CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461-0429

JAN 13 1997

SERIAL: BSEP 97-0010
10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-96-017

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Mark Turkal at (910) 457-3066.

Sincerely,

W. Levis

W. Levis, Director - Site Operations
Brunswick Nuclear Plant

SFT/sft

Enclosures
1. Licensee Event Report
2. Summary of Commitments

cc: Mr. L. A. Reyes, Regional Administrator, Region II
Mr. D. C. Trimble, NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable R. Hunt, (Acting) Chairman - North Carolina Utilities Commission

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Enclosure
List of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
1. Those procedures with the potential to cause ECCS actuations from a single contact closure will be identified and revised to provide specific warning statements prior to performing critical steps and provide requirements for independent review of equipment configuration prior to restart of test activities after problems have been encountered.	3/1/97
2. A training module will be developed and incorporated into the existing Maintenance ECCS training course to enhance technician knowledge and understanding of the effects of test equipment misalignment.	3/31/97
3. The Maintenance surveillance test procedures will be revised to delineate those which are inappropriate for utilizing Simpson model 260 VOMs.	12/18/97

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 1	05000325	1 OF 4

TITLE (4)

Invalid Loss Of Coolant Accident Logic Actuation

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 NAME	DOCKET NUMBER
12	13	96	96 --	17 --	00	01	10	97	BSEP - Unit 2	05000324
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 01

POWER LEVEL (10): 95

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	X 50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor, Sr. Analyst - Licensing	(910) 457-2178

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 15 single-spaced typewritten lines) (16)

On December 13, 1996, with both units operating at rated power, while performing a maintenance surveillance test, an invalid Unit 1 Division 1 Loss Of Coolant Accident (LOCA) initiation signal was received. Plant systems responded as designed. This initiation signal resulted in the automatic start of Emergency Diesel Generators 1, 2, 3, and 4; automatic start of the Unit 1 Core Spray Pump 1A; automatic start of the Unit 2 Nuclear Service Water Pump 2A; a Unit 1 Group 10 Division 1 actuation (pneumatic valves to primary containment 1-RNA-SV-5261 and 1-RNA-SV-5262 closed); the closure of the Unit 1 Reactor Building Closed Cooling Water Heat Exchanger Service Water Isolation Valve, 1-SW-V106; opening of the Nuclear Service Water Header To Vital Header Isolation Valve, 1-SW-V117; and shutdown of the 1A and 1D Unit 1 Drywell Coolers. Immediately following the event, Operations directed Maintenance to secure the test procedure and implemented the necessary plant procedures to verify appropriate plant response and restore the plant to normal operation. By approximately 1045 hours, affected systems had been returned to their normal operating condition.

The invalid LOCA initiation signal was caused by a human performance error during installation of test equipment to support surveillance testing. Corrective actions include appropriate administrative action with the involved technician, briefing Maintenance I&C technicians on this event prior to performance of further field work, providing Maintenance I&C personnel management expectations for the restart of surveillance tests after problems have been encountered, restricting the use of Simpson model 260 VOMs for circuit checks specified in Maintenance surveillance tests, developing training to enhance technician knowledge of the effects of test equipment misalignment, and revising Maintenance procedures to preclude similar events. The safety significance of this event is considered minimal in that the systems required to safely shutdown the plant functioned as designed.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 1	05000325	96 --	17 --	00	2 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

TITLE

Invalid Loss Of Coolant Accident Logic Actuation

INITIAL CONDITIONS

On December 13, 1996, Units 1 and 2 were operating at rated power. Maintenance was performing 0MST-RHR21Q, RHR-LPCI, CSS And HPCI Hi Drywell High Pressure Trip Unit Channel Calibration Test.

EVENT NARRATIVE

On December 13, 1996, at 0913 hours, while performing 0MST-RHR21Q, an invalid Unit 1 Division 1 Loss Of Coolant Accident (LOCA) initiation signal was received. Plant systems responded as designed. This initiation signal resulted in the following actuations:

Automatic start of Emergency Diesel Generators 1, 2, 3, and 4
Automatic start of the Unit 1 Core Spray Pump 1A
Automatic start of the Unit 2 Nuclear Service Water Pump 2A
A Unit 1 Group 10 Division 1 actuation (pneumatic valves to primary containment 1-RNA-SV-5261 and 1-RNA-SV-5262 closed)
Closure of the Unit 1 Reactor Building Closed Cooling Water Heat Exchanger Service Water Isolation Valve, 1-SW-V106
Opening of the Nuclear Service Water Header To Vital Header Isolation Valve, 1-SW-V117
Shutdown of the 1A and 1D Unit 1 Drywell Coolers

Prior to the LOCA initiation signal, the responsible maintenance technician connected a Simpson model 260 Volt/Ohm Meter (VOM) to terminal points BB-92 and BB-94 in panel H12-P626 in accordance with the instructions provided in 0MST-RHR21Q to verify circuit voltage readings. The expected voltage reading was 125 VDC; however, the actual indicated voltage was O VDC. The technician removed the VOM test leads from the terminal points and performed a VOM resistance check. The resistance check indicated a break in the VOM circuitry. The technician noticed that the VOM circuit breaker was tripped, reset the VOM circuit breaker, and performed a VOM resistance check. With the VOM function switch in the ohms position, the technician reconnected the VOM test leads to terminal points BB-92 and BB-94 in panel H12-P626. With the VOM connected in this manner, the VOM acted as a jumper and completed the Unit 1 Division 1 Loss Of Coolant Accident (LOCA) logic circuit and resulted in the LOCA logic actuation.

The involved technician recognized that the test equipment installation had resulted in the LOCA initiation and notified Operations. Operations directed Maintenance to restore the affected system to the pre-test configuration and secure the test procedure. Operations then implemented the necessary procedures to verify appropriate plant response and restore the plant to normal operation. By approximately 1045 hours, affected systems had been returned to their normal operating condition.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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Brunswick Steam Electric Plant, Unit 1	05000325	96 --	17 --	00	3 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) in that the improper installation of test equipment resulted in an automatic actuation of an Engineered Safety Feature.

CAUSE OF EVENT

The invalid LOCA initiation signal was caused by a human performance error. The responsible technician did not verify that the volts DC function was selected prior to reinstalling the test equipment. This error is attributed to the lack of self checking.

CORRECTIVE ACTIONS

Appropriate administrative action has been taken with the involved technician.

Maintenance I&C technicians were briefed on this event prior to performance of further field work.

Expectations have been provided to Maintenance I&C personnel delineating actions to be taken when problems are encountered during performance of surveillance procedures. These actions include independent review of equipment configuration prior to restart of test activities.

In an effort to further reduce the potential for similar events involving the use of test equipment, the following enhancements to the controls for the use of VOMs during surveillance testing are being taken:

> The use of Simpson model 260 VOMs for circuit checks specified in Maintenance surveillance tests has been restricted. Use of these VOMs will require Maintenance supervisor concurrence.
>
> Those procedures with the potential to cause ECCS actuations from a single contact closure will be identified and revised by March 1, 1997, to provide specific warning statements prior to performing critical steps and provide requirements for independent review of equipment configuration prior to restart of test activities after problems have been encountered.
>
> A training module will be developed and incorporated into the existing Maintenance ECCS training course by March 31, 1997, to enhance technician knowledge and understanding of the effects of test equipment misalignment.
>
> The Maintenance surveillance test procedures will be revised by December 18, 1997, to delineate those procedures which are inappropriate for utilizing Simpson model 260 VOMs.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

SAFETY ASSESSMENT

The safety significance of this event is considered minimal in that the systems required to safely shutdown the plant functioned as designed.

PREVIOUS SIMILAR EVENTS

Previous events involving actuations related to the use of a VOM were reported in LERs 2-93-009 and 1-94-015.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Core Spray System	BM
Containment Isolation Control System	JM
Emergency Diesel Generator System	EK
Service Water System	BI