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 1 1 |

TITLE (4)
Reactor Trip on Steam Generator 2C Lo-Lo Level Due to a Management Deficiency

| 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 2 | 2 1 | 8 9 | 8 9 | — 0 0 3 — | 0 1 | 0 4 | 1 8 | 8 9 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): 1
POWER LEVEL (10): 0 9 4

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) | 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) | X 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) | 50.72(b)(2,(ii) |
| 20.405(a)(1)(iv) | 50.73(a)(2)(ii) | 50.73(a)(2)(viii)(B) | 50.72(b)(i)(iv) |
| 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 | |
| R. L. White, Chairman, Catawba Safety Review Group | 8 0 3 | 8 3 1 - 3 3 9 3 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| D | S B | X B | E 1 6 9 | N | X | C B | X B | E 1 6 9 | N |
| X | C B | X B | E 1 6 9 | N | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| YES (If yes, complete EXPECTED SUBMISSION DATE) | NO | | | | |

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

On February 21, 1989, at 0112 hours, IAE Technicians placed a jumper to check relay contacts while investigating the failure of the "90 Percent Open" light to illuminate during the Main Steam Isolation Valve Movement Test. This jumper resulted in a short circuit, closing 2SM3, Steam Generator (S/G) 2C Main Steam Isolation Valve. The resulting pressure increase in S/G 2C caused a Reactor Trip on S/G Lo Lo Level, and caused 2SV7, S/G 2C PORV, and three S/G 2C Code Safety Relief Valves to open. The opening of these valves resulted in a rapid decrease in steam line pressure, which resulted in a Safety Injection on low steam line pressure (rate compensated). This incident is classified as management deficiency due to the lack of a policy for an independent assessment of actions to be taken under troubleshooting procedures, with contributing causes of equipment malfunction due to the opening setpoint for 2SV7 being found to be too high, and lack of attention to detail prior to jumper placement. Corrective actions will include providing a troubleshooting review guideline. The open setpoint for 2SV7 was recalibrated. Unit 2 was in Mode 1, Power Operation, at 94% power at the time of this incident.

IE22
1/1

BACKGROUND

The Main Steam (EIIS:SB) Isolation Valve (EIIS:V) Movement Test, PT/1,2/A/4250/01A, is performed quarterly to verify the operability of the Main Steam Isolation Valves (MSIVs). The procedure is to depress and hold the Test pushbutton for each MSIV until the "90 Percent Open" light illuminates, then to release the Test pushbutton. The test circuit is designed so that the MSIV travels from full open to 90 percent open, then back to the full open position.

The purpose of the Main Steam (SM) System is to transfer steam generated in the Steam Generators (S/Gs) to Turbine building loads. The MSIVs function to limit steam flow from the Steam Generators (EIIS:SG), and close if a break occurs in the steam piping (EIIS:PSP). A Main Steam Isolation signal will occur upon low steam line pressure.

A low steam line pressure signal is initiated by either low steam line pressure (725 psig) when Reactor Coolant (EIIS:AB) (NC) System pressure is greater than 1955 psig, or from high steam line pressure rate (100 psi/sec) when NC System pressure is less than 1955 psig (the P-11 setpoint). A low steam line pressure signal initiates a Safety Injection (S/I). The bistable for low steam line pressure is rate sensitive, so that a rapid drop in pressure can initiate a Safety Injection above as well as below the P-11 setpoint.

One Power Operated Relief Valve (PORV) is provided per S/G for overpressure protection. The S/G PORVs are set to automatically open at 1125 psig, and reclose at 1092 psig. Five safety valves are provided for each steam line. The lift setpoint of each S/G safety valve is set according to the values specified in Table 3.7.2 of Technical Specification 3.7.1.1. Technical Specifications do not specify a pressure at which the S/G safety valves are to reseat.

Technical Specification Surveillance Requirement 4.8.1.1.2.g.13 specifies that the Diesel Generator (EIIS:ENG) (D/G) accelerated sequence permissives are to be verified to have a minimum time delay of 2 +/- 0.2 seconds, every 18 months to verify D/G operability. A maximum allowable time delay is not specified (as it is for the committed sequence timers).

Technical Specification 3.0.3 requires that if the Unit is in a situation such that a Limiting Condition for Operation cannot be met except as stated in action requirements, then action must be taken within one hour to place the Unit in a Mode in which the specification does not apply.

EVENT DESCRIPTION

On February 20, 1989, at 1145 hours, Control Room Operators (CROs) began performing the Main Steam Isolation Valve Movement Test, PT/2/A/4250/01A. When testing 2SM3, S/G 2C Main Steam Isolation Valve (MSIV), the "90 Percent Open" light (EIIS:XI) did not illuminate when the Test pushbutton was depressed, as required by the procedure. At 1300 hours, the Operator at the Controls (OATC) initiated Work Request 42768 OPS to investigate the cause of the "90 Percent Open" light for 2SM3 not illuminating. Subsequently, IAE Technicians began to check the test circuit for 2SM3 (under the Controlling Procedure for Troubleshooting and Corrective Maintenance, IP/0/A/3890/01) by checking the light bulb, then placing a jumper from F-10 to F-12 in cabinet 2SMTC1 to verify that contacts coming off a 2SM3 limit switch were not causing the problem. As each contact was checked in the test circuit path, the IAE Technicians instructed a CRO to depress the Test pushbutton for 2SM3. In order to verify the operability of relay 2SMAR10 contacts 1 and 1a in cabinet 2SMTC1, at approximately 0111 hours on February 21, 1989, IAE Technician A placed a jumper from terminal F-11 to F-12 in 2SMTC1, and instructed a CRO to depress and hold the Test pushbutton for 2SM3 (this jumper was verified by IAE Technician B). This resulted in a short circuit across 2KXPA Breaker (EIIS:BRK) No. 16 (which feeds the test circuits for the MSIVs), since 2SMAR10 contacts 1 and 1a were closed (as expected). The short circuit did not trip the breaker, but caused voltage fluctuations in the MSIV test circuits, damaging the optical isolator (EIIS:XB) (2SMID11) which provides the output to relay (EIIS:RLY) 2SMAR10A (which de-energizes solenoid 2SMSV0030 to bleed air off the air cylinder (EIIS:XCV) for 2SM3, closing the valve). Optical isolator 2SMID11 failed in the position which continued to provide a constant valve closure output, regardless of its input. The OATC noted that all of the "Open" indicating lights for the MSIVs were flickering, notified the IAE Technicians, and released the Test pushbutton for 2SM3. Between 0111:27-38 hours, 2SM3 continued to the fully closed position (the other three MSIVs did not close at this time). Between 0111:30-49 hours, S/G 2C steam pressure increased from 1011 to approximately 1200 psig. At 0111:44 hours, S/G 2C Code Safety Relief Valves 2SV8, 2SV9 and 2SV10 opened to control steam pressure. S/G 2C narrow range levels rapidly decreased due to the increasing steam pressure. At 0111:44:199 hours, a Reactor trip occurred on Lo Lo Level in S/G 2C, and Auxiliary Feedwater (EIIS:BA) (CA) Motor Driven Pumps (EIIS:P) A and B autostarted on S/G 2C Lo Lo Level. At 0111:44:473 hours, the Main Turbine tripped on Reactor Trip. At 0111:46 hours, 2SV7, S/G 2C PORV, opened to control steam pressure. At 0111:48:369 hours, the CROs manually tripped the Reactor. By 0111:49 hours, S/G 2C steam pressure had reached a maximum of approximately 1215 psig, after which it rapidly decreased (2SV7, 8, 9 and 10 were open). A pressure drop of approximately 68 psi occurred in 4 seconds, satisfying the Low Steamline Pressure signal lead/lag logic. This resulted in a Safety Injection on Low Steamline Pressure in Loop C at 0111:52:703 hours. A Main Steam Isolation also occurred on Low Steamline Pressure, and a Feedwater Isolation was initiated on S/I. Main Feedwater (EIIS:SJ) (CF) Pump 2B tripped on S/I at 0111:52:727 hours.

At 0111:52:739 hours, Diesel Generator (D/G) Load Sequencer A actuated on S/I (accelerated sequence), and a 0111:52:743 hours, D/G Load Sequencer B actuated on S/I. At 0111:52:745 hours, CF Pump 2A tripped on S/I.

Subsequently, Trains A and B Load Group 1 energized, actuating Containment Isolation (EIIS:JM) Phase A motor operated valves and dampers, after which Trains A and B Load Group 2 energized, starting Centrifugal Charging Pumps A and B as expected. At 0111:55 hours, 2SV 7, 8, 9 and 10 closed, and at 0111:56:103 hours, the CA Turbine Driven Pump (CAPT), auto started on S/Gs A and C Lo Lo Level. Trains A and B Load Group 3 energized, starting Safety Injection (EIIS:BQ) (NI) Pumps A and B, and subsequently Trains A and B Load Group 4 energized, starting Residual Heat Removal (EIIS:BP) (ND) Pumps A and B. Load Group 5 energized on both trains, after which Trains A and B Load Group 6 energized, starting Component Cooling (EIIS:CC) (KC) Pumps B1 and B2. Load Group 7 energized for both trains, starting Low Pressure Service Water (EIIS:BI) (RN) Pump 2B, as expected. The remaining Load Groups (8, 11, 12 and 13) energized in the expected sequence.

At approximately 0112 hours, the Reactor Trip or Safety Injection Emergency Procedure, EP/2/A/5000/001, was entered and at 0120:28 hours, the CROs tripped the CAPT. At 0121:37 hours, the CROs reset A and B Train S/I, and at 0121:44 hours, both D/G Sequencers were reset. At 0122 hours, the CROs tripped ND Pumps A and B, NI Pumps A and B and subsequently both Centrifugal Charging Pumps. At 0124 hours, they reset both trains Phase A Containment Isolation. At 0126 hours, the CROs realigned Containment Valve Injection Water (NW) valves. At 0127 hours, EP/2/A/5000/001 was exited and the S/I Termination Following Spurious S/I Procedure, EP/2/A/5000/001B, was entered. At 0128 hours, CROs adjusted CA flow control valves to throttle CA flows and at approximately 0200 hours realigned S/G Blowdown (BB) valves. By approximately 0230 hours, the Unit had been returned to the normal Mode 3 alignment.

At approximately 0430 hours, the CROs entered the Unit Fast Recovery Procedure, OP/2/A/6100/05. Based upon a review of Event Recorder data, Performance submitted Work Request 6877 PRF to calibrate Train A D/G accelerated sequence timers for Load Groups 3, 5 and 6 (calculated times were 2.250, 1.174 and 2.962 seconds, respectively), and Work Request 6878 PRF to calibrate Train B D/G accelerated sequence timers for Load Groups 5, 7, 8 and 11 (calculated times were 2.202, 2.362, 2.224 and 2.222 seconds, respectively). Performance submitted two Technical Specification Operability Notification Sheets declaring D/G 2A and 2B inoperable, for accelerated sequence times not being within 2.0 +/- 0.2 seconds, at 0850 hours. At 0930 hours, Technical Specification 3.0.3 was entered. IAE Technicians subsequently began checking the specified accelerated sequence timers using the 2A(B) Diesel Load Sequencer Timers Calibration (EQB) Procedures, IP/2/A/3670/01A(B). As required by Technical Specification 3.0.3, at 1030 hours, the CROs began Reactor Coolant (NC) System cooldown toward Mode 5, Cold Shutdown, at 10 degrees per hour. During the performance of IP/2/A/3670/01A(B), IAE Technicians recorded as found times which were out of tolerance (high) for Train B Load Groups 7 and 8 (2.35 and 2.23 seconds, respectively) and Train A Load Group 3 (2.23 seconds). The remaining timers were found to be within tolerance: Train A Load Groups 5 and 6 (1.96 and 2.14 seconds, respectively), and Train B Load Groups 5 and 11 (2.20 and 2.09 seconds, respectively).

Based upon a review of D/G sequencer timer data by Maintenance Engineering Services (MES) and Performance, it was determined that the As Found times recorded on IP/2/A/3670/01A(B) were correct and that the Event Recorder times were erroneous. Design Engineering (DE) indicated that times exceeding 2.2 seconds did not render the sequencer inoperable, since the loads would still be picked up on the committed sequence timers (which are independent of the accelerated sequence timers). DE also indicated that D/G operation would not be affected unless loads were sequenced in extremely short (0.1 to 0.5 second) times. Compliance determined that the D/G sequencers were not inoperable, since the As Found times were not too fast, and since DE determined that long accelerated sequence times did not render the D/G sequencers inoperable.

IAE Technicians replaced the damaged optical isolator (2SMID11 in 2SMTC1) under the Field Checkout and Replacement of Digital Optical Isolators procedure, IP/0/A/3840/03B. Further investigation of the 2SM3 test circuit resulted in the replacement of a defective socket for the "90 Percent Open" light, under IP/0/A/3890/01. Following these repairs, the MSIV Movement Test was successfully completed for 2SM3 on February 21, 1989, at 1830 hours.

It was noted during review of the alarm typer that 2SA2, S/G 2B SM to CAPT, indicated excessive cycling following the Reactor trip. Performance submitted Work Request 6881 PRF at 1000 hours on February 21, 1989, and it was subsequently found that a crimped cover gasket had allowed dust and dirt to collect in limit switch 2SALL0021. The limit switch was then cleaned and returned to service following functional verification. It was also noted that the Reciprocating Charging Pump (which was in use at the time the S/I occurred) did not trip as expected following the S/I signal. Performance submitted Work Request 6882 PRF at 1600 hours on February 21, 1989, and it was found that optical isolators J082 in 2EATC12 and J082 in 2EATC13 were defective. Both isolators were replaced under IP/0/A/3840/03B and the auto trip circuitry was verified for both Trains A and B before the pump was returned to service on February 22, 1989.

CONCLUSION

This incident is classified as Management Deficiency, due to the lack of a policy for an independent assessment of actions to be performed under troubleshooting procedures. The policy of independent verification needs to be extended to include the independent review of plant documentation before an action is taken under a troubleshooting procedure.

This incident is classified a contributing cause of inappropriate action. The IAE Technicians placing the jumper from F-11 to F-12 in 2SMTC1 did not consider the possibility that when the Test pushbutton for 2SM3 was depressed and held, a short circuit path would be created if relay 2SMAR10 contacts 1 and 1A were operating properly (closed). This incident has been discussed with the technicians involved.

This incident is also classified a contributing cause of Equipment Malfunction. Transient Monitor data indicated that 2SV7, S/G 2C PORV, opened at approximately 1200 rather than at 1125 psig. Work Request 6885 PRF was initiated at 1400 hours on February 22, 1989, and IAE checked the pressure switch calibration for 2SV7 under the System Calibration Procedure Steam Generator Steam Line Pressure Control S/G C Procedure, IP/2/A/3030/14C. The As Found setpoint for 2SMPS5500 was out of tolerance (high), and would generate an open signal at approximately 1153 psig rather than the desired 1125 psig. The calibrated setpoint for this pressure switch had apparently drifted since the previous calibration. The pressure switch was recalibrated, a functional verification was performed, and 2SV7 was returned to service on February 22, 1989. This pressure switch is manufactured by Custom Control, Model No. 6862G1.

It cannot be determined with certainty that the rapid steam pressure drop in S/G 2C was entirely due to the late opening of 2SV7. Transient Monitor data indicated that S/G 2C Code Safety Relief Valves 2SV8, 9 and 10 lifted at approximately 1200 psig steam pressure. However, position indication for the S/G safeties is provided by downstream pressure switches rather than limit switches, so that the Transient Monitor data may not be accurate in determining the exact position of 2SV8, 9, or 10. The Technical Specification Table 3.7-2 settings for these valves are 1175, 1190 and 1205 psig respectively. If 2SV8 lifted at 1200 psig rather than within the 1166 to 1184 psig acceptance band, its late response could have contributed to this incident The Main Steam Safety Valve Setpoint Test, MP/0/A/7150/072, performed on February 23, 1988 indicated that 2SV8 lifted consistently within the tolerance band following adjustment, averaging 1178.2 psig. This test data was the basis for verifying the satisfactory performance of 2SV8 during the transient, since the indications provided by the pressure switches are not as accurate. Work Request 6890 PRF was initiated on February 27, 1989 to record the lift setpoints for 2SV8 when performing MP/0/A/7150/072 during the Unit 2 EOC2 Refueling Outage. This will help to determine the pressure switch response time in comparison to actual valve lift times. The exact pressure at which the S/G safeties (2SV8, 9, 10) closed during this transient also cannot be determined. The reseat pressure is not specified by Technical Specifications, and therefore not checked on a periodic basis. The combined effect of open safety valves and being at a high steam pressure (~1215 psig) caused a large steam flow, resulting in a rapid decrease in pressure.

The initial corrective action was the release of the Test pushbutton for 2SM3 by the OATC. Subsequent corrective actions included performance of required EPs by the CROs (EP/2/A/5000/001 and 001B), and IAE performed Work Requests 42768 OPS on February 22, 1989 (replacement of the damaged optical isolator and defective "90 Percent Open" light socket), 6877 and 6878 PRF (calibration of A and B Train D/G sequencer timers on February 21, 1989), 6881 PRF (cleaning of 2SA2 limit switch on February 22, 1989), 6882 PRF (replacement of defective optical isolators and verification that the Reciprocating Charging Pump will trip following S/I, on February 22, 1989), and 6885 PRF (recalibration of the pressure switch to ensure that 2SV7 opens at the proper setpoint, and the

"Open/Not Open" computer point was verified). This incident has been discussed with the individuals involved, and the importance of considering the full effects when placing jumpers was emphasized to all IAE Technicians.

Instrument and Electrical (IAE) will revise IP/0/A/3890/01 to provide for independent review of documentation before performing actions, and provide a guideline to identify the need for this review in other troubleshooting activities. Compliance will issue an interpretation for Technical Specification Surveillance Requirement 4.8.1.1.2.g.13 to clarify that the D/G and sequencer are not inoperable when times are greater than or equal to 2.2 seconds (although the timers should be promptly recalibrated). See LER No. 414/89-04 for corrective actions regarding the Event Recorder. MES will check the As Found lift setpoint of 2SV8 during the Unit 2 EOC2 REfueling Outage.

A review of previous incidents showed that during the twelve months preceding this incident, no ESF actuations were due to an inadequate policy. This is therefore not considered to be a recurring event.

A review of previous incidents during the past three years showed that Safety Injections are a recurring problem at Catawba. LER 414/88-003 describes an S/I caused by not following Station Directives regarding tagouts. LER 414/86-049 describes a spurious S/I caused by dirty contacts in the Westinghouse 7300 Process Control Cabinet lead/lag card (EIIS:IMOD). LER 414/86-041 describes an S/I due to the malfunction of an SM Bypass to Condenser (SB) valve. LER 414/86-028 describes an S/I caused by S/G PORV controls and procedural problems. The corrective actions taken for these incidents could not have prevent this event.

This is the fifth actuation of S/I to date. The nozzle usage factor does not exceed 0.70.

A review of the Industry Operating Experience Program did not provide any information which aided in the resolution of this LER. Numerous instances of safety valve problems were listed, some of which involved safety valve setpoint drift.

CORRECTIVE ACTION

IMMEDIATE

1) The OATC released the Test pushbutton for 2SM3 upon seeing the indicating lights for all four MSIVs flicker.

SUBSEQUENT

1) CROs performed actions required by Reactor Trip or Safety Injection procedure, EP/2/A/5000/001 to stabilize the Unit.

2) CROs performed actions required by S/I Termination Following Spurious S/I, EP/2/A/5000/001B to return the Unit to normal alignment.

3) Incident was discussed with the technicians involved, and the importance of considering the full effects when placing jumpers was emphasized to all IAE Technicians.

4) IAE replaced damaged optical isolator and defective "90 Percent Open" light socket under Work Request 42768 OPS.

5) IAE calibrated Trains A and B D/G accelerated sequencer timers specified on Work Requests 6877 and 6878 PRF, as needed.

6) IAE cleaned limit switch for 2SA2 under Work Request 6881 PRF.

7) IAE replaced defective optical isolators and verified that Reciprocating Charging Pump will trip following S/I under Work Request 6882 PRF.

8) IAE recalibrated pressure switch for 2SV7 position under Work Request 6885 PRF.

PLANNED

1) A Technical Specification Interpretation for Technical Specification Surveillance Requirement 4.8.1.1.2.g.13 will be issued to clarify D/G and sequencer operability concerns in light of accelerated sequence times.

2) The As Found lift setpoint for 2SV8 will be recorded under Work Request 6890 PRF.

3) The Controlling Procedure for Troubleshooting and Corrective Maintenance, IP/0/A/3890/01, will be revised to include independent review of documentation before an action is taken.

4) A guideline will be developed to identify those troubleshooting activities in which an independent assessment of actions to be taken must be performed.

5) The responsibility for resolution of all inadequate post-trip responses will be assigned and it will be ensured that all such items are identified on the Station Commitment Index.

SAFETY ANALYSIS

Upon inadvertent closure of the S/G C Main Steam Isolation Valve (MSIV), 2SM3, steam pressure immediately increased and caused a S/G 2C level shrink to the low-low level setpoint. A Reactor Trip signal occurred upon S/G 2C low-low level, and the Reactor Trip Breakers tripped within 76 milliseconds of the Reactor Trip signal. Unit 2 was operating at 94% full power at the time of the trip. All of the control rods fell to the bottom of the core, reducing power to decay heat level. The Motor Driven CA pumps autostarted upon S/G 2C low-low level. The S/G 2C PORV, 2SV7, and three Code Safety Relief valves opened to release steam to the atmosphere. This caused a rapid steam pressure decrease, and a S/I signal occurred upon rate compensated steam line pressure. The S/I signal initiated a CF Pump Turbine Trip and a CF Isolation. Also upon S/I signal, the Diesel Generators automatically started and the load sequencers successfully completed accelerated sequence loading. The S/I signal initiated closure of the S/G 2A, 2B, and 2D MSIVs. The S/G 2A, 2B, and 2D MSIVs reached the fully closed position within 3 seconds from the S/I signal. All appropriate major ECCS equipment was actuated upon S/I, including the Centrifugal Charging Pumps, the Safety Injection Pumps, and the Residual Heat Removal Pumps. S/G 2A Low-Low Level signal occurred approximately 12 seconds after S/G 2C Low-Low Level signal, thus satisfying the two-out-of-four logic for autostart of the Turbine Driven CA pump. The redundant steam supply valves for the Turbine Driven CA Pump, 2SA2 and 2SA5, opened within 3 and 5 seconds, respectively, of the Autostart signal. The Operator initiated manual Reactor trip within 4 seconds of the automatic trip.

Upon 2SM3 closure and the subsequent S/G 2C steam pressure increase, Reactor Coolant Loop 2C temperature followed steam pressure and increased 5 degrees F to a maximum value of 593 degrees F. This increase was momentary, and temperature decreased with steam pressure decrease upon opening of the S/G 2C PORV and Code Safety Relief valves. After Reactor trip, reactor coolant temperature in all four loops decreased to a minimum value of 545 degrees F, and then stabilized at 556 degrees F within 40 minutes post-trip, 1 degree F from the no-load target of 557 degrees F. Upon 2SM3 closure and subsequent S/G 2C pressure increase, Reactor Coolant pressure increased from 2250 psig to a maximum value of 2266 psig. This increase was momentary, and pressure then decreased to a minimum value of 2010 psig following the Reactor trip. Within 30 minutes post-trip, Reactor Coolant System pressure stabilized at 2250 psig, 15 psi from the no-load target of 2235 psig. Upon Reactor trip, pressurizer level immediately decreased to a minimum value of 33%, increased to a maximum value of 83% 23 minutes post-trip, and was at a value of 78% and decreasing 40 minutes post-trip. Upon closure of 2SM3, S/G 2C steam pressure increased to a maximum value of 1214 psig, and immediately decreased upon opening of the S/G 2C PORV and Code Safety Relief valves. S/Gs 2A, 2B, and 2D steam pressure increased to an average maximum value of 1093 psig upon Reactor trip. Steam pressure decreased to an average minimum value of 960 psig after Reactor trip, and stabilized at an average value of 1070 psig 40 minutes post-trip. S/Gs 2A, 2B, 2C, and 2D reached a minimum wide range indicated value of 48%, 47%, 42%, and 46%, respectively. Correction of these values for calibration condition variations yields actual wide range minimum levels of 63%, 62%, 54%, and 61% for S/Gs 2A, 2B, 2C, and 2D, respectively.

11 to 12 minutes post-trip the Main Steam Isolation Bypass Valves were opened to allow steam dump to the Condenser and thus provide long term core decay heat removal. An adequate supply of Auxiliary Feedwater flow was available to remove decay heat. As required by the Reactor Trip or Safety Injection Emergency Procedure, the Operators maintained auxiliary feedwater flow above the minimum required cummulative value of 450 gpm to the S/Gs while the S/G indicated wide range level was less than 47%. Per design, valves NV252A and 253B automatically opened upon S/I signal to swap Centrifugal Charging Pump suction from the Volume Control Tank to the Fueling Water Storage Tank. Also per design, valves NI9A and NI10B automatically opened to allow Centrifugal Charging Pump injection flow directly into the Reactor Coolant System cold legs. The Reactor Coolant System boron concentration was 51 ppm prior to S/I. During safety injection, a total of approximately 3000 gallons of water, at a boron concentration of 2000 ppm, was injected into the Reactor Coolant System. The Safety Injection Pumps did not inject water into the Reactor Coolant System, as Reactor Coolant System pressure did not decrease below S/I Pump dead head during the transient. Reactor Coolant was 32 degrees F subcooled at the point of minimum Reactor Coolant System pressure. Adequate heat sink for core decay heat removal was available and maintained at all times.

FSAR Section 15.2.4 discusses inadvertent closure of Main Steam Isolation Valves, stating that an inadvertent closure of an MSIV would result in a Turbine trip. An MSIV closure is therefore bounded by the Turbine Trip scenario as discussed in the Catawba FSAR, Section 15.2.3. The following comparison shows that this event is fully bounded by the Turbine Trip scenario:

* The Turbine Trip scenario is assumed to occur from 100% full power. This event occurred at 94% full power.

* Turbine Trip scenario overpressure protection is provided by the pressurizer PORVs and the S/G PORVs. In this event, the S/G PORV was available and the pressurizer PORVs were not required to be used.

* The Turbine Trip scenario assumes that Reactor trip occurs at the first RPS setpoint reached. In this event, Reactor trip occurred on S/G low-low level, but the low-low level did not correspond to an actual S/G inventory reduction. Rather, S/G low-low level occurred due to an inventory shrink due to steam pressure increase upon MSIV closure.

* The initiation of S/I further ensured adequate Reactor Coolant System shutdown margin, adequate Reactor Coolant System inventory, and DNBR well above the minimum allowable value.

The Reciprocating Charging Pump did not automatically trip upon S/I as designed. However, this is not a safety related interlock, and was not specified by Westinghouse as a required function. The continued operation of the pump did not affect the ability of the Centrifugal Charging Pumps to deliver flow to the Reactor Coolant System. The Reciprocating Charging Pump was manually tripped by the Operator, and then later restarted to supply Reactor Coolant Pump seal injection flow when operation of the Centrifugal Charging Pumps was terminated.

All plant safety equipment was available throughout this event. The value of the usage factor for the cold leg injection nozzle does not exceed 0.70. Unit 2 has experienced four previous Safety Injections. Therefore, this event constitutes the fifth actuation cycle to date. 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. The health and safety of the public were not affected by this event.

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April 18, 1989

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/89-03, Revision 1

Gentlemen:

My March 23, 1989 letter to the Document Control Desk transmitted Catawba Nuclear Station Unit 2 Licensee Event Report No. 414/89-03 concerning a reactor trip on Steam Generator 2C low-low level due to a management error. Please be advised that the March 22, 1989 submittal identified this Unit 2 report as a Unit 1 event. Additionally, in the March 23, 1989 submittal this event was assigned Docket No. 413 when it should be have been assigned Docket No. 414.

Therefore, pursuant to 10 CFR 50.73 please find attached Revision 1 to Licensee Event Report 414/89-03. The Unit and Docket Nos. have been revised to identify this Licensee Event Report as a Unit 2 event. 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
for
H. B. Tucker

JGT/3/PIR91R1

Attachment

xc: S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
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Mr. W. T. Orders
NRC Resident Inspector
Catawba Nuclear Station

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