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 1	0 5 0 0 0 3 2 5	1 OF 0 3

TITLE (4)
Reactor Scram Due to High Reactor Vessel Level

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	1 8	8 4	8 4	— 0 2 6 —	0 0	1 0	1 7	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 8 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)	[ ] 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

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 9-18-84, at 1039, a Unit 1 automatic reactor scram occurred due to a main turbine trip and stop valve fast closure caused by a high reactor vessel level of >/= +208". The reactor high level resulted from a mismatch of the input signal to the reactor feedwater master level controller, 1-C32-R600, during a routine calibration of the reactor feedwater flow recorder. At the time Unit 1 was at 89% power. In addition, the unit High Pressure Coolant Injection (HPCI) System, Reactor Core Isolation Cooling (RCIC) System, Residual Heat Removal low pressure coolant injection loops A and B, and Reactor Core Spray loops A and B were operable.

During the scram recovery, reactor level, with the lowest level of >/= +122", was controlled by manual control of the HPCI System. A Group 1 isolation occurred. The RCIC System did not initiate due to operation of the HPCI System. Reactor pressure peaked at 1053 psig and was controlled by manually opening reactor safety-relief valve 1-B21-F013A.

The subject input signal mismatch resulted from a deficiency in the utilized calibration procedure, which allowed the input signal leads to the feedwater flow recorder to be lifted. This caused R600 to sense a lower than actual vessel level and the feedwater flow increased till the vessel high level occurred.

Appropriate personnel were counseled concerning this event. Similar plant procedures will be reviewed to determine if similar problems exist.

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

On September 18, 1984, at 1039, a Unit 1 automatic reactor scram occurred as the result of a Unit 1 main turbine trip and stop valve fast closure caused by a reactor vessel high level of >/= +208". The reactor high level resulted from a mismatch of the three-parameter input (reactor level, reactor feedwater mass flow, reactor main steam flow) signal to the reactor feedwater master level controller, 1-C32-R600. This occurred when the signal input leads to the reactor feedwater flow recorder, 1-C32-R607, were lifted in accordance with procedure during a routine preventative maintenance calibration of the recorder. When the subject leads were lifted, a disruption of the feed flow input signal to the controller reactor feed flow/reactor steam flow proportional amplifier, 1-C32-K602, occurred. As a result, the R600 instrument control loop sensed a lower than actual reactor vessel level resulting in increased output flow rate of the reactor feed pumps and the incurred high reactor vessel level. Immediately following the high reactor vessel level, the subject R607 signal input leads were reconnected to the instrument. At the time of this event, Unit 1 was operating at 89 percent power. In addition, the unit High Pressure Coolant Injection (HPCI) System, Reactor Core Isolation Cooling (RCIC) System, Residual Heat Removal System low pressure coolant injection loops A and B, and the unit Reactor Core Spray System loops A and B were operable.

A unit reactor scram recovery was carried out in accordance with plant procedures. During the scram recovery, the lowest incurred reactor level was low level No. 2 (>/= +122"). A primary containment Group 1 isolation occurred and the reactor recirculation pumps, A and B, automatically tripped. The unit High Pressure Coolant Injection (HPCI) System received an automatic initiation signal; however, the unit Reactor Core Isolation Cooling (RCIC) System did not receive an automatic initiation signal. At the time, the Unit 1 Control Operator had already manually started the HPCI System and was controlling the rate of HPCI System makeup to the reactor vessel inventory. The HPCI System was utilized throughout the scram recovery to maintain and control reactor vessel level. It is believed the RCIC System actuation instrumentation did not sense the incurred reactor low level No. 2 as a result of the HPCI System manual initiation. The instrument setpoints for the RCIC System actuation instrumentation were subsequently verified to be within specifications. Reactor pressure, which peaked at 1053 psig, was controlled by manually opening Unit 1 reactor safety-relief valve 1-B21-F013A and through operation of the unit HPCI System. During opening of 1-B21-F013A, sonic probe and tail pipe temperature position indications of the valve were operable. An assessment of the safety consequences and implications of this event determined there are no reasonable and credible alternative conditions which would have been more severe.

This event resulted from performing an in-field calibration of the reactor feedwater R607 flow recorder. The procedure utilized to perform the routine preventative calibration of the recorder provided for an in-field or a bench calibration of the instrument.

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

During the initial preparation of the subject procedure, it was not recognized that lifting the signal input leads to the R607 recorder will result in a disruption of the input signal to the reactor feedwater master level R600 controller.

As a result of this event, a review of other appropriate plant procedures will be conducted to identify any additional procedure deficiencies which may have the potential consequence to cause similar plant transients. Until completion of the subject review and any required procedure changes, the utilization of these procedures will be administratively deferred on a case-by-case basis until the procedures are technically adequate. Appropriate plant Instrumentation & Controls (I&C) personnel were instructed to be made aware of the identified procedural deficiency involving this event and the possibility of other similar procedural problems which may have the potential to result in plant transients.

CP&L

Carolina Power & Light Company

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

October 17, 1984

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SERIAL: BSEP/84-2156

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-84-26

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

RMP/clh/LETCH4

Enclosure

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

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