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 3

TITLE (4)

Reactor Low Level Due to Misoperation of the Residual Heat Removal System

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 9	2 4	8 4	8 4	— 0 1 1 —	0 0	1 1	0 1	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

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)	
M. J. Pastva, Jr., Regulatory Technician	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
B	I P	P S X	X 9 9 9	Yes					

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 9-24-84, at 2134, an automatic trip of both Unit 2 Reactor Protection System (RPS) channels A and B with Primary Containment Isolation System (PCIS) Groups 2, 6, and 8 isolations occurred due to a reactor low water level No. 1. At the time Unit 2 was in a refuel/maintenance outage with a primary containment integrated leak rate test (ILRT) in progress. Reactor level was 177", reactor temperature was 95°F, reactor pressure was 42 psig, and reactor control rods were inserted. The B loop subsystem of the Residual Heat Removal (RHR) System was in reactor shutdown cooling. The A loop subsystem of the Reactor Core Spray (CS) System was in standby. The A loop subsystem of the RHR System and the B loop subsystem of the CS System were disabled for the ILRT.

While attempting to lower the suppression pool level, the duty Control Operator misconceived that the B loop subsystem of the RHR System was in use for suppression pool cooling. He opened the RHR System discharge valves, 2-E11-F040 and F049, to the Units 1 and 2 common Radiological Waste Control (RWC) System resulting in a flow path from the reactor vessel to the RWC System. Upon receiving the low level scram, the Control Operator realized his action and immediately closed the subject valves.

Reactor level was returned to normal within five minutes of the event by utilizing the Reactor Control Rod Drive System. The involved operator was appropriately counseled and disciplined. Licensed plant Operations personnel have completed real-time training concerning this event.

IE22
\\\

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Brunswick Steam Electric Plant Unit 2	0 5 0 0 0 3 2 4	8 4	— 0 1 1	— 0 0	0 2 OF 0 3

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

On September 24, 1984, at 2134, an automatic trip of Unit 2 Reactor Protection System (RPS) channels A and B with Primary Containment Isolation System (PCIS) Groups 2, 6, and 8 isolations occurred due to reactor low water level No. 1 (>/= +162.5 inches). At the time Unit 2 was in a refueling/ maintenance outage and an integrated leak rate test (ILRT) of the unit primary containment was in progress. The unit reactor control rods were inserted, reactor water temperature was 95 degrees fahrenheit, reactor water level was at 177 inches, and the reactor vessel was vented to the primary containment atmosphere, which was at a pressure of 42 psig due to the ongoing ILRT. The B loop subsystem of the unit Residual Heat Removal (RHR) System was in service to provide shutdown cooling of the reactor. The A loop subsystem of the unit Reactor Core Spray (CS) System was operable and in standby. For purposes of the ongoing ILRT, the A loop subsystem of the unit RHR System and the B loop subsystem of the unit CS System were disabled. In addition, the reactor level makeup sources and reject flow paths were secured for the ongoing subject test.

Prior to the event, a depressurization of the unit primary containment, in accordance with the ILRT, was in progress. The duty Unit 2 Control Operator observed that the primary containment suppression pool level was increasing. This was revealed to the Control Operator by indications shown by suppression pool level indicators 2-CAC-LI-2601-1 and 2-CAC-LI-4177. These indicators showed a level of approximately -27 inches. These indications were accompanied by appropriate Unit 2 Control Room Reactor-Turbine Gauge Board (RTGB) alarm annunciations of high suppression pool level.

The Unit 2 Control Operator, in assessing the increasing suppression pool level, mistakenly assumed the B loop subsystem of the unit RHR System was in service for the suppression pool cooling mode of operation. Consequently, he opened the RHR System discharge valves to the Units 1 and 2 common Radiological Waste Control System, 2-E11-F040 and F049. Reactor level decreased to a level of 165.8 inches, resulting in the incurred RPS trip and PCIS group isolations, which included the automatic isolation of the B subsystem of the unit RHR System. The involved Unit 2 Control Operator realized his error in opening F040 and F049 and immediately closed the valves. The unit Reactor Control Rod Drive System was placed into service to provide a source of level makeup to the reactor. Reactor level recovered to 174 inches and the RPS trip signal was reset. The incurred PCIS group isolations were reset, reactor shutdown cooling was reestablished using the B loop subsystem of the unit RHR System, and the unit primary containment was depressurized. Within five minutes of the event, a reactor level of 185 inches was established. Following depressurization of the unit primary containment, indication of the suppression pool level, as shown by level indicators 2-CAC-LI-2601-1 and 2-CAC-LI-4177, was approximately 4 to 5 inches higher than prior to depressurization. At the same time the level indication of suppression pool level, as shown by the redundant level indicators 2-CAC-LI-3342 and 2-CAC-LI-2602, had increased approximately one-half inch.

The involved Unit 2 Control Operator was appropriately counseled and disciplined concerning the event. The event has been covered by licensed plant Operations personnel through real-time training sessions.

Following this event, an investigation was performed to determine the reason for the suppression pool level indication discrepancy. Prior to this event, primary containment isolation/excess-flow check valve 2-CAC-PV-4345 was replaced. This valve isolates the variable leg sensing line piping to the respective instrument transmitters of the LI-2601-1 and LI-4177 level indicators. During the unit refueling/maintenance outage, the unit suppression chamber water volume was drained. Air became entrapped within the variable leg sensing line piping to the instrument transmitters of the LI-2601-1 and LI-4177 indicators when the suppression chamber was refilled. Air remaining in the piping following the PV-4345 replacement combined with the entrapped air in the piping resulted in the erroneous indication of suppression pool level. Filling and venting procedures for the subject sensing line piping did not allow for sufficient venting of the piping.

Appropriate fill and vent instructions were performed and the sensing line piping to the the Unit 2 suppression pool level indication instrument transmitters were filled, vented, and the indicators placed into service. Plant maintenance procedures will be appropriately revised.

CP&L

Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429
November 1, 1984

FILE: B09-13510C
SERIAL: BSEP/84-2203

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 2-84-11

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. In a letter to your office dated October 24, 1984, Serial: BSEP/84-2235, it was conveyed that this event would be reported by November 7, 1984. This report fulfills the requirement for a written report 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/clh/LETCH4

Enclosure

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

IE22
1/1