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January 24, 1994

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

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/93-012

Gentlemen:

Attached is Licensee Event Report 413/93-012, concerning UNIT 1 ENTERED MODE 3 WITH INOPERABLE AUXILIARY FEEDWATER PUMP.

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

Very truly yours,

W. R. McCollum, Jr. for D. L. Rehn

D. L. Rehn

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NRC FORM 366 5/92 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME(1)	DOCKET NUMBER(2)	PAGE(3)
Catawba Nuclear Station, Unit 1	05000413	1 of 5

TITLE(4)
Unit 1 Entered Mode 3 With Inoperable Auxiliary Feedwater Pump

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)
									N/A	05000
12	25	93	93	012	00	01	24	94		05000

OPERATING MODE(9)	3	THIS REPORT IS SUBMITTED PURSUANT TO REQUIREMENTS OF 10CFR (Check one or more of the following)(11)							
POWER LEVEL(10)	0		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
Z. L. Taylor, Compliance Manager	803 831-3812

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)		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 December 25, 1993, at 1546 hours, Unit 1 entered Mode 3, Hot Standby, from Mode 4, Hot Shutdown, following refueling outage 1EOC7 with the Turbine Driven Auxiliary Feedwater Pump (CAPT) inoperable. Two of the Main Steam Supply System (SA) valves to the CAPT had been previously tagged closed for maintenance. Unit 1 Operations shift personnel were aware of the SA valve tagout. The Unit 1 Operations shift reviewed the Technical Specification (T/S) along with the associated T/S Interpretation and concluded that entry into Mode 3 was allowable. On December 26, the SA valves were opened and the CAPT was successfully tested. This event has been attributed to incorrect use of T/S and T/S Interpretations. Corrective actions include review and revision of associated T/S and T/S Interpretations, issuance of an operator update referencing the causes of this event, and an evaluation of Operation procedures to determine if hold points are necessary to prevent reccurence.

BACKGROUND

The safety related function of the Main Steam To Auxiliary Equipment [EIIS:SA] (SA) System is to supply and regulate steam as required to drive the Auxiliary Feedwater [EIIS:BA] (CA) System Pump [EIIS:P] Turbine [EIIS:TRB]. During normal power operation, the safety related portions of the SA System are in a standby mode with valves [EIIS:V] 1(2)SA-2, Steam Generator (S/G) 'B' Main Steam To Auxiliary Feedwater Pump Turbine Isolation Valve, and 1(2)SA-5, S/G 'C' Main Steam To Auxiliary Feedwater Pump Turbine Isolation Valve, in the closed position.

The safety related function of the CA System is to provide an assured source of emergency feedwater to maintain secondary side level within the steam generators at times when the normal supply from the Main Feedwater [EIIS:SJ] (CF) System is not available. In this function the CA System is relied upon to remove heat from the Reactor Coolant [EIIS:AB] (NC) System in order to prevent over-pressurization which could result in fuel damage. Each units CA System consists of two motor [EIIS:MO] driven pumps and one steam driven turbine pump. The steam supply for the steam driven turbine pump is provided from the SA System as previously described.

CAPT Testing is required by Catawba Nuclear Station (CNS) Technical Specification (T/S) Auxiliary Feedwater Surveillance Requirements 4.7.1.2.1 when the associated unit is in Modes 1, Power Operation, Mode 2, Startup, and Mode 3, Hot Standby. CNS T/S Interpretation 4.7.1.2.1a.2, CAPT Operability Requirements For Entry Into Mode 3 From Mode 4, allows the unit to enter Mode 3 with an expired head curve surveillance. The interpretation states that when entering Mode 3 from Mode 4 with an expired surveillance interval for the CAPT, the CAPT shall be operable in the sense that it is ready to be surveillance tested and CNS management has confidence in the ability of the CAPT to pass the surveillance test.

The Tagout Removal And Restoration (R&R) Procedure, Operations Management Procedure (OMP) 2-18, describes the R&R process performed by Operations. The Tagout (R&R) Record Sheet provides a combined list of safety tags (Red Tags and White Tags), equipment removed from service, and other pertinent information for safe performance of work within the plant. A copy of an active tagout is found within the Control Room copy of the procedure(s) affected by the equipment removed from service.

EVENT DESCRIPTION

On December 23, 1993, Unit 1 was in Mode 5, Cold Shutdown. 1SA-1, Main Steam 1B To CA Pump Number 1 Maintenance Isolation Valve, 1SA-4, Main Steam 1C To CA Pump Number 1 Maintenance Isolation Valve, and 1SA-89, Auxiliary Steam To CA Pump Number 1, were tagged out for CAPT overspeed trip testing under Work Order Number 93055813-01.

The CAPT overspeed trip testing was completed on December 23. Mechanical Maintenance (MNT) signed and returned the stubs of Red Tags 13696, 13697, and 13698 to Operations (OPS) so that 1SA-1 and 1SA-4 could be reopened. 1SA-89 was to remain closed in its normal position.

Operations did not remove the Red Tags in case isolation was needed for additional work on valve 1SA-145, CA Pump Turbine Stop Valve. The Red Tag stubs were located in the Unit 1 Shift Supervisor office.

On December 25, prior to entering Mode 3, the Work Control Center (WCC) Senior Reactor Operator (SRO) performed the review of the open R&R's and Temporary Station Modification logs in order to determine if outstanding work affected entry into Mode 3. Noting that the open Red Tags for R&R 13-2980 affected the CAPT, the WCC SRO and the Unit 1 Supervisor referenced T/S 3/4.7.1.2, T/S 4.0.4, T/S Interpretation 4.7.1.2.1a.2 and T/S Interpretation 4.0.4 to determine if Unit 1 could enter Mode 3.

At approximately 0900 hours, the WCC SRO communicated to the Shift Supervisor who was performing the Mode 3 Checklist for OP/1/A/6100/01, Controlling Procedure For Unit Startup, that no outstanding work or R&R's prevented entry into Mode 3.

On December 25, 1993, at 1546 hours, Unit 1 entered Mode 3 with the CAPT inoperable because 1SA-1 and 1SA-4 were closed which isolated the steam supply to the CAPT.

On December 26, at approximately 0800 hours, after discovering that 1SA-1 and 1SA-4 were required to be open, the SA valve red tags were cleared. 1SA-1 and 1SA-4 were reopened for CAPT surveillance testing.

CONCLUSION

This event has been attributed to the T/S documents not being followed correctly. Technical Specification 3.7.1.2 requires that three auxiliary feedwater pumps and associated flow paths be operable in Modes 1, 2, and 3. T/S Interpretation 4.7.1.2.1a.2 allows entry into Mode 3 without performing an expired surveillance that is required by T/S 4.7.1.2.1a.2, Head Curve Performance Test. The CAPT test requires that the unit be in Mode 3. The T/S Interpretation

states that the CAPT must be declared inoperable and the applicable action statement must be entered when the secondary steam pressure reaches 600 PSIG. The WCC SRO and the Unit 1 supervisor determined that entry into Mode 3 would not be affected by an inoperable CAPT since entry into Mode 3 was allowed by the T/S Interpretation with an expired CAPT surveillance test. The T/S Interpretation further states that when entering Mode 3 from Mode 4 with an expired surveillance interval for the CAPT, the CAPT shall be operable in the sense that it is ready to be surveillance tested and CNS management has confidence in the ability of the CAPT to pass the surveillance test. The T/S Interpretation also requires that CAPT steam supply valves 1SA-2 and 1SA-5 be operable. The Unit 1 CAPT was actually inoperable prior to entry into Mode 3 because 1SA-1 and 1SA-4 were red tagged closed which would render the CAPT steam supply line inoperable.

1SA-1, 1SA-4, and 1SA-89 had been red tagged closed so that Mechanical Maintenance could perform the CAPT overspeed trip test which is a required surveillance. Work that required 1SA-1, 1SA-4, and 1SA-89 to be closed had been completed and the red tag stubs had been signed by MNT and given to OPS. Red Tag R&R 13-2980 remained in the tagout book and the red tag stubs remained in the Unit 1 Shift Supervisor office. Operations shift personnel were aware that the tagout was still open.

Upon discovery that 1SA-1 and 1SA-4 were required to be open prior to entering Mode 3, the red tags were cleared and the valves were opened. The CAPT was successfully tested and declared operable. Planned corrective actions include informing Operations shift personnel about use of T/S Interpretations along with a review of the T/S and T/S Interpretations involving CAPT operability.

A review of reportable incidents during the past twenty four months revealed one event where a document was not followed properly. LER 414/93-001 describes an Essential Auxiliary Power [EIIS:EB] System B Train Blackout while Unit 2 was in No Mode on February 21, 1993. Instrument and Electrical technicians were performing timer calibrations on the Emergency Diesel Generator Load Sequencer [EIIS:EK] when a procedure step to place the sequencer in the test mode was inadvertently missed. Both reportable events had the same cause, but involved different equipment, systems, and procedures. The corrective action for LER 413/93-001 is not applicable to the current event. Therefore, this event is determined not to be a recurring problem.

CORRECTIVE ACTION

SUBSEQUENT

1) Red Tags were cleared. 1SA-1 and 1SA-4 were opened.

2) CAPT was surveillance tested and returned to service.

PLANNED

1) Operations shift personnel will be informed about this event through an Operator Update.

2) Operation Management will evaluate the appropriate procedures in order to identify potential hold points for discussion of condition changes.

3) T/S Interpretation 4.7.1.2.1a.2 and T/S Interpretation 4.0.4 will be reviewed and revised. Reference to valves SA-2 and SA-5 will be removed from the T/S Interpretation 4.7.1.2.1a.2 in order to appear more like the mode change procedure step.

SAFETY ANALYSIS

The Unit 1 CAPT was inoperable when entered Mode 3 on December 25, 1993, at 1546 hours because two of the steam supply line valves were red tagged closed. The safety impact of the unavailable CAPT is not significant. The primary concern when performing a Loss Of Coolant Accident (LOCA) event analysis is the ability to remove decay heat assuming only a single Motor Driven CA Pump is available. The amount of decay heat present at the time of this event was minimal due to the fact that the core was newly refueled and had yet to reach power operation. It can be concluded that a single Motor Driven CA Pump would be more than adequate and that the impact of the unavailable CAPT during Mode 3 on the accident analysis is not significant. The severe accident analysis evaluation on the significance of this event determined that the increase in probability of a core damage event were negligible.

Therefore, the health and safety of the public were not affected by this event.