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March 1, 1994

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

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/93-006, Revision 1

Gentlemen:

Attached is Licensee Event Report 413/93-006, Revision 1, concerning REACTOR TRIP DUE TO BLOWN FUSE IN INTERMEDIATE RANGE CHANNEL.

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

Very truly yours,

D. L. Rehn

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Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY 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 INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	1 OF 06

TITLE (4)
Reactor Trip Due To Blown Fuse In Intermediate Range Channel

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
06	12	93	93	006	01	03	01	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 2
POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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 (Include 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
F	JG	JX	5675	Y					

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 15 single-spaced typewritten lines) (16)

On June 12, 1993, a Unit 1 Reactor (Rx) Trip occurred at 1042 hours while shutting down to make repairs in the Main Condenser. Unit 1 was in Mode 2, Unit Startup, at Hot Zero Power, on the way to Mode 3 via control rod insertion, when an automatic Rx trip was initiated by the Intermediate Range (I/R) Channel N35 Hi Flux Rx Trip Bistable. This signal was generated by the loss of control power voltage to N35 caused by a blown fuse. Main Feedwater (CF) System isolation occurred immediately upon Rx Trip due to Rx Coolant (NC) System Average Temperature being below 564 degrees F. Both Motor Driven Auxiliary Feedwater (CA) System Pumps were manually started to reinitiate NC cooldown. CA flow was throttled to control NC cooldown. Pressurizer (Pzr) pressure and level had already decreased significantly before the trip; the subsequent cooldown, caused by CA, resulted in a decrease of Pzr level below 17 percent which initiated letdown isolation. Letdown was restored approximately nine minutes later when Pzr level and pressure were recovered. The cause of the blown fuse was determined to be a combination of two factors: 1) control power in rush capacity problem and 2) an intermittent internal failure of the detector high voltage power supply. Corrective actions included replacing the power supply and replacing all control power fuses with a type that has a much higher in rush current capacity.

BACKGROUND

The Ex-core Nuclear Instrumentation [EIIS:JG] (ENB) System provides trip and interlock signals to the Reactor [EIIS:RCT] Protection [EIIS:JC] (IPE) System and a continuous indication of Reactor power from Startup through Power Operation to the Control Room Operators (CROs). This is accomplished by measuring neutron leakage from the Reactor Core over the entire operating range of the Reactor and providing the trips and interlocks at various power levels to ensure an adequate margin of safety during all phases of Reactor startup and operation.

The ENB System consists of three separate overlapping ranges of instrumentation: Source Range (S/R), Intermediate Range (I/R), and Power Range (P/R). The Source Range provides indication and protection during startup and covers the first six decades of operation. The Intermediate Range provides indication and protection over the next eight decades of operation and overlaps the Source Range by approximately 2.5 decades and the Power Range instrumentation by 100% of its range. The Power Range instrumentation provides indication and protection while the Reactor is operating at power. It is calibrated from 0 to 120% Reactor power.

The Main Feedwater [EIIS:SJ] (CF) System consists of two steam driven feedwater pumps [EIIS:P], two stages of high pressure feedwater heaters (A and B), piping [EIIS:PSP], valves [EIIS:V], and instrumentation. Normally, both feedwater pumps will be operating with each pump handling half the feedwater flow. Downstream of the feedwater pumps, the feedwater passes through two stages of high pressure heaters to a final header where the temperature is equalized. The feedwater is then admitted to the steam generators [EIIS:HX] (S/G) through four steam generator feedwater lines, each of which contains a control valve and a flow nozzle.

The purpose of the feedwater isolation signal is to initiate isolation of each steam generator and rapidly terminate feedwater flow and steam blowdown inside containment [EIIS:NH] following a main steam or feedwater line break in containment, to prevent loss of steam generator water inventory due to a pipe rupture outside containment, and to prevent overfilling the steam generators if for some reason the normal means of controlling steam generator level malfunctions. Feedwater isolation is activated by any one of the following signals: safety injection, reactor trip plus low average reactor coolant temperature (T-ave less than 564 degrees F), or Hi-Hi Steam Generator level. A feedwater isolation signal closes the Feedwater Isolation Valves, Feedwater Purge Valves, Feedwater Control Valves, Feedwater Control Bypass Valves, Feedwater Preheater Bypass Valves, and Feedwater Bypass Tempering Flow Valves.

The Auxiliary Feedwater [EIIS:BA] (CA) System assures sufficient feedwater supply to the steam generators in the event of loss of the CF System, to remove primary coolant stored and residual core energy. The system is designed to start automatically in the event of loss of offsite electrical power, trip of both CF pumps, safety injection signal, or low-low steam generator water level; any of which may result in, coincide with, or be caused by a Reactor trip. In addition, the CA System will supply sufficient feedwater flow to maintain the Reactor at hot standby for two hours followed by cooldown of the Reactor Coolant [EIIS:AB] (NC) System to the temperature at which the Residual Heat Removal [EIIS:BP] (ND) System may be operated.

The purpose of the Pressurizer [EIIS:VSL] is to control the NC System pressure. In the Pressurizer water and steam are maintained in equilibrium by electrical heaters [EIIS:EHTR] and water spray. Steam can be formed (by the heaters) or condensed (by the spray) to reduce pressure variations due to contraction and expansion of the Reactor Coolant. The liquid level in the Pressurizer is maintained by level transmitters [EIIS:XT] which provide signals for use in the Reactor Coolant and Protection System (RCPS), the Safety Injection [EIIS:BQ] (NI) System and the Chemical and Volume Control [EIIS:CB] (NV) System. Each transmitter provides an independent high water level signal that is used to actuate an alarm and, upon 2 out of 3 coincident alarms, will cause a reactor trip. The transmitters also provide independent low water level signals that will activate an alarm. If the liquid level falls to a fixed low level alarm setpoint, it will trip the Pressurizer heaters "off" and close the letdown line isolation valves.

EVENT DESCRIPTION

On June 12, 1993, at 0001 hours, Unit 1 shutdown was commenced to make repairs in the main condenser.

At 1042 hours, Unit 1 was in Mode 2, at Hot Zero Power, on the way to Mode 3 via control rod insertion. NC Average Temperature was 549 degrees F with NC Pressure at 2215 PSIG. Pressure level was at 20 percent.

At 1042:13 hours, Reactor power had decreased to 2E-08 amps (less than 1 percent Reactor power) on the Intermediate Range Channel when the control power fuse in Channel N35 failed causing the Intermediate Range High Flux Reactor Trip Bistable [EIIS:XIS] to trip, opening the Reactor Trip Breakers [EIIS:BKR], and dropping the uninserted rods [EIIS:ROD] into the core. Concurrent with the Reactor Trip Breakers opening, a Feedwater Isolation occurred due to the average Reactor Coolant temperature being less than 564 degrees F.

At 1047 hours, CA Pumps 1A and 1B were manually started to reinitiate NC cooldown which had been halted by the Reactor Trip and CF isolation. CA pumps were started as required action in response to CF isolation per Operation Reactor Trip Procedure.

At 1050 hours, CA flow to the Steam Generators was manually throttled to control NC cooldown.

At 1050:11 hours, Pressurizer heaters automatically de-energized at the proper setpoint of less than 17 percent Pressurizer level.

At 1050:22 hours, letdown isolation occurred due to Pressurizer Level Channel 2 dropping below 17 percent. Pressurizer pressure and level had already decreased significantly before the trip; so the subsequent cooldown, caused by CA, resulted in decrease of Pressurizer level below 17 percent which initiated letdown isolation.

At 1050:59 hours, Pressurizer heaters re-energized because Pressurizer level was above 17 percent.

At 1059 hours, letdown was restored.

At 1100 hours, work order 9304170701 was issued to investigate and repair the cause of the blown fuse on Intermediate Range Channel N35.

CONCLUSION

Root cause of the blown fuse was determined to be a combination of two factors: 1) control power in rush capacity problem and 2) an intermittent interval failure of the detector high voltage power supply. When the power supply output voltage would exhibit the failure, the intermediate range drawer log current amplifier would produce an extremely erratic output of such amplitude that the P6 bistable saturable reactor transformer action would result in a high control power in rush current. The combination of this failure and the fuse in rush capacity problem resulted in the control power fuse failure. The in rush current capacity for the MH-5 control power fuses in the Nuclear Instrumentation System (NIS) had been previously identified by Westinghouse. Catawba had initiated a modification (CE-4146) to replace all NIS control power fuses prior to this event. Corrective actions included replacing the power supply and replacing all control power fuses with a type that has a much higher in rush current capacity. Failure of the detector high voltage power supply is NPRDS reportable.

During this event the CROs entered the correct emergency response and abnormal procedures (EPs and APs) and performed the required steps to maintain the plant in a safety shutdown condition. All safety systems responded as designed to shutdown the Reactor and also maintain it in a safe shutdown condition.

A review of the Operating Experience Program (OEP) database for the past 24 months prior to this event did not identify any Reactor trips that were attributed to blown fuses or problems associated with Intermediate Range Channels. Therefore, this incident is considered not to be recurring.

CORRECTIVE ACTIONS

IMMEDIATE

1) Control Room Operators entered procedure EP/1/A/5000/01 (Reactor Trip or Safety Injection) and performed the required actions.

2) Per guidance in EP/1/A/5000/01 (No Safety Injection Occurred), Control Room Operators entered procedure EP/1/A/5000/01A (Reactor Trip Response) and performed the required actions.

3) Control Room Operators manuall[illegible] ted CA pumps and manually throttled CA flow to control NC cooldown.

4) Control Room Operators entered procedure AP/1/A/5500/12 (Loss of Charging or Letdown) and performed the required actions.

5) Control Room Operators entered procedure AP/1/A/5500/16 (Malfunction of Nuclear Instrumentation System) and performed the required actions.

SUBSEQUENT

1) Work Order 9304170701 was issued to determine the root cause of loss of control power fuse failure on Intermediate Range detector N35.

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

2) Systems Engineering investigated the Reactor Trip and prepared Reactor Trip report. Reactor Trip report concluded that all systems responded as designed to shutdown the Reactor and maintain it in a safe shutdown condition.

3) Intermediate Range drawer that caused the trip was replaced and the new drawer was installed and tested under work order 9304170701.

4) Work order 9304331101 was issued to monitor and perform testing on the faulty drawer to determine the cause of the blown fuse.

5) Intermediate Range detector (N-35) high voltage power supply was replaced.

6) Nuclear Instrumentation System control power fuses for both units were replaced per minor modification CE 4146.

SAFETY ANALYSIS

The Unit was in Mode 2 at the time of this incident. Unit 1 Reactor Trip occurred due to loss of control power voltage to Intermediate Range Channel N35. Since the Unit was virtually shutdown at the time of this event, a very minor transient occurred. CF isolation occurred immediately upon Reactor trip due to Reactor Coolant Average Temperature being below 564 degrees F. Both Motor Driven Auxiliary Feedwater (CA) System Pumps were manually started to reinitiate NC cooldown which had been halted by the Rx Trip and CF Isolation. CA flow was manually throttled to control NC cooldown. Pressurizer pressure and level had already decreased significantly before the trip so this subsequent cooldown, caused by CA, resulted in a decrease of Pressurizer level below seventeen percent which caused the Pressurizer heaters to de-energize and letdown isolation occurred. Pressurizer level was below seventeen percent for approximately one minute. Once the Pressurizer level increased above seventeen percent, the Pressurizer heaters re-energized. Letdown isolation was restored approximately nine minutes later once Pressurizer level and pressure were recovered.

After review of this incident, all systems responded as designed to shutdown the Reactor and maintain it in a safe shutdown condition. There were no unusual releases of radioactive material.

The health and safety of the public were not affected by this incident.