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 1	0 5 0 0 0 4 1 3	1 OF 0 5

TITLE (4): Reactor Trip Due to Trip of the 1C2 Heater Drain Tank Pump

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 1	1 9	8 6	8 6	— 0 0 6 —	0 0	0 2	1 8	8 6	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 1

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)	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)	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)	
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	
Roger W. Ouellette, Associate Engineer - Licensing	7 0 4	3 7 3 - 7 5 3 0

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 January 19, 1986, at 1835 hours, the 1C2 Heater Drain Tank (C2HDT) Pump tripped while the unit was undergoing Turbine Acceptance Testing. For this testing, the normal bypass to the Main Condenser for the 1C2HDT Pump was isolated. This caused condensate to back up into the Moisture Separator Reheater, initiate a Hi Level alarm, and trip the Main Turbine. Tripping of the Main Turbine resulted in an automatic Reactor Trip. The unit was at 91% power at the time of this incident.

This event is assigned Cause Category X, Other, due to no apparent reason being found for the tripping of the C2HDT Pump.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv) and 10 CFR 50.72, Section (b)(2)(ii).

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BACKGROUND

Procedure PT/1/A/4150/14, Enclosure 13.5, Turbine Acceptance Test (TAT) requires the isolation of the Main Condenser Steam Dump valves in order to perform the test. The C2 Heater Drain Line to Condenser is isolated by valve 1HW052. This prevents bypass flow from the C2 Heater Drain Tank (C2HDT) to the Main Condenser. In addition, this test requires the use of a radioactive isotope, Na-24, to measure steam quality through the Main Turbine System (EIIS:TA). Na-24 has a half life of approximately 15 hours and it exists in particulate form in water and not steam. Therefore, the existence of the particulate in steam is a function of the quality of the steam.

1C2HDT Pump can be tripped from the control room by a start-stop switch or from the remote location at the pump by a start-stop switch. In addition, two conditions will trip the pump. Heater drain tank 1C2 Emergency Low Level or a generator full load rejection, to protect the pump from cavitation. Only the protective trips will initiate an alarm in the Control Room.

DESCRIPTION OF INCIDENT

Prior to 1835 hours on January 19, 1986, personnel were performing the TAT per PT/1/A/4150/14. At 1835:47 hours, with 145 valves of 151 valves isolated per the procedure, the 1C2HDT Pump tripped. No alarms were received in the Control Room when the trip took place. At 1836:40 hours, the Control Room received a Hi Hi Level alarm on the 1C2HDT. The Operator at the Controls (OATC) checked the 1C2HDT Pump and found that it had tripped. He immediately contacted personnel in the area of the pump who were performing the TAT. The OATC advised the personnel to open 1HW054, 1C2 Heater Drain Tank to Condenser Control Valve. Personnel proceeded to the valve. At 1838:01 hours, 1C2 Heater Emergency Hi Level alarm annunciated in the control room. At 1840:25 hours, while the personnel were in the process of opening valve 1HW054, the Moisture Separator Reheater (MSRH) Hi Level alarm actuated, the Main Turbine tripped on the MSRH Hi Level alarm actuation, and an automatic Reactor Trip occurred due to the Main Turbine Trip. At 1840:26 hours, Lo Lo Level alarms annunciated for Steam Generators (S/G) A, B, C and D. Valves 1SB015 and 1SB024, Main Steam to Condenser Control Valves, subsequently opened, but due to isolation for the TAT, delivered no steam to the condenser. The Auxiliary Feedwater (CA) (EIIS:BA) Pump Turbine started due to 2 out of 4 S/G's being in Lo-Lo Level alarm. Exhaust of the CA Pump Turbine vents to the atmosphere which permitted steam containing Na-24 to be released. At 1840:28 hours, valves 1SV07, S/G C Power Operated Relief Valve (PORV), opened. This actuation also permitted steam containing Na-24 to be released to the atmosphere. Also at 1840:28 hours, the Pressurizer (PZR) heaters energized due to low PZR pressure, and the Motor Driven CA Pumps A and B automatically started on Lo Lo S/G level. At 1840:38 hours, Main Feedwater (CF) (EIIS:SJ) Pump B tripped. At 1840:49 hours, CF Pump A tripped. At 1841:00 hours, Valve 1SV07 closed. At 1841:55 hours, the Main Condenser Booster Pumps tripped. At 1842:42 hours, the Turbine Driven CA Pump was secured. The unit entered Mode 2 at 2200 hours on January 20, 1986, and entered Mode 1 at 0420 hours on January 21, 1986.

CONCLUSIONS

This event is assigned Cause Category X, Other. The root cause of the event is the tripping of the 1C2HDT Pump. Under normal operating conditions valve 1HW054 is locked open, and upon a 1C2HDT Pump trip, water is diverted to the Main Condenser, prohibiting the backup of water into the C Heater Drain System. Under normal conditions, the tripping of the 1C2HDT Pump would have caused no more than fluxuation in the Main Feedwater flow. Due to the TAT being performed, an abnormal condition existed with valve 1HW054 being closed. A Work Request was written to investigate and repair 1C2HDT Pump. No problems with the pump could be identified. Therefore, the cause of the pump trip remains unknown.

While the TAT is being performed, isolation of the Main Steam Dump Valves is required. When these valves are isolated on the down stream side, the valves lift off their seats. Eventually, the valves reseat. This results in no effect upon system operation, but in this incident, inhibited the verification of the Steam Dump valve response. Prior to this incident, Work Requests were written to investigate and repair valves 1SB18, 1SB27, and 1SB06, (see LER 413/85-67). Due to this incident other Work Requests were written in order to investigate and repair valves 1SB03, 1SB09, 1SB12, and 1SB021. Valves 1SB12 and 1SB21 apparently never reseated prior to the trip, but after the trip they were verified as being closed. Valves 1SB03 and 1SB09 showed no response prior to and during the event.

After the event occurred, personnel were unable to reset the MSRH Hi Level alarm in the control room. This prevented the Main Turbine from being reset. A Work Request was written to investigate and repair the MSRH instrumentation. Upon investigation, the actuator arms on the micro switches had apparently been warped by the rapid increase in the water level due to the trip of the 1C2HDT Pump. The logic was tested and found to be operable. The Main Turbine was then reset. Malfunction of this device is not reportable to the NPRDS Network.

The Safety Parameter Display System (SPDS) Heat Sink Logic Tree received an invalid input during this event from the CA flow indicator to S/G B. A Station Problem Report (SPR) was written to initiate a modification of this input. At present Operations has a priority list for Nuclear Station Modifications (NSM) which will not allow an SPR to generate an NSM when the SPR is not mandatory for the continued operation of the plant. The SPR will generate an NSM at a future date as this work achieves a higher priority.

The Condensate (CM) (EIIS:SD) Booster Pumps are provided with a bypass flowpath through valve 1CM127, CM-CF Cleanup Flow Control Valve, which will prevent flow from the CM Pumps reducing below 3000 gpm and causing a pump trip. In this event 1CM127 responded 3 seconds after the CM Booster Pump tripped. A Unit 1 NSM will replace the pneumatic controller on the valve with an electronic controller (see LER 413/85-49). This will provide a more immediate response.

A pressure spike was experienced for CF Pump A and B suction pressure after CF isolation occurred. This problem has been evaluated (see LER 413/85-67).

No incidents of this type have occurred in the past. This incident occurred under abnormal conditions while the TAT was being performed.

Upon startup of the Turbine Driven CA Pump and the opening of valve 1SV07 or S/G C PORV, Na-24 that was contained in the Main Steam System for TAT purposes was released to the atmosphere. The activity released was approximately 1.83 E-3 Curies. This is a very conservative estimate due to the isotopes being in particulate form. Adequate dispersion of the activity occurred prior to reaching the site boundary. Due to the half life of Na-24 being 15 hours, any Na-24 released would have decayed in a relatively short period of time. Radiation exposure was considered insignificant to on-site personnel. Review of the HP Daily Dose Summary shows that no change in daily dose can be attributed to this incident. Off-site exposure was considered to be non-detectable. No personnel injuries occurred as a result of this incident.

CORRECTIVE ACTION

(1) Work Requests were written to investigate and repair the failure of valves 1SB03, 1SB09, 1SB12, and 1SB21, during this incident.

(2) A Work Request was written to investigate and repair ˜2HDT Pump trip. The work request was completed and the cause of the trip could not be found.

(3) A Work Request was written to investigate and repair the inability to reset the MSRH Hi Level alarm. Investigation showed the actuation arms on the level switches were out of adjustment.

(4) A SPR was written to modify the SPDS Heat Sink Logic Tree input from the CA flow indicator to S/G B.

SAFETY ANALYSIS

Following the Reactor Trip due to MSRH Hi Level, power immediately dropped to zero. Pressurizer (PZR) pressure dropped to 2049 psig, causing the PZR heaters to energize, and pressure leveled out at approximately 2235 psig after 35 minutes. PZR level dropped to 23% before leveling out at 27% after 30 minutes. Tave dropped to approximately 554 degrees F and leveled out at approximately 559 degrees F within 30 minutes of the trip.

Main Steam Bypass to Condenser Valves were isolated during this incident due to the Turbine Acceptance Testing being performed, and therefore were unable to perform their function. S/G C PORV 1SV07 lifted during this incident in order to relieve pressure in the secondary system. Both CA Motor Driven Pumps A and B and the CA Turbine Driven Pump started automatically as required. Pressure stabilized for S/G A, B, C, and D at 1080 psig, 1070 psig, 1090 psig and 1080 psig, respectively, after approximately 30 minutes. S/G levels did not drop below 17% and stabilized at 38% ($\pm$5%). Adequate core heat removal was available through the S/G's in the post trip mode. Although minimal amounts of Na-24 were released to the environment, the effects to on-site personnel and the public were undetectable.

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

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HAL B. TUCKER
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February 18, 1986

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

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/86-06 concerning a reactor trip due to tripping of the 1C2 heater drain tank pump. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

RWO:slb

Attachment

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cc: Dr. J. Nelson Grace, Regional Administrator
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