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): Catawba Nuclear Station, Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 4 1 4

PAGE (3): 1 OF 0 9

TITLE (4): Manual Reactor Trip Because Of Imminent Low-Low Steam Generator Level Due To A Failed Switch

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 6	2 0	8 8	8 8	0 2 3	0 0	0 7	2 0	8 8	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

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)	[ ] 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	
	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S J	H S	C 7 7 0	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 fifteen single-space typewritten lines) (16)

On June 20, 1988, at 1008:15 hours, a shutdown of Main Feedwater Pump Turbine Lube Oil (LF) Pump 2A1 caused Main Feedwater Pump Turbine (CFPT) 2A to automatically trip on low bearing oil pressure. LF Pump 2A2 and the Emergency Bearing Oil Pump automatically started upon sensed low oil pressure. The output pressure from these pumps did not build quickly enough to prevent the trip of the CFPT. The loss of one CFPT automatically initiated a runback of the Main Turbine. The Condenser Steam Dump valves opened to control the primary coolant temperature as designed. Steam Generator (S/G) levels decreased due to decreased feedwater flow. The Control Room Operators attempted to reset the tripped CFPT but could not clear the interlocks in time to prevent decreasing S/G levels towards the Reactor Trip setpoint in all S/Gs. The Operators tripped the Reactor manually due to imminent automatic trip on low low S/G level. Unit 2 was at 98% power prior to this incident.

This incident has been attributed to a failed Cutler Hammer Model E30JY8 pushbutton switch. The failure of the latching mechanism of the switch allowed the contacts to open, which maintain LF Pump 2A1 in operation, and caused the pump to trip. The faulty switch was replaced.

Implementation of modifications to the LF Pump start circuitry will be pursued. The need for an accumulation device on the oil system to minimize the effects of an oil system transient will be evaluated. Implementation of a more reliable and responsive Feedwater and Turbine control systems are also being pursued. The slow response of the Feedwater Pump 2A Recirc valve will be evaluated. This event had no effect on the health and safety of the public.

☆U.S.GPO:1984-0-454-481/15750

BACKGROUND:

The Main Feedwater [EIIS:SJ] (CF) Pumps [EIIS:P] supply feedwater to the Steam Generators [EIIS:SG] (S/Gs) at the proper pressure to maintain S/G level within the appropriate operating ranges. The CF Pump Turbines [EIIS:TRB] (CFPTs) are interlocked to trip both the Turbine Stop and Control valves closed on receipt of a CFPT or a CF Pump low bearing oil pressure signal. Additionally, the signal is interlocked to close the CF Pump Recirc valve [EIIS:V] to prevent windmilling the pump with low bearing lube oil pressure and causing damage to the pump or Turbine. This windmill protection feature is also interlocked in such a manner that if both CF pumps are off, and a low lube oil bearing pressure signal occurs while the same pump's suction valve is open, all Hotwell and Booster Pumps for the Unit will automatically trip.

The CF Pump Recirc valves are also interlocked to open upon the receipt of a CF Pump Low Suction Flow signal of 4500 gpm decreasing, or upon the receipt of a Reactor Trip [EIIS:RCT] signal. The valves open to provide a flow path for the CF Pumps to prevent low suction flow CFPT trips, and to minimize feedwater transient severity on a Reactor trip.

The Main Turbine Generator [EIIS:TG] is interlocked to reduce (runback) generator power output to 70% of rated power on trip of a CFPT while operating above 80% power. This runback rate is 20% per minute.

DESCRIPTION OF INCIDENT:

On June 20, 1988, at 1008:15:659 hours, CFPT Lube Oil (LF) Pump 2A1 tripped due to the failure of a Cutler Hammer pushbutton control switch [EIIS:HS], type E30JY8. As the pump stopped, oil pressure in the LF System decreased and initiated autostarts of the Standby LF Pump 2A2 and the LF Emergency Bearing Oil Pump. As these pumps were starting, bearing lube oil pressure could not recover fast enough to prevent reaching the trip setpoint. At 1008:16 hours, CFPT 2A tripped. At this time, it is likely that CF Pump 2A Recirc Valve, 2CF6, received an automatic signal to close for windmill protection due to low bearing oil pressure. The Turbine Generator received a signal to runback on loss of one CFPT.

At 1008:16:183 hours, the idle Condensate Booster Pump started and at 1008:16:187 the idle Hotwell Pump started on the trip of a CF Pump. At this time, the low lube oil pressure signal to the CFPT 2A cleared.

Because of the method in which the Turbine Generators (T/G) are controlled in the "Throttle Limiting" mode, the "Demand" runback circuit did not cause the T/G to decrease load until 1008:35 hours. At 1008:37 hours, the CF Pump 2A Recirc Valve, 2CF6, started to open. At this time it is believed that the CF Pump Suction flow interlock initiated the automatic opening of the 2CF6 at 4500 gpm decreasing.

As temperature increased in the primary coolant, the Condenser Steam Dump valves started opening to reduce the primary system Tave.

Despite the opening of 2CF6, suction flow to CF Pump 2A decreased to the 3000 gpm trip setpoint at 1009:18:155 hours. As T/G runback continued, level in the S/Gs and in Heater Drain Tank 2C1 decreased. At 1009:45 hours, level in the drain tank reached the low level trip setpoint and the Heater Drain Pump 2C1 tripped, further decreasing suction pressure to the Feedwater Pumps.

Control Room Operators (CROs) were trying to reset CFPT 2A during this period of time but could not because of the low suction flow trip. Their attempts to clear this interlock by manipulation of the Control Room manual loader for 2CF6 and establish more flow through CF Pump 2A were not successful.

Levels in all S/Gs were decreasing rapidly at this point and at 1009:56:773 hours, level reached the alarm setpoint for S/G 2D. Because the levels were not recoverable, the CRO manually tripped the Reactor.

As S/G levels continued to decrease, the Motor Driven Auxiliary Feedwater [EIIS:BA] (CA) Pumps automatically started on 2 of 4 channels low level in S/G D, and began to supply feedwater to the S/Gs. Approximately one second later, at 1009:59:431 hours, S/G B low level was reached and the Turbine Driven CA Pump automatically started on 2 of 4 S/Gs having low level.

Main Steamline pressure trended upward and at 1010:07 hours, the PORV for S/G 2C, 2SV7, opened.

As the runback of CFPT 2B continued, flow increased through CF Pump 2A and at 1010:07:413 hours, the suction flow trip cleared.

Temperature in the primary coolant decreased and at 1010:34 hours, Feedwater Isolation was automatically initiated on Reactor trip coincident with low Tave.

To minimize the rate of cooldown, the CROs secured the Turbine Driven CA Pump.

As S/G levels increased, CROs throttled CA discharge flow control valves and secured unnecessary Hotwell and Booster pumps.

The Unit was stabilized in Mode 3, Hot Standby, and all S/G levels returned to normal.

At 1104 hours, Operations personnel started to realign the valves affected by the Blowdown Isolation signal.

At 1110 hours, CA Pump 2B was secured by Operations personnel. At 1319 hours, Operators secured CA Pump 2A. Operations personnel started realignment of valves which were affected by the Feedwater Isolation signal at 1329 hours.

CONCLUSION:

This incident has been attributed to the failure of the Cutler Hammer Model E30JY8 pushbutton switch for LF Pump 2A1. The failure of the lactchin mechanism of the switch allowd the contacts to open and caused LF Pump 2A1 to trip. The pump trip resulted in reduced bearing lube oil pressure and subsequently tripped CFPT 2A.

A review of the NPRDS Database showed that no previous failures of Cutler Hammer Model E30JY8 pushbutton switches have been reported. Four previous incidents of failures of the Cutler Hammer E-30 generic model have occurred. Considering the extensive applications in use, the failures are not considered to be excessive.

A review of previous occurrences in which Reactor Trips at Catawba occurred due to the loss of a Main Feedwater Pump revealed nine previous incidents. In one of these (LER 414/87-11) the inadvertent loss of an LF Pump occurred and the CF Pump tripped, even though the alternate LF pump automatically started. During the subsequent detailed investigation, it was determined that the design of the LF System would require modification for the second pump to automatically start in time to prevent a CFPT trip on low oil pressure. Orifices were installed to delay the trips (which worked), but large viscosity changes in the oil, due to temperature changes, resulted in an unacceptable time delay for tripping. In all cases, however, automatic starting of the second pump at the instant of the loss of the first pump avoided a CFPT trip. Although the trip was avoided, the subsequent perturbation of the oil system caused the CFPT control valves and stop valves to cycle. This could cause a fluctuation in Feedwater pump speed and a subsequent Feedwater system transient. The addition of an accumulator in the oil system was suggested. Both recommendations were submitted in Duke Power Station Problem Report CNPR02596. The design change request was discussed with Duke Power Design personnel, General Office Maintenance personnel and Catawba Management. Due to budget and manpower constraints, the SPR was placed on the Inactive List for Implementation. However, it had been re-evaluated only one week before this incident and placed on the Active List. Nuclear Station Modifications (NSMs) 11081 and 20465 have been scheduled for the Unit 1 End-of-Cycle 4 and Unit 2 End-of-Cycle 3 refueling updates, respectively.

In three previous incidents, problems with CF Pump Suction Flow trips had occurred (LER 414/86-38, LER 414/86-51, and LER 414/87-27). Corrective actions for these incidents included obtaining more reliable suction flow switches and replacement of the existing control system for the recirc valves. These corrective actions have not yet been implemented.

Therefore, Reactor trips due to Condensate/Feedwater malfunctions are determined to be a recurring problem. A Feedwater Reliability Task Force has provided numerous recommendations which are to be evaluated for implementation at the earliest possible outages.

During the replacement of the Cutler Hammer E-30 pushbutton switch, it was discovered that mis-stocking of the proper pushbutton switch had occurred and none of the proper colored switches were available. The two improper switches were re-identified and restocked and materials personnel at McGuire Nuclear Station were notified. They checked their stocks and found 10 switches which were improperly sent by the distributor. These were re-identified and restocked. A Temporary Station Modification to install the generic (wrong color) pushbutton switch was performed, and the switch was installed.

During restart activities it was discovered that S/G D CF Bypass to CA Nozzle Valve, 2CA152, was leaking approximately 100 gpm. The valve actuator linkage was adjusted and the leakage was stopped. The valve was successfully retested.

During this period of time, the Unit entered a Technical Specification Action Statement for CA which was exited on completion of the valve repair.

CORRECTIVE ACTION:

SUBSEQUENT

(1) Instrumentation and Electrical (IAE) personnel replaced the faulty pushbutton switch.

(2) Mechanical Maintenance adjusted the actuator for 2CA152 valve, and the valve passed the retest.

(3) Materials personnel corrected the stock for the pushbutton switch.

PLANNED

(1) The actions of 2CF6, CF Pump 2A Recirc valve will be investigated.

(2) A transient test for 2CF6 will be conducted.

(3) The acceptance criterion for the IWV test program will be clarified.

(4) NSMs 11081 and 20465 are to be placed on the appropriate Outage and Trip schedules.

(5) Further review the failure history of the Cutler Hammer E-30 pushbutton switches will be conducted to determine if corrective actions are warranted.

(6) A program to enhance/correct the Demand mode of the Main Turbine/Generator will be established.

(7) Recurring Auxiliary Feedwater System flow and suction pressure alarms will be reviewed and recommendation provided if appropriate.

(8) Corrective actions for all inadequate response items identified in the Post-Trip Review are to be developed.

SAFETY ANALYSIS:

The CF pump trip initiated a runback from 98% full power. The Reactor was manually tripped at 75% full power in anticipation of an automatic trip on S/G low-low level. CF Isolation was automatically initiated on Reactor trip with low Tave (564 degrees F). Both Motor Driven CA pumps autostarted upon S/G D low low level, and a Turbine Driven CA Pump Autostart signal occurred approximately 1 second later upon low low level in 2 out of 4 S/Gs. The redundant steam supply valves for the Turbine Driven CA Pump, SA2 and SA5, opened within 7 seconds of the autostart signal. The Reactor trip breakers [EIIS:BRK] opened within 60 milliseconds of the manual Reactor trip and all control rods [EIIS:ROD] fell to the bottom of the core, reducing power to decay heat level.

After CF Pump trip occurred and Reactor and Turbine runback were automatically initiated, steam pressure increased from 1000 psig to 1035 psig. The steam dump to condenser valves opened to dump steam and remove core heat. Reactor Coolant System temperature followed steam pressure and increased from 589 degrees F to 591 degrees F at the time of the CF transient, and then began to decrease again just prior to Reactor trip. Reactor Coolant System temperature decreased to a minimum value of 554 degrees F post-trip, and stabilized at the no-load target of 557 degrees F within 30 minutes post-trip. Pressurizer pressure increased slightly during the CF transient, and then decreased to a value of 2200 psig at the time of the Reactor trip. Pressurizer [EIIS;PRZ] pressure decreased to a minimum value of 2014 psig post-trip, and stabilized at the no-load target of 2235 psig within 30 minutes post-trip. Pressurizer level decreased to a minimum value of 24% post-trip, and stabilized at 30% within 30 minutes post-trip, 5% from the no-load target of 25%. Steam pressure reached a maximum value of 1117 psig immediately post-trip, and stabilized at 1085 psig within 30 minutes post-trip, 5 psi from the no-load target of 1090 psig. S/Gs A, B, C, and D reached a minimum wide range indicated value (immediately post-trip) of 42%, 38%, 41%, and 39%, respectively. Steam pressure correction of these values yields actual levels of 54%, 48%, 53%, and 49% for S/G's A, B, C, and D, respectively.

With the exception of one valve (SB24) which was isolated, banks one and two of the steam dump to condenser valves opened to dump steam during the Unit runback response, limiting steam pressure to 1035 psig. With the exception of one valve (SB24) which was isolated, all three banks of steam dump to condenser valves opened immediately upon Reactor trip. S/G C PORV, SV7, also opened to mitigate the pressure increase upon Reactor trip, and remained open for 47 seconds. A Safety Parameter Display System (SPDS) heat- sink alarm was generated as all four S/G Narrow Range levels reached low-low level post-trip (heat-sink alarm generated by all four S/G levels low-low with auxiliary feedwater flow < 450 gpm), but cleared upon auxiliary feedwater delivery to the S/Gs. The auxiliary feedwater flow rate was acceptable and well above the 450 gpm minimum cumulative flow to the S/Gs as required in the Reactor Trip Response Emergency Procedure. The Reactor Coolant was 45 degrees F subcooled at the point of minimum post-trip Reactor Coolant System pressure. Adequate heat-sink was available and maintained at all times for core decay heat removal.

This event is bounded by the "Loss of Normal Feedwater Flow" transient as described in section 15.2.7 of the Catawba FSAR, due to the loss of one of the two operating CF pumps. During the Reactor and Turbine runback prior to Reactor trip, pressurizer pressure decreased (as hot leg temperature decreased) from 2235 psig to 2200 psig. This is a temporary deviation from the lower Technical Specification limit of 2222 psig, but still within the pressurizer pressure envelope considered in Chapter 15 of the Catawba FSAR. For accidents which are not DNB limited, an allowance of +30 and -42 psi was considered by summing these values with the nominal pressurizer pressure of 2235 psig to account for steady state fluctuations and measurement error. Therefore, a fluctuation and measurement range of 2193 psig to 2265 psig is assumed for analysis purposes. The "Loss of Normal Feedwater Flow" transient is not DNB limited, because forced circulation is assumed to be available throughout the duration of the postulated transient, and because the plant is tripped well before S/G heat transfer capability is reduced. The transient is limited by peak pressurizer pressure and the availability of heat sink, and is assumed to begin at 2280 psia (2265 psig). The decreased pressurizer pressure prior to Reactor trip in this event is therefore conservative and provides additional margin against an undesirable peak value of pressurizer pressure. Pressurizer pressure response was normal and as expected, and adequate heat sink was available and maintained at all times, both before and after the Reactor trip. At the lowest pressurizer pressure during the runback (just prior to the Reactor Trip) the Reactor Coolant was 22 degrees F subcooled.

It also should be noted that the FSAR Chapter 15 scenario assumes that the Reactor is at 102% full power, and that steam dump to condenser is not available. Thus, maximum heat is being generated and minimum heat is being removed prior to Reactor Trip. In this event, less than maximum heat was being generated due to the Reactor runback, and the Turbine runback and steam dump function contributed to core, and RCS pump and stored heat removal prior to trip. The maximum load change assumed for an operational transient (condition I event) is a step load change of up to 10% or a ramp load change of up to 5% per minute. In this event, the Unit ranback 23% in one minute (98% to 75%). Therefore, the Unit runback prior to the Reactor trip is not considered a separate condition I event as a precursor to a condition II event (i.e., "Loss of Normal Feedwater Flow"), but rather, the pre-trip response and the post-trip response are both taken to be part of the condition II event, "Loss of Normal Feedwater Flow", and are fully bounded by this transient.

All safety related equipment was available throughout this transient. 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. Integrity of the fuel cladding, Reactor Coolant System, and Containment structure was maintained at all times.

During the post-trip investigation, valve 2CA152, S/G D Main Feedwater to Auxiliary Feedwater Nozzle Isolation Valve, was found to be leaking. To address past operability concerns, leakage in either direction between the CF and CA systems was considered.

Case 1 - CA System Operability Based on Flow Diversion of Auxiliary Feedwater to the CF system.

The CA system is designed to start automatically and supply sufficient feedwater to maintain the Reactor at Hot Standby for two hours followed by a cooldown to a temperature at which the Residual Heat Removal System can be operated. The effect of backleakage through 2CA152 would divert CA flow to S/G 2D from Motor Driven CA Pump 2B. This leakage would not affect CA operation, since check valves 2CF169 and 2CF173 upstream of 2CA152 function to prevent backleakage of flow from the CA System to the CF System. These check valves are not currently in the IWV program, but will be recommended for addition to the program per Design Study CNSD 113. Should a pipe break occur upstream of 2CA152, the resultant loss of CA flow through the break via a leaking 2CA152 is bounded by a previously analyzed condition of a faulted Steam Generator. Therefore, leakage through 2CA152 does not prevent the CA system from performing its design function.

Case 2 - CA System Operability Based on Leakage of Feedwater Flow Through Valve 2CA152.

Valve 2CA152 is identified as a Containment Isolation valve per Table 3.6-2a of Catawba Technical Specification 3/4.6.3. This valve receives a Feedwater Isolation signal to automatically close during an ESF actuation to restrict mass and energy release to Containment following a high energy line break inside Containment. The effect of leaking feedwater through 2CA152 into Containment is addressed in the discussion below.

Chapters 6 and 15 analyses in the Catawba FSAR have been reviewed to determine the effect of above normal feedwater flow due to leakage through valve 2CA152. The maximum leakage has been estimated to be 374 gpm with a 1400 psi differential pressure between the pump discharge and S/G 2D. Failure of the Feedwater Control Valve to S/G 2D (2CF55) is assumed to be the single failure and no credit is taken for feedwater pump trip or discharge valve closure, since the signals are not safety grade. Therefore, additional feedwater flow of up to 374 gpm could be present until the CF pumps coast down as a result of loss of steam to drive them.

Chapter 6: A steamline break accident is limiting with respect to peak Containment temperature. Consideration of superheated steam generation results in essentially the same peak temperature as found in the FSAR analysis. An additional 374 gpm of feedwater would reduce, if not prevent, generation of superheated steam as well as allowing the Reactor Coolant System more time to cool down before S/G tubes are uncovered. The additional mass and energy released due to the extra feedwater flow would not impact the peak Containment pressure analysis, since the limiting transient (LOCA) is unaffected by a feedwater leak and no increase in peak temperature would be realized.

Chapter 15 Feedwater Line Break: If S/G 2D is affected by the feedwater line break, feedwater would not reach the generator through the lower nozzle, but some would enter through the upper nozzle, favorably affecting the transient. If one of the other S/Gs is affected, leakage through 2CA152 would result in a larger inventory in S/G 2D, enhancing heat removal from the Reactor Coolant System and favorably impacting the transient. Therefore, the FSAR analysis is still bounding.

Chapter 15 Steamline Break: In performing the FSAR analysis, Westinghouse assumed auxiliary feedwater flow of 1845 gpm to the faulted S/G. A more recent analysis has shown the maximum feedwater flow to be 1486 gpm (including pump overspeed), which, when added to the 374 gpm leaking through 2CA152, results in 1860 gpm being added to the S/G, 15 gpm more than in the Westinghouse analysis. This 15 gpm represents less than 1% of the total flow to the S/G and is not considered to have a significant effect on the mass and energy release to Containment or on the overcooling transient. Also, since the single failure has been accounted for, two trains of safety injection are assumed operable, which would result in more boron entering the core and dampening the core, resulting in quicker shutdown of the Reactor and a less severe transient.

Chapter 15 Steam Generator Tube Rupture: For this accident the single failure is mandated to be the PORV on the affected S/G. Since the control valve (2CF55, S/G D Main Feedwater Control Valve) can now be assumed to close on receipt of the appropriate signal, the addition of feedwater through 2CA152 would be terminated less than 10 seconds into the event. This small amount of feedwater is negligible compared to the S/G water mass, but would result in less dose by further diluting the radioactive water migrating from the primary side.

The Environmental Qualification (EQ) curve will remain unaffected. Using the same logic as for the Chapter 6 Steamline Break, the onset of superheated steam generation is delayed, if not prevented; therefore, the short term part of the EQ curve is unaffected. Since a LOCA defines the EQ curve in the long term, the entire curve remains bounding.

In conclusion, appropriate sections of Chapters 6 and 15 of the Catawba FSAR have been reviewed to determine the impact of leakage through valve 2CA152. None of the analyses is significantly affected.

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

Based on the preceding analysis, it may be concluded that the health and safety of the public was not affected by this event.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

July 20, 1988

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-23

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-23 concerning a manual reactor trip because of imminent Low-Low steam generator level due to a failed switch. 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/63/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

IE22
1/1