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April 22, 1991

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

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/91-06

Gentlemen:

Attached is Licensee Event Report 413/91-06, concerning TECHNICAL SPECIFICATION VIOLATION WHEN NUCLEAR SERVICE WATER VALVES WERE LEFT WITHOUT AN EMERGENCY POWER SUPPLY 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

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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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	1 OF 0 9

TITLE (4) Technical Specification Violation When Nuclear Service Water Valves Were Left Without An Emergency Power Supply 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 3	2 3	9 1	9 1	0 0 6	0 0	0 4	1 8	9 1	CNS, Unit 2	0 5 0 0 0 4 1 4
										0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 1 0 0

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)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 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	
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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

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

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

On March 23, 1991 at 0300 hours, Unit 1 was in Mode 5, Cold Shutdown, in preparation for the End of Cycle refueling outage. Unit 2 was in Mode 1, Power Operations. At 0337 hours, Operations removed the 1A Diesel Generator (D/G) from service using procedure OP/1/A/6350/02, Diesel Generator Operation. A Non-Licensed Operator (NLO) requested Operator at the Controls' (OATC) assistance to ensure that 1EMXG, Essential Motor Control Center, was aligned to 2ELXA, Alternate Blackout Power Loadcenter. The OATC and NLO used the Operator Aid Computer (OAC) graphics and apparently misread the graphic to conclude that 1EMXG was powered by 2ELXA. The Nuclear Service Water System (RN) 'A' Train was removed from service on March 23 at 0500 hours and was returned to service on March 26 at 0445 hours. At 1400 hours, an Engineered Safeguard Feature Bypass Panel (1.47) alarm was received in the Control Room and the operators determined that 1EMXG was without emergency power. The operators aligned 1EMXG to 2ELXA. This incident has been attributed to an Inappropriate Action in that the NLO and OATC misread the OAC graphics while verifying the alignment of 1EMXG to 2ELXA. OPS personnel have been informed not to use the OAC graphics for procedure sign-offs and OPS Management Procedures will be revised. OAC graphics training will be provided along with an evaluation of OAC graphics accuracy. A clear policy on OAC use will be established.

NRC Form 366
(9-83)

BACKGROUND

The Nuclear Service Water [EIIS:BI] (RN) System serves as the ultimate heat sink in providing the station with a nuclear safety related cooling system. Most of the heat loads are cooled directly by heat transfer to the once-through river water. Those heat exchangers [EIIS:HX] in which a tube leak could allow radioactive fluid to enter the cooling water are cooled through the closed loop Component Cooling [EIIS:CC] (KC) System. Heat is then transferred to RN via the KC heat exchangers. The one exception is the Containment Spray [EIIS:BE] (NS) System Heat Exchangers, which returns are monitored for radioactivity before returning to the RN discharge line.

The RN System is served by two bodies of water, Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP). The SNSWP serves as the nuclear safety water supply sufficient to bring the station to a cold shutdown condition following a Loss of Primary System Coolant Accident (LOCA) on one unit. Water is supplied to the RN Pump [EIIS:P] Structure via separate intake lines from Lake Wylie and the SNSWP. The RN Pump Structure is a seismically designed concrete structure which provides protection for the RN Pumps. There are two separate pits within the structure, physically separating Train 'A' and Train 'B'. Two pumps in each pit (four total) provide discharge flow to a common header which supplies cooling to the related train on both Units.

Technical Specification 3.7.4 identifies the limiting condition for operation (LCO) for the RN System. With both Units 1 and 2 above Mode 5, Cold Shutdown, two independent RN loops shall be operable with each loop containing two operable RN pumps and associated emergency diesel generators [EIIS:GEN] (D/G), two essential supply and return headers, and a flow path capable of being aligned to the SNSWP. With only one Unit above Mode 5, the two independent RN loops are required to be operable with each loop containing one operable RN pump and the before mentioned equipment associated with the operating Unit. If the LCO cannot be met, the required action is to restore operability within 72 hours, or place the affected Unit in Mode 3, Hot Standby, within 6 hours, and in Mode 5 within the following 30 hours.

The RN System provides essential support functions to Engineered Safety Features (ESF) of the station. The system is designed to supply cooling water to various heat loads in both the safety and non-safety portions of each Unit. Provisions are made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under accident conditions. Sufficient redundancy of piping [EIIS:PSP] and components is provided to ensure that cooling is maintained to essential loads at all times.

The RN System can meet its safety function, provide cooling water to essential loads, with:

1. One operating RN pump in the one-pump analysis mode supplying cooling water to one essential loop.

2. Two operating RN pumps supplying cooling water to one essential loop (two essential headers) and both nonessential headers, or two RN pumps supplying total flow demands of one unit with limited flow to the other.

The RN System layout is designed such that mechanical components in the RN pump house are not unit related. Each essential RN loop has a single supply line and a single return line that serves both units. Therefore, the 1E electrical bus that provides power to the components is the only tie to a specific unit. Major loop isolation and crossover valves [EIIS:V] are normally powered from Unit 1, but can be supplied from Unit 2 during prolonged Unit 1 diesel generator outages. The common A and B supply lines and crossovers between units allow flow from any RN pump to be directed to any RN header, while ESF actuated valves provide loop separation and essential header alignment during design basis events.

The 600 VAC Essential Auxiliary Power System (EPE) for Unit 1 is provided to supply Class 1E power through load centers to the 600 VAC essential motor control centers (MCC) and consists of two redundant safety trains, A & B. MCC 1EMXG and its subfed MCC, 1EMXO, are Train 'A' rather than unit related and can be fed from either Unit 1 or 2 essential load centers (1,2 ELXA). Each of the load center breakers [EIIS:BKR] can be operated manually by means of the controls provided on the load center or automatically by the diesel generator load sequencer under accident or blackout conditions. The feeder breakers to the MCC's have no automatic control.

MCC 1EMXG supplies 600 VAC to valves 1RN-54A (RN Discharge Crossover Isolation Valve), 1RN-57A (RN Discharge to Conventional Service Water (RL) System Valves), and 1RN-63A (Loop A&B Return to Standby Nuclear Service Water Pond (SNSWP)). MCC 1EMXG also feeds MCC 1EMXO which in turn supplies 600 VAC to 1RN-1A and 1RN-5A (RN Pump Pit Intake from Lake Wylie Isolation Valves), 1RN-3A (RN Pump Pit Intake from SNSWP Isolation Valves), and 1RN-36A (RN Pump Lube Injection Strainer Inlet Crossover Isolation Valves).

The diesel generator and its load sequencing system are designed to automatically energize the necessary blackout and/or LOCA required loads. A loss of voltage sensed at the 4160 VAC essential switchgear bus, 1ETA, 1ETB, or a safety injection actuation signal will actuate the sequencer.

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

The Engineered Safeguard Features (ESF) Bypass Indication System is provided to alert the Control Room (C/R) operator of a bypass (inoperable) status of a train of any safety related system. The indicating light panel is located in the C/R. The awareness generated from these alarms should assure that both trains of a system are not bypassed at the same time.

EVENT DESCRIPTION

On March 22, 1991 at 1900 hours, Unit 1 was in Mode 5, Cold Shutdown, in preparation for End of Cycle refueling activities (U1EOC5). Unit 2 was in Mode 1, Power Operations. The Operations (OPS) night shift was scheduled to initiate the Unit 1 crud burst, complete the 1A D/G 24 hour run followed by its removal from service, swap 1ETA to SATA, and remove RN 'A' Train from service.

On March 23, at 0100 hours, the 1A D/G 24 hour run was complete. OPS swapped 1ETA feed from ATC to SATA to allow for the 4160 Essential Power System Test on Unit 1 per procedure PT/1/A/4350/06.

At 0300 hours, OPS was removing 1A D/G and RN 'A' Train from service. RN 'A' Train was being removed from service to perform maintenance and modifications. Procedure OP/1/A/6350/02, Diesel Generator Operation, was being used to remove the D/G from service. The procedure had been reviewed by the Unit Supervisor and the Non-Licensed Operator (NLO) to ensure all necessary steps were performed due to the complexity of the RN 'A' Train and 1A D/G removal from service. When the NLO reached step 2.7.1 which states, "Ensure MCC 1EMXG is being fed from 2ELXA", he conferred with the Unit 1 Supervisor again. The Unit 1 Supervisor emphasized that 1EMXG must be powered from Unit 2.

At 0337 hours, 1A D/G was removed from service. 'A' Train RN and Emergency Core Cooling System (ECCS) 'A' Train became inoperable due to shared RN valves. All RN 'A' Train dependent safety equipment was rendered inoperable as a result.

At 0400 hours, the NLO asked for the Unit 1 Operator at the Controls' (OATC) help in ensuring the 1EMXG was powered from Unit 2. The OATC called up the 'A' Train 4160V graphic on the Unit 1 OAC. The OATC and NLO concluded that the Unit 2 feeder breaker (2ELXA) to 1EMXG was closed and the feeder breaker from 1ELXA was open. The NLO signed off step 2.7.1 based upon the OAC indication and proceeded with OP/1/A/6350/02.

At 0500 hours with 'A' Train RN declared out of service, 2A D/G, 'A' Train Control Room Ventilation (VC), and 'A' Train Emergency Core Cooling Systems (ECCS) were removed from service.

On March 26, 1991, at 0445 hours, 2A D/G and Unit 2 RN 'A' Train was declared operable.

At 0525 hours, 'A' Train VC was declared operable.

*U.S. GPO: 1988-520-589:00070

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

At 1400 hours, the C/R received an alarm on the Engineered Safeguard Feature Bypass Panel (1.47 Bypass). C/R personnel investigated and found that 1EMXG was still being fed from 1ELXA which had not been backed by an emergency power supply since 1A D/G was removed from service.

At 1504 hours, 1EMXG was placed on 2ELXA. RN Train 'A' and associated equipment on both Units were without emergency back-up power for 3 days, 11 hours and 27 minutes.

When Unit conditions permitted on April 17, the 1EMXG breaker alignment that existed on March 23 was recreated and proper Unit 1 OAC graphic display was verified. Without more definitive proof, it was concluded that the NLO and OATC may have misread the graphic display on March 23.

CONCLUSION

From 0337 hours on March 23 to 1504 hours on March 26, certain RN valves (needed for operability of RN Train 'A') were without emergency backup power, extending beyond the established 72 hour time limit for RN train inoperability. During this period RN Train 'A' and associated safety systems were inoperable.

This incident has been attributed to an apparent Inappropriate Action on the part of the NLO and OATC due to misreading the OAC graphics to ensure the alignment of 1EMXG.

The 1.47 Bypass failure was caused by a damaged terminal strip which caused the signal to the panel to fail. Had this alarm responded properly, the operators could have responded by placing 1EMXG on 2ELXA within the required 72 hour action statement. The damaged terminal was repaired by Instrument and Electrical (IAE) per W/R 49167OPS.

Subsequent corrective actions by Operations personnel included alignment of 1EMXG to 2ELXA and initiation of work request 49167OPS to investigate and repair the 1.47 Bypass malfunction. Operations personnel involved with the incident were counseled on the importance of proper completion of procedure steps.

An OPS update was issued to inform operators that OAC graphics are not to be used for procedure sign-offs. OMP 2-33 will be enhanced to specify that the OAC graphics should not be used to complete procedure sign-offs involving breaker position. These measures will remain in place until further corrective action is taken.

A previous event, LER 414/88-19, involved a reactor trip as a result of cycling of a 120v AC supply breaker to the Auxiliary Control Power System. Operators cycled the breaker following a review of OAC graphics, from which they incorrectly concluded that the panel was deenergized. Corrective action for this event included a review of OAC graphics for accuracy and usefulness.

Needed changes to the OAC graphics were documented in change requests to the cognizant computer group. Also as a result of the previous event, training on proper use of OAC graphics was added to licensed operator lesson plans.

LER 414/87-07 involved a reactor trip as a result of cycling the breaker supplying 2EPD power. In this event, the CROs consulted the OAC graphics and then dispatched an operator to check local indication; they decided to wait for a report from the operator before taking action. Upon local inspection, the breaker appeared to be loose and tripped and a decision was made to reclose it. In this event, the OAC graphic was useful in diagnosing the source of the undervoltage indication.

CORRECTIVE ACTION

SUBSEQUENT

1) OPS personnel aligned 1EMXG to 2ELXA.

2) OPS personnel initiated W/R 48255OPS to investigate and repair OAC graphics.

3) OPS personnel initiated W/R 49167OPS to investigate and repair 1.47 Bypass.

4) The damaged 1.47 Bypass terminal was repaired by the Instrument and Control section.

5) OPS emphasized through an operator update that the OAC graphics should not be used for procedure sign-offs.

PLANNED

1) Enhance OMP 2-33 to specify that procedures should not be completed by determining a breaker's position from the OAC indication or graphics unless the procedure specifies to do so.

2) Evaluate the OAC graphics accuracy and implement appropriate enhancement to include control of OAC changes and logic verification if needed.

3) Evaluate and provide enhanced training on the OAC policy, use, indications, and limits of the OAC.

SAFETY ANALYSIS

The significance of this event will be evaluated by considering two different time periods. The first period corresponds to the 72 hours beginning at 0337 hours on March 23. During this time, Unit 1 was in Mode 5 and Unit 2 was in Mode 1. 1A D/G was removed from service for outage maintenance. At 0500 hours, Train 'A' RN was removed from service for outage related maintenance with the Unit 1 RN header and 'A' pit drained and the Unit 2 header essentially filled. Train 'B' RN was operable along with 1B, 2B D/G. Unit 2 correctly entered the Tech Spec Action Statement for one train of RN inoperable. The second period is the approximately 12 hour interval following the initial 72 hour period. During this interval, Unit 2 had exited the Tech Spec Action Statement. However, the lack of an operable emergency power source for 1EMXG affected the RN valves needed to swap RN Train 'A' to the SNSWP in the event of a loss of Lake Wylie due to a seismic event. The effects of this condition have been evaluated.

Scenario A

In the event of a design basis earthquake, during the initial 72 hour period of time, the Wylie dam is assumed to fail resulting in a loss of lake level in addition to a loss of offsite power (LOOP). At least 10 hours would elapse before lake levels would decrease to the point where swap to the SNSWP would be required.

1B & 2B D/G would start on detection of the LOOP and Train 'B' RN valves would align to the Standby Nuclear Service Water Pond (SNSWP) on 2 of 3 emergency low pit level. Unit 1 core cooling would be accomplished by Train 'B' of the Residual Heat Removal (ND) System.

Unit 2 core cooling would be accomplished by natural circulation, turbine driven auxiliary feedwater addition, and heat removal from the steam generators via the main steam safety valves. 1A D/G would not start due to being removed from service. Due to the LOOP and 1A D/G being inoperable, 1ETA would be deenergized and its associated loads (including 1EMXG) would not actuate. RN Train 'A' would have been unable to swap to the SNSWP.

During the 72 hour period in which both units were in the RN Tech Spec Action Statement, the station would have been able to mitigate an additional postulated single failure only with operator action. This is consistent with the Catawba Design Basis Specification CNS-1574.RN-00-0001, Section 20.2.1 and Technical Specifications.

During the subsequent 12 hour period in which 1EMXG was without emergency power, the 2A D/G had been returned to service and the Unit 2 portion of RN Train 'A' also had been returned to service. Had the postulated accident scenario occurred during this period, the plant response would be the same as described above. In addition, the 2A D/G would start and operate as long as cooling water was available from the lake. During this period of time, operator action (as described below) to align 1EMXG to its alternate power source would restore Train 'A' RN operability for Unit 2 equipment. Assuming unavailability of Train 'A' RN (without realignment of 1EMXG) the station would be vulnerable to an additional postulated single failure (without operator action), just as it was during the preceding 72 hour period, when the 2/3 emergency low low level signal energized swapover logic and the SNSWP Train 'A' valves were not able to respond.

The probability of a design basis earthquake leading to dam failure during the 12 hour period is extremely small. Additional failures would need to occur in order to have placed either Unit at risk. The incremental core melt risk associated with the 12 hour period was calculated to be much less than 10E-7. This analysis demonstrates a very small impact on the health and safety of the public.

While the 2A D/G was running, Control Room Operators (CROs) would have had ample indication to determine that 1EMXG was without power and at least 10 hours to align 1EMXG to its alternate source. The CROs would receive several indications that could be used to determine that 1EMXG was deenergized. After the earthquake/LOOP event, the CROs would respond per the Reactor Trip Response procedure, EP/0/A/5000/01, which requires that the CROs monitor 1ETA 4160 Essential Switchgear condition. With 1A D/G out of service, 1ETA would be deenergized. Unit 1 C/R annunciators "600V ESSENTIAL POWER LOAD CENTER TRN A TROUBLE" would alarm along with a Unit 2 C/R annunciator "600V/120V ESSENTIAL POWER/MCC PANEL TRN A TROUBLE". (These annunciators are train related rather than unit related.) The CROs would also receive annunciator alarms associated with Train 'A' of the Control Room Ventilation [EIIS:UC] (VC) System. With 1EMXG aligned to 1ELXA, Train 'A' of the VC System would not have started as required during the early stages of the DBE/LOOP event. Operators would have been dispatched to investigate why VC Train 'A' was not functioning and would have discovered that Train 'A' was not energized. (Note that, the 84 hour period in which VC Train 'A' was without emergency power is within the 7 day time limit allowed by the VC System Tech Spec, for an inoperable train. Therefore, an additional single failure on the VC 'B' Train need not be considered.)

Operators would also be required to perform the Loss of Nuclear Service Water procedure, AP/0/A/5500/20, upon receipt of a Lo Lo RN Pit Level annunciator. This would require that the CROs verify that the RN valves cycled the intake from Lake Wylie to the SNSWP. Per the Annunciator Response procedure, they would dispatch an operator to the RN pumphouse. The C/R switches on the main control board would indicate that the 'A' Train RN valves had not cycled. Breakers which provide power to these valves are identified with engraved nomenclature. From this evaluation, operator action to realign 1EMXG is seen to be realistic and probable.

Scenario B

A second, postulated accident scenario was also considered: a loss of coolant accident and concurrent loss of offsite power on Unit 2. FSAR Section 9.2.1.2.2 states that the RN System design basis is for operation under the worst initial conditions of operation. This condition is assumed to be the low probability combination of a loss of coolant accident in one unit, extended shutdown of the

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other unit, loss of the downstream dam, and a prolonged drought and hot weather and its effect on the Standby Nuclear Service Water Pond. The plant response would be the same as for the above postulated earthquake scenario, except that Unit 2 decay heat would be removed by the emergency core cooling system. As described above, ample time would be available to the operators to detect the lack of power to 1EMXG and to take action to align it to its alternate source.

In summary, operator action to realign 1EMXG to its alternate power supply would be expected based on available indications, procedural guidance, and time. Although the station was vulnerable to a postulated single failure during the period of time that 1EMXG was without emergency power, during the first 72 hours of this period the Units were within the Tech Spec Action Statement. The risk of operating in this condition has been assumed to be acceptable. For the subsequent 12 hour period, the incremental core melt risk was shown to be much less than 10E-7. Thus, the health and safety of the public were not affected.