal indicated flow of 100 gallons per minute to each S/G (A, B, C, D). Flow elements on the CA System indicated approximately 100 gpm flowing to each S/G. The Unit Supervisor then referred to AP/2/A/5500/06, Loss of S/G Feedwater, and noticed that the CF isolation signal was not present.

At 15:00:35 hours, after discussion between the Shift Supervisor and the Unit Supervisor, the CRO initiated a manual CF isolation and the OATC manually inserted rods to reduce power. Indicated CA flow achieved expected values after the manual CF isolation. Operations notified the NRC of the SF actuation to comply with RP/0/B/5000/13, NRC Notification Requirements.

On April 17, 1991, IAE technicians replaced the CF control valve positioner pilot valve assembly and performed a functional verification using the selector station; correct indications were obtained.

An investigation by Performance Engineers after the event, determined that approximately 200 gpm of CA flow went to each S/G (A, B, C, D) through unseated check valves (2CF166, 167, 168, 169) and the S/Gs lower feedwater nozzles.

On April 19-21, 1991, Maintenance repaired valves 2CF166, 167, 168, 169 under work request 004369MES (9102877901), 004370MES (9102878001), 004371MES (9102878201), and 004372MES (9102878301).

On April 23, 1991, IAE replaced Steam Dump Load Rejection Modulation PC Card and retested the system. The test results indicate the new PC Card functioned as designed.

CONCLUSION

The P-14 Engineered Safety Feature (ESF) actuation event is attributed to Inappropriate Action by the IAE technician. When installing the elastomer diaphragm with tight tolerances, the potential for damage increases. The IAE procedure was revised to caution the technicians of the potential damage that could occur during reassembly. MES will include steps in the SWRs to verify the integrity of the pilot valve assembly after maintenance. From evaluation of the valve assembly after the incident, it has been determined that work was performed with insufficient precision. The diaphragm in the pilot valve assembly on valve 2CF-46 was apparently damaged during earlier maintenance activities under W/R 010771SWR (dated 5/30/90).

During the event, the CROs did not understand why the feedwater isolation did not actuate. This is attributed to the lack of training in that the P-14 signal is non-latching and is designed to automatically reset if a reactor trip signal is not present. P-14 is not truly a Reactor Protection function. Its purpose is to prevent water induction in the steam pipes and turbine. Production Support Division, Operations Training, changed lesson plans, OP-CN-ECCS-ISE and OP-CN-CF-CF, to include additional information about the P-14 signal.

When the turbine tripped, Condenser Steam Dump Bank '1' opened to provide a flow path for the steam release. In addition, Bank '2' Condenser Steam Dumps should have been modulating at approximately 40% open. The Bank '2' valves

did not open because a load rejection modulation PC card provided no output. A failed load rejection modulation card was replaced with a new PC card. The system was tested and operated properly.

During the event, the Main Feedwater By-pass Check valves, 2CF-166, 167, 168, 169, failed to properly seat (PIR # 2-C91-0160 was written). This allowed CA to flow in the reverse direction through the check valves into the CF line and the S/Gs lower feedwater nozzle. Evaluation concluded that throughout the transient the required total expected flow was delivered to the S/Gs.

On August 31, 1989, the NRC had issued an Information Notice 89-62. This notice was intended to alert utilities of the potential malfunctioning of Borg-Warner pressure seal bonnet check valves. The CF By-pass valves (2CF-166, 167, 168, 169) are Borg-Warner Pressure Seal Bonnet Check valves. On September 27, 1989, Mechanical Maintenance revised procedure MP/0/A/7600/37, Borg-Warner Pressure Seal Swing Check Valve, to include a caution concerning the removal and reinstallation of the disc. During routine acoustic emissions monitoring, audible tapping against the valve backstops on CF By-pass valves (2CF166, 167) was identified. A later evaluation identified "tapping" on all four check valves (2CF-166, 167, 168, 169). Mechanical Maintenance made necessary repairs under work requests 002673MES, 002674MES, 002676MES, and 002675MES during the last Unit 2 refueling outage.

On May 2, 1990, the Superintendent of Maintenance approved Station Problem Report (SPR) CNPR-04892. The problem report resolution was to replace existing Borg-Warner check valves 2CF166, 167, 168, 169 with an Anchor Darling tilting disc check valve. Inspection of 2CF-166, 167, 168, 169 after this event revealed that their failure to reseat was due to wear of the disc hinge pin, and not due to improper alignment of the valve internals as described in the NRC Notice.

An additional concern with the leaking check valves is that they are an essential boundary to prevent water loss if an Safe Shutdown Facility (SSF) Security event were to occur, since no credit can be taken for piping outside the vital areas. After this event, Operations issued Technical Mamorandum 21-11 indicating steps to be taken to ensure a flow path to the S/Gs on an SSF Security event. Security developed and implemented compensatory measures. The compensatory measures are being forwarded to the Office of Nuclear Reactor Regulation (NRR) section in a letter dated May 16, 1991.

A search of the Operating Experience Program showed that no events have occurred over the past two years as a result of a CF control valve positioner diaphragm failure. This is therefore not considered to be recurring.

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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

CORRECTIVE ACTION

SUBSEQUENT

1) CROs referred to AP/2/A/5500/02, Turbine Generator Trip procedure, for guidance and performed the required actions.

2) Unit Supervisor referred to AP/2/A/5500/06, Loss of S/G Feedwater, after manually initiating CF feedwater isolation CA flow achieved expected values and the procedure was exited.

3) OATC drove rods to reduce reactor power.

4) Operations notified the NRC of the CF isolation (ESF) actuation to comply with RP/0/B/5000/13, NRC Notification Requirements.

5) IAE replaced the damaged diaphragm in valve positioner pilot valve assembly for 2CF-46 under work request 48266OPS (9102836101).

6) IAE revised IP/0/B/3820/30 to include steps to preclude damage to the diaphragm when installing the pilot valve assembly.

7) Production Support Division, Operations Training, changed training lesson plans, OP-CN-ECCS-ISE and OP-CN-CF-CF, to include information about CF isolation valves that will reopen if not fully closed on a momentary P-14 signal if a reactor trip is not present.

8) Mechanical Maintenance repaired valves (2CF166, 167, 168, 169) to reduce back leakage. [004369MES (9102877901), 004370MES (9102878001), 004371MES (9102878201), 004372MES (9102878301)]

9) Operations will trend, on S/G CF control valves, the demand position indication against actual valve position. This information will be provided to MES for evaluation to anticipate possible equipment failure.

PLANNED

1) Station Problem Report # CNPR-04892 was activated. The proposed resolution is to replace Borg Warner check valves 2CF166, 167, 168, 169 with an Anchor Darling tilting disc check valve.

2) MES will include steps in SWRs to test the pilot valve assembly on all control valves after routine maintenance.

NRC FORM 366A (9-83)

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

SAFETY ANALYSIS

The response to the S/G 2C Hi Hi level was as designed, with the exception of CF By-pass Check valves not sealing properly and Bank '2' of the Condenser Steam Dumps not opening.

The failure of the CF By-pass Check valves and subsequent S/Gs temperature drop of 200°F in 2 minutes had no impact. This event is enveloped by the normal condition transient of Large Step Load Decrease with Steam Dump. However, the worst case scenario would be where the thermal stress resulted in brittle fracture of the CF nozzle. This would result in an unisolable break in the secondary side of the S/G. This type accident is bounded by FSAR 15.2.8. The potential NC System cooldown (by excessive energy discharge through the break) is bounded by FSAR 15.1.5. The potential NC System heatup (because of the loss of feedwater and thus heat removal capability) is bounded by FSAR 15.2.8. Thus, this event is bounded by the accident analysis in the FSAR.

The S/G safety relief and power-operated relief valves provide the necessary flow path to remove excessive energy (steam). Due to the additional relief protection, the failure of Bank '2' Condenser Steam Dumps to operate had no effect.

The health and safety of the public were unaffected by this event due to the appropriate response by the ESF actuation system as designed.