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)
Brunswick Steam Electric Plant Unit 2	0 5 0 0 0 3 2 4	1 OF 0 4

TITLE (4) Reactor Low Level No. 1 and Inoperability of Residual Heat Removal RHR Subsystem Loop A Due to Failure of RHR Subsystem A Loop Heat Exchanger Level Control Valve E11-LV-F053A

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	2 7	8 4	8 4	0 1 4	0 0	1 2	2 7	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	3	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. Poulk, Jr., Regulatory Specialist	9 1 9 4 5 7 - 9 5 2 1

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)		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)

While attempting to establish shutdown cooling on the A loop of Residual Heat Removal (RHR) System, two separate events occurred: 1) A water hammer event damaging supports on the steam condensing line, and 2) a Reactor Protection System (RPS) trip on low vessel level. At the time of these events, the reactor was shut down and at approximately 90 psi. Following both events, the plant was restored to its normal condition.

Both events occurred when the control signal to the E11-F053A valve caused the valve to travel to its "open" position instead of the desired "close" position. There is no position indication of the F053A valve on the reactor turbine gaugeboard (RTGB). An investigation determined that the output jack was in the "high" position instead of the "low" position. This mispositioning caused the F053 controller to select the higher of two signals (RHR heat exchanger pressure/level) instead of the lower. When the controller was turned on in preparation for warming up the RHR lines prior to establishing shutdown cooling, the valve immediately went to the full open position. The first time this was attempted the water hammer occurred; the second attempt resulted in a low level scram and water hammer.

Vessel level was restored to the normal operating band, the output jack was connected to its correct position, and the damaged supports were repaired. The cause for the subject output jack being in the wrong position could not be determined.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

While attempting to place the RHR System into shutdown (S/D) cooling on the A loop at approximately 2000 on November 27, 1984, it was determined that a water hammer event occurred on the steam condensing line of the A loop of RHR. This was identified approximately a day following the event.

An inspection of the affected piping determined that of the 19 supports inspected, 15 supports required repairs. These supports are identified in Table 1 and are catagorized as follows: seven snubbers, one anchor, one strut, two fixed hangers, and four spring hangers. Four of the supports were repaired by a plant modification while the remaining supports were corrected using in-house maintenance procedures. The snubbers were satisfactorily functionally tested prior to being restored to service. The portions of the RHR piping involved were visually inspected by ASME Section XI qualified, VT-3 Level II/III inspectors. In addition, 8 of 25 welds on the line received NDE testing--7 due to their proximity to pipe anchors and the associated high stress levels and 1 which was located at the location of two broken snubbers (E11-47SS227 and 228). The examinations included one or more of the following: magnetic particle, radiographic, or ultrasonic. No evidence of piping overstress or fatigue was discovered.

Approximately five hours following the water hammer event (not identified by Operations at the time of occurrence), a warm-up of the A loop of RHR was again attempted. While the warm-up was in progress, a reactor vessel low level alarm was received. Manual and automatic shutting of the E11-F008 valve (S/D cooling suction valve) terminated the decreasing water level; however, due to the stroke time of this valve, the low level trip setpoint was reached causing an RPS trip actuation. Vessel level was restored to normal.

A review of this event revealed that the level instrument, B21-LT-N017A-1, failed to actuate at the low level 1 setpoint due to the transmitter being out of calibration. This instrument causes the following Group 2, 6, or 8 valves to isolate: E11-F009, E11-F079A, G16-F003, and G16-F019. In each case, the other isolation valves on those lines isolated their respective systems as required. The transmitter was recalibrated, verified operable, and restored to service.

Following these events, an investigation was conducted to determine the cause. This investigation determined that the E11-F053A valve was traveling to the full open position when the controller was turned on, whereas the F053A valve should remain closed. The E11-SS-R605A high/low auto selector station for the F053A valve receives a signal from the E11-LIC-R604A RHR heat exchanger level controller and the E11-PIC-R609 RCIC suction pressure controller. When the E11-SS-R605 signal jack is in the "high" position, the higher of the two signals (level/pressure) is used for control of the F053 and when in "low," the lower of the two is used. Plant procedures require that the jack be left in the "low" position. During both of these events, the E11-SS-R605 jack was in the "high" position. Therefore, when the RCIC suction pressure controller was set to auto in accordance with the procedure (OP-17) with a setpoint greater than the actual (existing) RCIC suction pressure, the controller output increased, causing the E11-F053A valve to ramp open. However, the E11-F053A valve should have remained closed until the RHR heat exchanger level controller output was subsequently increased.

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

To prevent this problem in the future, the following actions have been or will be taken:

1. The signal jack was repositioned to the correct plug.

2. Plant procedures will be revised to require stroke checks or other position verification on the F053 valves prior to opening the downstream F011 valves.

3. Engineering will evaluate the need to modify the steam condensing portion of the system and complete any modifications required.

4. Water hammer damage to the supports has been repaired.

5. Real time training in this event will be provided to licensed operators.

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TABLE 1

	SUPPORT	PROBLEM
Snubbers	2E11-47SS223	Bent paddle, bent shaft
	47SS224	Loose bolting, bent paddle
	47SS225	Broken shaft
	47SS227	Bent coupling, bent shaft
	47SS228	Broken shaft, damaged paddle
	47SS326	Bent steel, broken welds
	47SS328	Spherical bearing broken
Anchor	47A320	Bent loose anchor bolts
Fixed Hangers	47FH230	Bent hanger rod
	47FH229	Bent steel, broken welds
Struts	47PG456	Bent steel, insert sheared
Spring Hangers	47VH226	Loose bolting, spring can misaligned
	47VH457	Clamp misalignment
	47CH325	Bent hanger rod
	47CH327	Hanger misalignment

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429

December 27, 1984

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SERIAL: BSEP/84-2776

NRC Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
LICENSEE EVENT REPORT 2-84-14

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

MJP/dj/LETDJ1

Enclosure

cc: Mr. R. C. DeYoung
Mr. J. P. O'Reilly

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