CP&L

Carolina Power & Light Company

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

April 6, 1993

FILE: B09-13510C
SERIAL: BSEP-93-0051

10CFR50.73

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-324
LICENSE NO. DRP-62
LICENSEE EVENT REPORT 2-93-004

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,

J M Brown for

C. C. Warren, Plant Manager - Unit 2
Brunswick Steam Electric Plant

TMJ/

Enclosure

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

120110

JE22

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Mr. M. D. Bradley
Mr. J. M. Brown
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Onsite Licensing
SHEEC Training Reference
INPO

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

LICENSEE EVENT REPORT (LER)

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TITLE (4)
UNEXPECTED ISOLATION WHEN THE WRONG FUSE WAS PULLED DURING A CLEARANCE

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
03	07	93	93	- 04 -	00	04	06	93		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 4
POWER LEVEL (10): 000

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
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract and Text)
	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
Theresa M. Jones, Regulatory Compliance Specialist	(919) 457-2039

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 fifteen single space typewritten lines) (16)

Unit 2 was in day 320 of a dual unit outage. The Residual Heat Removal (RHR) system was operating in the Shutdown Cooling (SDC) Mode. Reactor coolant temperature was ≈ 97° F. Placing of clearance #2-93-698 required removal of the Unit 2 Reactor Recirculation system (B32), sample line inboard isolation (F019) solenoid valve control circuit fuse, 2-A71-F13. The tag sheet and special instructions located the fuse at position F14 (i.e., 14 positions from the top of the fuse block). The Senior Reactor Operator (SRO) and the Reactor Operator (RO) discussed the requirements for pulling the fuse and directed the RO to place the clearance. The RO found what he believed to be fuse 2-A71-F13 located at position F14 and removed it at 2224. The fuse was actually 2-A71B-F15, which supplies power to the relay logic that initiates an isolation signal for the RHR SDC Inboard Suction Throttle valve. The suction valve closed when the fuse was removed, as designed. The SRO directed the RO to install the fuse and SDC was restored at 2400. The cause of this event was determined to be inadequate labeling of the fuse block and failure of the RO to count to the 14th fuse. The corrective action is to improve the labeling of back-panel fuse blocks. This event had minimal safety significance. The loss of SDC for one hour and 36 minutes resulted in an approximate one degree F increase in reactor coolant temperature. The decay heat load would have been greater had the reactor recently been shutdown from an extended run which could have resulted in a more significant event. The cause classification for this event per the criteria of NUREG-1022 is "A", Personnel Error.

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

TITLE

UNEXPECTED ISOLATION WHEN THE WRONG FUSE WAS PULLED DURING A CLEARANCE

INITIAL CONDITIONS

Unit 2 was in COLD SHUTDOWN in day 320 of a dual outage. The A loop of the Residual Heat Removal (RHR) system was in service, operating in the Shutdown Cooling (SDC) Mode. Reactor coolant temperature was approximately 97° F and level was being maintained between 200 - 220 inches.

EVENT NARRATIVE

03/04/93

The Clearance Center completed and approved Clearance #2-93-698 for the Unit 2 Reactor Recirculation system (B32), sample line inboard isolation (F019) solenoid valve, 2-B32-SV-F019. The purpose of the clearance was to isolate the power to 2-B32-SV-F019 to allow its replacement for an environmental qualification preventive maintenance route, 93YABD03. The clearance required removal of the F019 control circuit fuse, 2-A71-F13. The clearance tag sheet and special instructions included information stating the fuse was located in the Control Room Back Panel 2-H12-P622, Terminal Strip AA (TBL AA), Fuse F14 (5 AMP), (label A71- F13). This information indicates that the desired fuse is located 14 positions from the top of fuse block TBL AA.

03/[illegible]

The Unit SRO directed the RO to hang Clearance #2-93-698. The RO and SRO discussed the requirements for pulling fuse 2-A71-F13 located on TBL AA at fuse F14 as indicated by the special instructions. They agreed that if the desired fuse was not clearly identified, the RO would obtain the applicable drawings before removing the fuse. The RO informed the Plant Monitor RO (i.e., a second RO on the same unit) that he was preparing to pull fuse 2-A71-F13 for the 2-B32-F019. The RO went to Panel 2-H12-P622 and located Terminal Strip AA. He found what he believed to be fuse 2-A71-F13 located at position F14, based on these indicators:

- There was an engraved piece of white plastic behind the fuse clearly marked with the identification F14 adjacent to the fuse.
- A wire with an attached wire label, reading A71-F13-2, was landed at terminal 2 of the fuse. The RO interpreted this as consistent with the label information included in the Special Instructions (i.e., label A71-F13). However, the interpretation was in error; the wire labeled A71-F13-2 is associated with fuse 2-A71B-F15.

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

At 2224, the RO removed the indicated fuse. The fuse was actually 2-A71B-F15, which supplies power to relay logic that initiates an isolation signal for the RHR SDC Inboard Suction Throttle valve, 2-E11-F009. The RO returned to the Control Room to verify that position indication for 2-B32-F019 had extinguished. He noted that the indication lights were still illuminated. Because some valve indications are powered from auxiliary sources, he and the SRO decided to attempt to stroke the valve to verify power was removed. When the valve moved the SRO went to the Clearance Center to obtain the applicable prints. At about the same time, the PM RO noted a one inch increase in reactor water level and asked if anything had been done that could have caused the change. An extra RO noted and announced that 2-E11-F009, was closed. The shift operators immediately confirmed that SDC had tripped, by noting the 2-E11-F009 was closed, the 2C RHR pump was not running and flow was not indicated on the RHR Loop A and B Flow recorder, 2-E11-FR-R608. The elapsed time from fuse removal to discovery of the SDC interruption was less than two minutes. The SRO directed the RO to install the fuse, and actions were started to restore SDC. SDC was restored at 2400. During the time that SDC was not in service the reactor coolant temperature rose from 97.1° to 98.4° F. The Reactor Water Clean-up system was utilized to monitor the coolant temperature for the duration of the event.

CAUSE OF EVENT

The label on TBL AA in H12-P622 was incorrect. The attached label either slipped down one position or was improperly positioned at installation. As a result F15 is labeled as F14.

The RO did not use all available indications to verify the proper fuse before pulling it: he did not count down from the top of the fuse block to verify its position, nor did he have a copy of the panel drawing available and in-hand to [illegible] the identification of the fuse by wire labels.

CORRECTIVE ACTIONS

An appropriate method of improving the labeling of back-panel fuse blocks to minimize confusion will be implemented prior to each Unit's start-up.

Standing Instruction # 93-055 was issued on March 08, 1993, directing that fuse removal be done only by licensed operators, and with applicable drawings in-hand. This instruction will remain in place until the labeling of back-panel fuse blocks is improved.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

SAFETY ASSESSMENT

This event had minimal safety significance. The reactor had been shutdown for ten months prior to this event and the loss of SDC for one hour and 36 minutes resulted in an approximate one degree F increase in reactor coolant temperature. This event could have been more significant had the reactor recently been shutdown from an extended run.

PREVIOUS SIMILAR EVENTS

No previous events involving removal of an incorrect fuse which resulted in a 10CR 50.73 report were identified.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
RHR/SDC	BO
2-B32-SV-F019	AD/ISV/SV
2-A71-F13	AD/FU
2-A71B-F15	BO/FU
2-E11-F009	BO/ISV
2C RHR PUMP	BO/P
2-E11-FR-R608	BO/FR