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 Resulting From False Recirculation Loop FLow Signals

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 8	0 1	8 4	8 4	0 1 4	0 0	0 8	3 1	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 9 4

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 9 457-9521

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
X	B D	P S V	A 4 9 7	Yes					

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 August 1, 1984, at 1417, a Unit 1 automatic reactor scram occurred due to a reactor average power range monitor upscale trip initiation of the Reactor Protection System. The upscale trip resulted from erroneous signals of decreasing flow spikes, occurring simultaneously in both reactor recirculation system loops. At the time Unit 1 was at 94.6% power with a planned increase to rated power. In addition, the unit High Pressure Coolant Injection System was out of service pending periodic testing.

During the scram recovery, reactor level, with the lowest recorded value of 142.8", was controlled by the Reactor Core Isolation Cooling System. A Group 1 isolation occurred. Reactor pressure, which peaked at 997 psig, was controlled by manual opening of reactor safety-relief valves 1-B21-F013A, E, J, and B. Following the Group 1 isolation, main steam line isolation valve 1-B21-F022A did not close automatically or manually.

The subject erroneous flow spike signals were induced into the unit recirculation loop flow instruments due to electronic keying of two-way radios in use during periodic testing in the vicinity of the loop flow instrumentation racks in the unit Reactor Building.

By September 14, 1984, various types of plant communication radios will be electronically keyed in the vicinity of Unit 2 instrument racks to determine if Control Room instrumentation is adversely affected. Unit 2 is currently shut down for a planned maintenance outage.

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

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

On August 1, 1984, at 1417, a Unit 1 reactor scram was automatically initiated by the Reactor Protection System (RPS) due to an instrument upscale trip of the reactor Average Power Range Monitor (APRM) System. At the time Unit 1 was operating at 94.6% power with a planned increase to rated power in progress. In addition, the unit High Pressure Coolant Injection System was out of service pending the performance of periodic testing.

A unit scram recovery was carried out. Reactor level, with the lowest recorded value of 142.8", was controlled through use of the Reactor Core Isolation Cooling System (RCIC). A Group 1 isolation occurred, as per design, when the reactor pressure decreased to the low pressure setpoint while the unit mode switch was in Mode 1. Reactor pressure, which peaked at 997 psig, was controlled by manual opening of the unit reactor safety-relief valves 1-B21-F013A, E, J, and B. Following the Group 1 isolation, the Unit 1 Control Operator discovered that inboard main steam line isolation valve (MSIV) 1-B21-F022A did not automatically close. An attempt to manually close the valve proved unsuccessful. An assessment of the safety consequences and implications of this event determined there are no reasonable and credible alternative events which would have been more severe under these conditions.

The subject APRM System upscale trip resulted from the reactor recirculation loop flow instrumentation receiving erroneous input signals. The erroneous input signals caused the instruments to sense simultaneous decreasing flow spikes in each reactor recirculation loop (A and B). This resulted in an automatic reduction of the APRM System high reactor power scram setpoint to less than the actual reactor power, thereby causing the APRM System upscale trip. The cause of the erroneous input signals to the recirculation loop flow instrumentation was electronic keying of two-way radios in use in the immediate vicinity of the subject instrumentation in the unit Reactor Building. Plant Auxiliary Operators were using the two-way radios in the performance of an annual periodic test (PT) of the Reactor Building fire protection sprinkler systems, PT-35.12.11.1.

Following the Unit 1 scram recovery, an investigation was conducted to determine why inboard MSIV 1-B21-F022A did not close. The three-way ac/dc air operator actuation solenoid pilot valve for F022A apparently failed and was continuously sending an air signal to the four-way pilot valve. With electrical power to both of the ac and dc pilot valve solenoids, resulting from either a Group 1 isolation or a manual command to close, the three-way ac/dc solenoid pilot valve should have actuated to remove this air signal. However, the investigation revealed the subject air signal was still present. The F022A three-way ac/dc solenoid pilot valve, ASCO part number ER8320A183E, was replaced and the removed component was subsequently bench-tested.

No evident signs of failure were noted during testing of the three-way solenoid pilot valve.

An inspection of the removed three-way ac/dc solenoid pilot valve revealed an outward discoloration of the pilot valve body. Nothing was found during inspection of the valve which may have prevented it from operating properly.

1-B21-F022A was satisfactorily tested and returned to service.

Subsequent testing was performed on Unit 2 which supported the determination that the failure of the Unit 1 MSIV to close was the result of a failure of the ac/dc three-way solenoid valve.

As a result of this event, various types of communication radios utilized in both units will be electronically keyed in the vicinity of the Unit 2 instrumentation racks in the Reactor Building to determine if Control Room instrumentation is adversely affected. This testing of plant communication radios will be completed by September 14, 1984. Following this testing, signs prohibiting the use of plant communication radios within specific identified plant areas will be appropriately posted in those areas. In addition, plant Engineering will be requested to evaluate the apparent failure of the MSIV F022A solenoid pilot valve to determine applicable corrective action.

Form 244

CP&L

Carolina Power & Light Company

Company Correspondence

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

August 31, 1984

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SERIAL: BSEP/84-1905

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 1-84-14

Gentlemen:

In accordance with Title 10 tc 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/sdl/LETCH2

Enclosure

cc: Mr. J. P. O'Reilly

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