Duke Power Company
Catawba Nuclear Station
P.O. Box 256
Clover, S.C. 29710

(803) 831-3000

DUKE POWER

June 19, 1989

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

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/89-13

Gentlemen:

Pursuant to 10 CFF 50.73 Section (a) (2) (iv) and 10 CFR 50.72 Section (b) (2) (ii), attached is Licensee Event Report 414/89-13 concerning feedwater isolation on hi-hi steam generator level during testing due to inadequate procedural precautions.

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

Very truly yours,

Tony B. Owen

Tony B. Owen
Station Manager

KEB\LER-NRC.TBO

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulator Commission
101 Marietta Street, NW, Suite 2900
Atlanta, GA 30323

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

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

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

Mr. K. Jabbour
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Mr. W. T. Orders
NRC Resident Inspector
Catawba Nuclear Station

IE22
1/1

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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 4

TITLE (4) Feedwater Isolation On Hi-Hi Steam Generator Level During Testing Due to Inadequate Procedure Precautions

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 5	2 1	8 9	8 9	— 0 1 3 —	0 0	0 6	2 1	8 9	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	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	
R.M. Glover, Compliance Manager	8 0 3	8 3 1 - 3 2 3 6

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

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 May 21, 1989, a Feedwater Isolation occurred at 1435:47 hours, as Steam Generator 2B level increased to the Hi-Hi level setpoint. Periodic testing of the Auxiliary Feedwater, Main Feedwater, and Main Turbine Interlocks was in progress, and the valve lineup to support testing had been completed at 1434:35 hours. The final valves to be opened for testing were the Main Feedwater Bypass Control valves, which allowed condensate flow to be supplied to the Steam Generators. Performance personnel were unable to then initialize the Response Time Test computer program and initiate testing prior to the increase in level. The procedure allowed the test to be performed at a Unit status where operation of the condensate system would affect Steam Generator level. Therefore, this event is attributed to inadequate procedural precautions. The test procedure will be revised so that Steam Generator levels will not be affected by condensate flow. Outage schedules will also be revised where necessary. Unit 2 was in Mode 4, Hot Shutdown, during the End-of-Cycle 2 Refueling Outage, at the time of this event.

NRC Form 366
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 9	— 0 1 3	— 0 0	0 2 OF 0 4

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

BACKGROUND

PT/2/A/4350/02E, Auxiliary Feedwater [EIIS:BA] (CA), Main Feedwater [EIIS:SJ] (CF), and Turbine [EIIS:TRB] Interlocks Periodic Test, is performed every refueling outage to verify proper response to a Hi-Hi Steam Generator [EIIS:HX] (S/G) Level signal. The test may currently be performed in Mode 4, Hot Shutdown, through Mode 6, Refueling, or no Mode. Prior to initiating the Hi-Hi level signal, the proper valve [EIIS:V] alignment is established and the Response Time Test Program on the Operator Aid Computer [EIIS:IMOD] is then initialized for data retention. A test switch [EIIS:XIS] is then actuated to initiate the Hi-Hi level signal.

EVENT DESCRIPTION

On the morning of May 21, 1989, Unit 2 was in Mode 5, Cold Shutdown, during the End-of-Cycle 2 Refueling Outage. The condensate [EIIS:SD] system was in high pressure cleanup and flow was being provided to the S/Gs through windmilling CF Pumps [EIIS:P]. Reactor Coolant [EIIS:AB] (NC) System temperature was approximately 200 degrees F with heatup in progress, and S/G pressure was negligible. Unit 2 later entered Mode 4 at 1015 hours.

At approximately 1400 hours, Performance and Operations personnel were preparing to perform Section 12.7 of PT/2/A/4350/02E. As required by the test procedure, the Feedwater Isolation valves were being opened so that response to a Hi-Hi S/G level signal could be verified. Control Room Operators (CROs) had also placed S/G levels at about 45% narrow range to prepare for the test. The CF Bypass Control Valves for all four S/Gs were the last valves to be aligned prior to test initiation. The final valve was fully opened at 1434:35 hours. After verifying opening of the valves, Performance personnel initialized the Response Time Test program on the Operator Aid Computer to verify acceptance criteria. The final valve alignments and initialization of the computer program took place in the horseshoe area of the Control Room. To initiate the test, Performance personnel would then move outside the horseshoe and actuate a test switch.

Opening of the CF Bypass Control Valves provided additional condensate flow to the S/Gs. With negligible steam pressure, S/G levels began rising rapidly. The CROs observed the increasing levels, but did not take action due to the expectation that the test would be initiated and Feedwater Isolation would occur. Before Performance personnel could initiate the test, S/G 2B level rose to approximately 77% narrow range, which actuated Feedwater Isolation on Hi-Hi level at 1435:47 hours. Response to the Feedwater Isolation was as expected with the exceptions that valves 2CA149, S/G 2A CF Bypass to CA Nozzle, and 2CA151, S/G 2C CF Bypass to CA Nozzle, did not indicate closure within the required 5.8 seconds. The valves indicated fully closed in slightly over 7 seconds. The Hi-Hi level signal cleared at 1442:35 hours, and the valve alignments were subsequently reestablished, and the test was successfully completed. During the retest, both valves successfully responded.

On May 25, 1989, 2CA149 and 2CA151 were also successfully tested, per PT/2/A/4200/13E, CA Valve Inservice Test, and indicated acceptable closure times.

CONCLUSION

The Hi-Hi S/G level was initiated following the opening of the CF Bypass Control Valves. Hi-Hi level actuated about 1 minute and 12 seconds after the CF Bypass Control Valves were verified to be fully open. Prior to initiating the test, Performance personnel were able to initialize the computer program and then move to the test location outside of the horseshoe, prior to the increase in S/G level to the Hi-Hi setpoint.

Performance personnel stated that previous testing of PT/2/A/4350/02E had been accomplished earlier in the refueling outage schedule, when the condensate system had not yet been placed in service. The Unit status required by Section 12.7 is Modes 4 through 6 or No Mode. The procedure did not provide a caution concerning the effects of condensate flow on S/G level, nor did the procedure make reference to the status of the Condensate and Feedwater Systems.

There have been no ESF actuations in the previous twelve months due to testing being performed with inappropriate systems status. Therefore, this is not considered to be a recurring event.

CORRECTIVE ACTION

SUBSEQUENT

(1) The Feedwater Isolation was cleared after S/G 2B level was decreased.

(2) The CROs reestablished the previous S/G level.

PLANNED

(1) PT/1,2/4350/02E will be revised to ensure that required system/equipment lineups will not affect S/G level.

(2) Outage schedules will be revised to reflect the revisions to PT/1,2/4350/02E.

*U.S. GPO: 1988-520-589/00070

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	8 9	— 0 1 3	— 0 0	0 4 OF 0 4

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

SAFETY ANALYSIS

The response to the Hi-Hi S/G 2B level was as designed, with the exceptions that 2CA149 and 2CA151 did not indicate expected closure times. Both valves were satisfactorily stroke tested prior to Unit restart.

PT/2/A/4350/02E is not performed with the Unit at power. During Section 12.7, the CA Motor Driven Pumps were disabled. However, Technical Specifications require alternate means of decay heat removal be available. During this event, Residual Heat Removal [EIIS:BP] Pump 2B had been secured to allow Unit heatup, but was available if additional heat removal was necessary.

The health and safety of the public were unaffected by this event.