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 0 8

TITLE (4) Manual Reactor/Auxiliary Feedwater Pump Turbine Trip Following Turbine Trip/Main Feedwater Isolation Due To A Spurious High Steam Generator Level Signal

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)
1 1	0 3	8 7	8 7	0 2 9	0 0	1 2	0 3	8 7	N/A	0 5 0 0 0
										0 5 0 0 0

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

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)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	50.72(b)(1)(i)(A)
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	50.72(b)(2)(ii)
[ ] 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)
Julio G. Torre, Associate Engineer - Licensing	7 0 4 3 7 3 - 8 0 2 9

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
0	J C	V	D 2 3 2	N					
0	B A	6 5	W 2 9 0	N					

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 November 3, 1987, at 0836 hours, Steam Generator (S/G) 2C Narrow Range (N/R) Level Channel 2 inoperable due to its reading being out of tolerance. The Unit was at 93% power. Since S/G 2C N/R Level Channel 4 was also inoperable, a Technical Specification required Unit Shutdown was commenced at 0926 hours. The Channel 2 reading became increasingly erratic and at approximately 1146 hours it caused a spurious S/G high high level Turbine trip, a Main Feedwater (CF) Isolation, and a CF Pump Turbine trip. Motor Driven (M/D) Auxiliary Feedwater (CA) Pump 2B started automatically to supply feedwater to the S/Gs (M/D CA Pump 2A was already operating). At 1146:39 hours, the Reactor was manually tripped from 59% power. Approximately 11 seconds later, all four S/G levels decreased below the low low level Reactor trip setpoint. The Turbine Driven (T/D) CA Pump started automatically, however, it tripped within seconds. The M/D CA Pumps were utilized to restore S/G levels to normal and to cool down the Unit. The Unit entered Mode 4, Hot Shutdown, at 2013 hours.

This incident is attributed to the failure of the S/G 2C N/R level channels due to their reference leg vent line isolation valves leaking steam past the seat. The reference leg vent line steam leaks were stopped, and the leaking vent line isolation valves will be repaired. The exact cause of the T/D CA Pump trip could not be identified. After the T/D CA pump governor valve travel was adjusted, the Turbine was started five times and it operated properly. The recommended governor valve/linkage will be added to the Station's Preventive Maintenance program. The health and safety of the public were unaffected by this incident.

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BACKGROUND:

The Steam Generator (EIIS:SG) (S/G) Level Control System maintains a constant programmed water level in each S/G. This protects against excessive reactivity additions and Containment pressure on a steam line break and prevents Main Turbine (EIIS:TRB) damage from water carryover.

Each S/G is equipped with four channels of narrow range (N/R) level instrumentation. A Reactor (EIIS:RCT) trip signal is generated when any two of four N/R level channels in one S/G indicate 17% or less (S/G Lo Lo Level). A P-14 signal (Turbine trip, Main Feedwater (EIIS:SJ) (CF) Pump trip, CF Isolation) is generated when two of four N/R level channels in one S/G indicate 78% or higher (S/G Hi Hi Level).

The Auxiliary Feedwater (EIIS:BA) (CA) System provides an independent means of supplying feedwater to the S/Gs in addition to the CF System. The CA System functions to maintain S/G secondary side water inventory sufficient to permit an orderly plant cool down and to remove residual heat stored in the Reactor Coolant (EIIS:AB) (NC) System for the duration of all Design Basis Events. The CA System also provides condensate grade feedwater during normal Unit startup and shutdown operations when use of the CF System at such low flow rates would be undesirable. The CA System contains two full capacity Motor Driven (M/D) Pumps (EIIS:P) and one full capacity steam Turbine Driven (T/D) Pump. Any of these pumps may be started manually or automatically. The M/D pumps start automatically when both CF pumps trip or upon two out of four low low level alarms in any one S/G. The T/D Pump starts automatically if two out of four low low level alarms occur in any two S/Gs.

The T/D CA Pump Turbine is protected from damage due to overspeed by an electrical overspeed trip circuit set at 115% of rated speed and a mechanical overspeed trip device set at 125% of rated speed (rated speed is 3600 rpm). Both devices close the Turbine's Trip and Throttle (T&T) valve (EIIS:V). An electrical overspeed trip may be reset remotely in the Control Room. A mechanical overspeed trip must be reset manually at the Turbine. When the Turbine is in standby, the T&T valve and the governor valve are in the fully open position and the steam supply isolation valves are closed. Upon receipt of an auto-start signal, the steam supply isolation valves open rapidly and the Turbine comes up to rated speed within seconds. The Turbine's governor must be capable of throttling closed the governor valve quickly and smoothly to prevent an overspeed trip. There are no computer alarm points or Turbine speed recorders to document a Turbine overspeed trip. There is an electrical overspeed trip indicating light in the Control Room which illuminates when an electrical overspeed trip occurs. However, this light goes out when the Turbine's speed drops below the trip setpoint. There is a computer alarm point and Control Room annunciator to indicate that the Turbines T&T valve has closed.

Technical Specifications allow one out of four S/G N/R level channels per S/G to be inoperable in Mode 2, Startup, and Mode 1, Power Operation, provided the inoperable channel is placed in the TRIP condition within one hour. Also, one additional channel may be BYPASSED for up to two hours for surveillance testing.

With more than one N/R level channel on one S/G inoperable, Technical Specifications require that the minimum number of channels be made operable within one hour or that the Unit be placed in at least Hot Standby within the following six hours and in Hot Shutdown within the following six hours.

DESCRIPTION OF INCIDENT:

On October 14, 1987, at 2050 hours, S/G 2C N/R Level Channel 4 was declared inoperable due to it's reading being greater than 4% different from the other three channels. The channel was placed in the TRIP condition within one hour as required and a Work Request was originated to repair the channel. On October 21, 1987, Instrumentation and Electrical (IAE) Duke Power personnel installed a new transmitter in the Channel 4 loop. The channel remained out of tolerance and the original transmitter was reinstalled. The channel was left in the TRIP condition. The problem was suspected to be inside lower Containment in an area inaccessible due to very high radiation levels during power operation. The work request was placed on the Unit Trip List to be performed during the next Reactor shutdown.

On October 29, 1987, S/G 2C N/R Level Channel 2 was recalibrated due to it's reading being greater than 2% different from the remaining operable channels. On November 1, 1987, S/G 2C N/R Level Channel 2 analog/ channel operational test was completed satisfactorily. On November 3, 1987, at 0826 hours, S/G 2C N/R Level Channel 2 was declared inoperable due to it's reading being greater than 4% different from the remaining operable channels. A Work Request was originated to repair the channel. With two out of four inoperable S/G level channels Technical Specification 3.0.3 was applied. IAE personnel confirmed that the Channel 2 Transmitter was not malfunctioning and that the problem was suspected to be inside lower Containment. At 0926 hours, Control Room personnel initiated shutdown of the Unit by commencing power reduction from 93% power. The NRC was notified of the forced Unit shutdown by the Shift Supervisor at 0927 hours.

At approximately 1100 hours, the Channel 2 level reading became increasingly erratic. IAE personnel were notified to report to the Control Room to BYPASS the channel to permit an orderly Unit shutdown. IAE personnel were preparing to place the channel in BYPASS when at 1146:31:173 hours, it spuriously spiked above the P-14 setpoint. The spike caused an automatic Main Turbine trip, CF Isolation and CF Pumps protective trip. At 1146:32 hours, M/D CA Pump 2B started automatically. M/D CA Pump 2A was already operating to cool CA check valves, S/G Blowdown (BB) and Sample (NM) Isolation were also initiated automatically. At 1146:34 hours, the Main Condenser (EIIS:COND) (S/B) and Atmospheric (SV) Steam Dump valves began to open automatically due to the Main Turbine trip. The S/G levels then began to decrease rapidly.

The Shift Supervisor ordered a manual Reactor trip due to imminent S/G low low levels. At 1146:39:577 hours, the CRO manually tripped the Reactor. At 1146:39:631 hours, Reactor Trip Breaker (EIIS:BRK) RTA tripped open and at 1146:39:643 hours, Reactor Trip Breaker RTB tripped open. Between 1146:51:057 hours and 1146:52:421 hours, all four S/G

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low low level Reactor trip alarms occurred. At approximately 1147:03 hours, the steam supply valves to the T/D CA Pump opened automatically. The Turbine's T&T valve tripped closed within a few seconds of the Turbine's steam supply valves opening. However, the CROs were not aware of the trip. The T/D CA Pump steam supply valves were closed by the CRO as allowed by procedure at approximately 1150 hours, to limit the Unit cooldown rate. The M/D CA Pumps were utilized to restore S/G levels to normal and to cool down the Unit. Between 1205:02:411 and 1206:11:031 hours, all four S/G low low level Reactor trip alarms cleared as S/G levels were being restored. S/G 2C N/R Level Channel 2 continued to intermittently spike causing sporadic P-14 signals. Unit cool down continued normally and Mode 4, Hot Shutdown, was entered at 2013 hours.

During the Trip Review, Performance personnel discovered that the T/D CA Pump had tripped. A Work Request was originated to investigate and repair the cause of the trip. The T&T Valve Not Open annunciator was still energized in the Control Room. However, Control Room personnel were not aware that it had been received. The T&T valve was then opened from the Control Room indicating it had not closed due to a mechanical overspeed trip.

On November 5, 1987, S/G 2C N/R Level Channels 2 and 4 were declared operable after steam leaks on their reference leg vent lines were stopped by installing compression fitting caps on the ends of the vent tubing. Unit heatup was then commenced and Mode 3, Hot Standby, was entered at 0415 hours.

On November 6, 1987, Mechanical Maintenance (MM) personnel working under a Work Request discovered the T/D CA Pump Governor (EIIS:65) valve travel adjustment to be in excess of 3/4 of an inch. The manufacturer's technical manual specified the setting should be 5/8 of an inch. MM personnel then readjusted the valve travel to the recommended setting.

IAE personnel tested the Turbine's electrical overspeed trip circuit which was found to be operating properly. The T/D CA Pump was tested by starting it, once manually from the Control Room, and four times by generating a fast start signal. The Turbine started properly and continued to operate each time. The T/D CA Pump was then declared operable.

On November 7, 1987, at 0030 hours, the Unit entered Mode 2, Startup, and at 2337 hours, the Unit entered Mode 1, Power Operation.

CONCLUSION:

This incident is attributed to the S/G 2C N/R Level Channel 2 and 4 reference leg vent line isolation valves leaking past the seat. The steam leakage resulted in the reference legs not being maintained full of water as required for accurate and stable S/G water level indication. The steam leakage was discovered after Unit shutdown inside lower Containment in an area inaccessible during power operation due to very high radiation levels.

The reference leg vent line steam leaks were stopped by installing compression fitting caps on the ends of the vent tubing. The associated work requests have been left active to repair the leaking isolation valves during an upcoming refueling outage. Additionally, the remaining S/G reference leg vent lines will be checked prior to Unit cool down at the beginning of the refueling outage to identify leaks and facili..te isolation valve repair as necessary.

A Station Problem Report was originated and approved to allow installation of compression fitting caps on all remaining Unit 2 S/G level reference leg vent lines. This modification is scheduled to be accomplished during an upcoming refueling outage. The Unit 1 S/G level reference legs do not have vent lines with isolation valves, therefore, this modification is not required on Unit 1. Primary System vent lines of similar design were originally installed with compression fitting caps.

The exact cause of the T/P CA Pump trip has not been identified due to the lack of alarm recorder inputs for the Turbine's speed indication and overspeed trip mechanisms, and since no duplication of the trip was attempted prior to adjustments being performed. The electrical overspeed trip circuit was subsequently tested by IAE personnel and found to be operating properly. However, the possibility that a spurious intermittent trip signal occurred cannot be ruled out. The Turbine governor valve travel adjustment was found set to allow the valve to open further than specified in the manufacturer's technical manual. Also, the manufacturer's recommended preventive maintenance (PM) to ensure smooth governor valve operation was not included in the Station's PM program for the Turbine. The excessive governor valve opening combined with increased governor valve/linkage binding over time may have caused an actual overspeed condition.

The governor valve travel adjustment was subsequently set to the manufacturer's recommended setting. The Turbine was then started and it operated properly (five starts were performed). The recommended governor valve/linkage PMs will be added to the Station's PM Program.

There have been two previous incidents involving Engineered Safeguards Feature (ESF) Actuations due to spurious S/G high high level alarms (see LER 414/86-01 and LER 414/86-10). LER 414/86-01 involved a spurious S/G high high level signal with the Unit in Mode 4, the cause could not be determined. LER 414/86-10 involved another spurious S/G high high level alarm with the Unit in Mode 3 and one S/G level channel in TRIP. The cause was determined to be a reference leg vent line isolation valve leaking steam past the seat, the valve was further closed to stop the leak. Due to these previous similar incidents, this is considered to be a reoccurring event.

There have been no previous LERs involving the failure of a T/D CA Pump to start and continue to operate following an autostart signal. Therefore, the T/D CA Pump trip is not a recurring event.

An NPRDS search revealed 10 failures of level transmitters associated with Reactor Protection due to leakage, out of 559 applications. The failures of S/G 2C N/R Level Channels 2 and 4, are NPRDS reportable as Reactor Protection System (EIIS:JC) level transmitter failures due to released leakage. Since no malfunction of the transmitters themselves occurred, the manufacturer and model numbers will not be entered. The leaking vent line isolation valves are Model V28S manufactured by Dragon Valve Inc. The valve leakage is not NPRDS reportable as a valve failure.

An NPRDS search also revealed 10 failures out of 13 applications of T/D Auxiliary Feedwater Pumps attributed to governor related malfunctions. Although the exact cause of this incident is not known, the failure will be reported to NPRDS. The Turbine is a model GS 2N, manufactured by Terry Corporation, with a Woodward Governor Company Model PG-PL governor.

CORRECTIVE ACTION:

IMMEDIATE

S/G 2C N/R Level Channel 2 was declared inoperable.

SUBSEQUENT

(1) Control Room personnel commenced Unit shutdown.

(2) Control Room Operator manually tripped the Reactor.

(3) IAE personnel repaired the steam leaks on the S/G 2C reference leg vent lines.

(4) MM personnel adjusted the T/D CA Pump governor valve travel to the manufacturer's recommended setting.

PLANNED

(1) A linkage adjustment dimension will be added to common procedure for the T/D CA Pump Turbine governor.

(2) Investigation of the overspeed occurrence on the T/D CA Pump will be continued.

(3) S/G and Pressurizer cavities will be surveyed for steam leaks.

(4) Governor linkage adjustments will be compared between Unit 1 and Unit 2. Measurements will be taken and evaluation made as to correct set up.

(5) The Manufacturer's recommended PMs will be incorporated, to ensure smooth governor valve/linkage operation, into the Station's PM Program.

(6) A Station Problem Report will be originated to modify the T/D CA Pump Turbine's Electrical Overspeed Trip indicating light circuits to lock in following an electrical overspeed trip.

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(7) CRO actions regarding Steam Dump valve control and CA throttling for appropriateness will be evaluated, and any concerns generated from the evaluation will be discussed with the Shift Supervisors.

SAFETY ANALYSIS:

The Reactor was manually tripped after a loss of CF in anticipation of a S/G low low level Reactor trip. The Reactor trip breakers opened within 54 milliseconds of the Reactor Trip signal. All of the control rods fell to the bottom of the core, reducing power to decay heat level. Prior to the Reactor trip, the P-14 initiated a Main Turbine trip, a protective trip of both CF pumps, and CF Isolation. Motor Driven CA Pump 2A was operating prior to initiation of the transient. Motor Driven CA Pump 2B auto-started upon loss of both CF pumps, and the T/D CA Pump started upon low low level in 2 out of 4 S/Gs.

Average Reactor Coolant System temperature momentarily increased 3 degrees F upon P-14, and then decreased to a minimum value of approximately 550 degrees F post-trip. Average Reactor Coolant System temperature stabilized at 555 degrees F 30 minutes post-trip, 2 degrees F from the no-load target of 557 degrees F. Pressurizer pressure momentarily increased approximately 55 psi to a maximum of 2290 psig upon P-14, and then decreased to a minimum value of approximately 2060 psig post-trip. Pressurizer (EIIS:PRZ) pressure stabilized at 2230 psig 30 minutes post-trip, 5 psi from the no-load target of 2235 psig. Pressurizer level momentarily increased approximately 4% to a maximum value of 48% upon P-14, and then decreased to a minimum value of approximately 22% post-trip. Pressurizer level stabilized at 30% 30 minutes post-trip, 5% from the no-load target of 25%. Steam pressure increased approximately 60 psi to a maximum average value of 1095 psig upon P-14, and then decreased to a minimum of 1010 psig post-trip due to steam dump to condenser. Average steam pressure was trending upward at a value of 1055 psig 20 minutes post-trip. A review of the Nuclear Steam Supply System (NSSS) logs indicates that steam pressure stabilized at an average value of 1093 psig within approximately one hour post-trip, 2 psi from the no-load target of 1095 psig. S/G wide range level decreased slightly to an indicated minimum level of 51% post-trip, and then stabilized at approximately 70% indicated level 30 minutes post-trip.

The Control Room Operators acted to limit post-trip cooldown and maintain Reactor Coolant System inventory by swapping from the 75 gpm to the 45 gpm letdown orifice, closing the Main Steam Header Blowdown Control valves, and manually increasing charging flow. All three banks of the Steam Dump to Condenser valves and two Atmospheric Steam Dump valves automatically opened immediately following initiation of the transient, and the Condenser Steam Dump valves were manually opened by the Operators several minutes post-trip to remove decay heat. CA was available and used to quench steam pressure and maintain S/G inventory. The Reactor Coolant was 82 degrees F subcooled at the time of minimum Reactor Coolant System pressure. Adequate core heat removal was available and maintained at all times.

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. This event is

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bounded by the "Turbine Trip" transient as described in section 15.2.3 of the Catawba Final Safety Analysis Report. As stated in section 15.2.3, "Reactor trip is actuated by the first Reactor Protection System trip setpoint reached. No credit is assumed in this analysis for a Reactor trip due to a Turbine trip". Reactor power was 59% at the time of Turbine trip (less than 69%, P-9 interlock); therefore, an automatic Reactor trip was not initiated upon Turbine trip. However, manual Reactor trip was initiated approximately 11 seconds prior to an automatic trip on S/G low low level.

The T/D CA Pump started and then tripped due to a possible electrical overspeed signal. An overspeed trip of this pump is detectable from the Control Room. Had operation of the T/D CA Pump been desirable during this event, it is likely that manual control of the pump speed could have been maintained by manipulation of the Turbine's Trip and Throttle valve. Normal power was available to M/D CA Pump 2A and normal and emergency power were available to M/D CA Pump 2B during this event. It also should be noted from section 15.2.3 of the Catawba FSAR that "no credit is taken for CA flow since a stabilized plant condition will be reached before CA initiation is normally assumed to occur". For this event, adequate CA flow was available and maintained to remove decay heat. Therefore, this event is well within limits delineated in the FSAR. Integrity of the fuel cladding, Reactor Coolant System, and the Containment structure was maintained at all times.

This event is reportable pursuant to 10 CFR 50.72, Sections (b)(1)(i)(A) and (b)(2)(ii) and 10 CFR 50.73, Sections (a)(2)(i)(A) and (a)(2)(iv).

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

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

December 3, 1987

Document Control Desk
U. S. Nuclear Fegulatory Commission
Washington, D. :. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/87-29

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/87-29 concerning a Manual Reactor/Auxiliary Feedwater Pump Turbine Trip Following Turbine Trip/Main Feedwater Isolation due to a Spurious High Steam Generator Level signal. 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

JGT/1055/sbn

Attachment

xc: Dr. J. Nelson Grace
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

M&M Nuclear Consultants
1221 Avenue of the Americas
New York, New York 10020

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

American Nuclear Insurers
c/o Dottie Sherman, ANI Library
The Exchange, Suite 245
270 Farmington Avenue
Farmington, CT 06032

Mr. P. K. Van Doorn
NRC Resident Inspector
Catawba Nuclear Station

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