CP&L

Carolina Power & Light Company

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U.S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-92-028 SUPPLEMENT ONE

Gentlemen:

In accordance with Title 10 of 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 submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

JM Brown

J. M. Brown, Plant Manager - Unit 2
Brunswick Nuclear Plant

SFT/

Enclosure

cc: Mr. S. D. Ebneter
Mr. P. D. Milano
BSEP NRC Resident Office

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NRC FORM 366 (5/92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Units 1 and 2	05000325	1

TITLE (4)
Group 6/Reactor Building Ventilation System Isolations and SBGT Initiation For Both Units Occurred During Plant Modification Wire Lift

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	11	92	92	- 028 -	001	2	15	93	BSEP Unit 2	50-0324
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 04

POWER LEVEL (10): 00

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 and Text)
	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
Steve F. Tabor, Regulatory Compliance Specialist	(919) 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)

		X	NO	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)

Units 1 and 2 were in Cold Shutdown for a dual unit outage that started April 21, 1992. On December 12, 1992 at approximately 0936 hours, while performing a modification to install a hardened wetwell vent, a wire lift resulted in the actuation of the Containment Atmospheric Control (CAC) isolation logic. By design, the CAC isolation resulted in a loss of power to the stack radiation monitor isolation circuit which caused both Units 1 and 2 to experience an isolation of the CAC and Containment Atmospheric Dilution systems primary containment isolation valves, reactor building ventilation system isolations, and initiations of the Standby Gas Treatment systems.

Immediate corrective actions included the restoration of all affected systems to the normal standby lineup. A root cause analysis is in progress. The results of this analysis including further corrective actions will be reported in a supplement to this LER. The safety significance of this event is considered minimal since both Units were in Cold Shutdown and the affected systems responded in accordance with design. Previous similar events have been reported in LERs 1-90-025 and 1-90-027.

Supplement One to this LER includes a supplemental information section to address the results of the root cause analysis and associated corrective actions.

NRC FORM 366A (5/92) U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Brunswick Steam Electric Plant Unit 1	05000325	92	- 028 -	001	2

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

TITLE

Group 6/Reactor Building Ventilation System Isolations and SBGT Initiation For Both Units Occurred During Plant Modification Wire Lift

INITIAL CONDITIONS

Units 1 and 2 were in Cold Shutdown for a dual unit outage that started April 21, 1992.

EVENT NARRATIVE

On 12/11/92, wiring modifications in the Unit 2 Control Room Panel XU-53 were in progress in accordance with the installation instructions of the Hardened Wetwell Plant Modification, PM 92-073. At approximately 0936 hours, during performance of a step in the plant modification installation procedure which required lifting and moving a neutral connection for Terminal Block K24 to Terminal Block K22 to support the addition of a relay to the Containment Atmospheric Control (CAC) System Isolation System, relays located in Panel XU-53 unexpectedly operated. The lifted wire was immediately reterminated to its original termination point and the Operations Shift Supervisor was notified.

The lifting of the neutral connection at Terminal Block K24 resulted in the de-energization of the CAC isolation logic relay 2-CAC-63-3. By design, this resulted in a loss of power to the stack radiation monitor isolation circuit. The loss of the stack radiation monitor isolation circuit resulted in isolation of the Units 1 and 2 CAC and Containment Atmospheric Dilution (CAD) related Primary Containment Isolation System (PCIS) valves, isolation of the Reactor Building Ventilation System for Units 1 and 2, and the start of Units 1 and 2 Standby Gas Treatment (SBGT) Systems.

Following the event, Operations verified that the affected systems had responded in accordance with plant design. Upon retermination of the lifted wire to its original termination point, the isolation logic was reset and the systems were realigned to their configuration.

CAUSE OF EVENT

A root cause analysis is currently in progress. The results of this analysis will be reported in a supplement to this LER.

CORRECTIVE ACTIONS

Corrective actions to prevent recurrence of this event will be reported following completion of the root cause analysis.

SAFETY ASSESSMENT

The safety significance of this event is considered minimal in that both Units were in Cold Shutdown at the time of the event and the affected systems responded in accordance with design.

PREVIOUS SIMILAR EVENTS

NRC FORM 366A (5/92) U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant Unit 1	05000325	92	- 028 -	001	3

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

Previous similar events have been reported in LERs 1-90-025 and 1-90-027.

SUPPLEMENTAL INFORMATION:

A root cause analysis was performed to determine the causes of this event. The results of this analysis indicate that the unplanned safety system actuation resulted from inadequate modification installation instructions. The installation instructions required the lifting of a lead without addressing the impact of the wire lift. The instruction deficiency was due to an inadequate review of design changes which occurred during the final phase of the design process and a lack of sufficient independent design verification.

The initial design did not require the lifting of the wire that resulted in the unplanned safety system actuation. During a field walkdown to verify the modification design, the responsible engineer identified a discrepancy between the design drawings and the field condition. The field condition required a change in the initial modification installation instruction such that a wire lift was required. Prior to incorporating the wire lift into the installation instruction, the responsible engineer performed a review of the affected system logic. The scope of this review was deficient in that it did not adequately assess the impact of the final design on the affected system and components.

Additionally, the design verification effort failed to detect the impact of the wire lift. The verification engineer inappropriately relied on the design engineer for technical information during his review of the design.

The following corrective actions have been taken:

Nuclear Engineering Department (NED) personnel have been trained on the lessons learned from this event.

Design drawing F-09781 has been revised to reflect plant conditions.

The following additional measures will be taken to prevent recurrence:

Appropriate NED personnel will receive training on the importance of fully understanding the interaction of systems and components during the initial design process and maintaining independence during the verification process. This training will be completed by April 30, 1993.

A "lessons learned" Instrumentation and Control Design Guide will be developed by June 30, 1993. This design guide will list specific issues to be addressed during the design verification process. Training of appropriate NED personnel on the design guide will occur as part of the new procedure development process.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
PCIS	JM
CAC/CAD	IK
Reactor Building Ventilation	VA
Standby Gas Treatment	BH