CP&L

Carolina Power & Light Company

Brunswick Nuclear Project
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November 15, 1990

FILE: B09-13510C
SERIAL: BSEP/90-0770

10CFR50.73

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
<u>LICENSEE EVENT REPORT 1-90-022</u>

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. L. Harness

J. L. Harness, General Manager
Brunswick Nuclear Project

TMJ/

Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
BSEP NRC Resident Office

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant Unit 1	05000325	01 OF 03

TITLE (4) Primary Containment Isolation System Actuation of an Excess Flow Check Valve while Performing a Maintenance Surveillance Test.

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQ. NO.		REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	22	90	90	-	22	-	00	11	15	90		

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

OPERATING MODE (9)	5								
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	000		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
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)

On October 22, 1990, at 0339, during the performance of Excess Flow Check Valve Reactor Instrument Penetration System Isolation Valve Functional Test - Penetrations X53 and X82, the excess flow check valve (EFCV) for two Reactor Core Isolation Cooling (RCIC) system pressure instruments on penetration X61 closed. The event was caused by personnel error, self checking/verification not completely applied, resulting in the involved personnel testing the wrong EFCV. Factors contributing to the error include the test frequency, noise in the area, fatigue, a test interruption and the fact that the location of the EFCV test connection was different than the expected location based on past experience with similar tests. Personnel have been counselled on attention to detail and proper communications. Training will be provided to appropriate personnel on this event. The specific location of the EFCV test valve will be added to the procedure and the procedure will be evaluated to possibly eliminate the need for a test interruption. This event had minimal safety significance as the reactor was defueled, primary containment was not required and the valve functioned as designed. Past similar events include LERs 1-90-19, 1-89-21, 2-90-10, 2-89-03, 2-89-11.

NRC FORM 366A 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	90	-	022	-	00	02 OF 03

TEXT (IF MORE SPACE IS REQUIRED, USE ADDITIONAL NRC FORM 366A'S) (17)

EVENT

During the performance of Maintenance Surveillance Test 1MST-EFCV13R, valve 1-E51-V90 was connected to the test equipment instead of 1-E21-V90 resulting in a unexpected Primary Containment Isolation System (PCIS) isolation of the excess flow check valve 1-E51-F043A.

INITIAL CONDITIONS

On 10-22-90, the Unit 1 reactor was in the 26th day of a scheduled refuel and maintenance outage. The reactor was defueled. 1MST-EFCV13R, Excess Flow Check Valve Reactor Instrument Penetration System Isolation Valve Functional Test - Penetrations X53 and X82, was in progress. Six of seven excess flow check valves (EFCV) had been completed. The test equipment had been moved from Unit 1 50 foot, to the 20 foot for the last EFCV (1-E21-F017A, EFCV for Core Spray delta Pressure instrument 1-E21-PDS-N004A) to be tested.

EVENT DESCRIPTION

Maintenance Instrumentation and Control (I&C) and Electrical (E) personnel were performing the test. The test had been in progress since approximately 2000 on 10-21-90. While the electrician had been moving the test equipment, the lead technician located valve 1-E51-V90 at the instrument rack where the procedure had them hook up for communications and a demineralized water supply. The lead technician read the tag from approximately three feet away and informed the electrician that he had located the valve. The electrician left the area to go dress out. When he returned, he connected to the demineralized water supply and set up communications at panel H12-P009, in accordance with the procedure. He then asked the lead technician which test connection valve to connect to. The lead technician pointed to the 1-E51-V90 valve and the electrician responded by reading out "V90" and the lead technician confirmed it. The electrician then proceeded, in accordance with the procedure, and connected the vacuum pump test assembly to the valve. The surveillance was then performed on the EFCV. At 0339, 1-E51-F043A, EFCV for Reactor Core Isolation Cooling (RCIC/E51)flow and pressure instruments 1-E51-PS-N019A and 1-E51-PDT-N017, closed as evidenced by the annunciation PEN X61 ELEVATION 36 foot AZIMUTH 90 degrees, a green closed indication and a X61 LINE BREAK status light in the control room. Coincident with its closing, the annunciation RCIC STEAM LINE BREAK DELTA PRESSURE HIGH alarmed. The test was suspended, the valve tag was rechecked and found to be 1-E51-V90, test valve for EFCV 1-E51-F043A, instead of 1-E21-V90, test valve for EFCV 1-E21-F017A. The EFCV 1-E51-F043A was reset to the normal open position, the 1-E51-V90 was restored to the closed position and the vacuum test assembly was disconnected. After a review of the event with Operations personnel, the MST was allowed to continue. The correct valve, 1-E21-V90, was located on instrument rack H12-P016, about four feet from the 1-E51-V90. The Operations Shift Foreman verified that the correct valve was to be used and the test assembly was connected to it. The MST was completed satisfactorily.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Brunswick Steam Electric Plant Unit 1	05000325	90	-	022	-	00	03 OF 03

TEXT (IF MORE SPACE IS REQUIRED, USE ADDITIONAL NRC FORM 366A'S) (17)

ROOT CA_SE/EVENT INVESTIGATION

The event was caused by personnel error, self checking/verification not completely applied. Contributing to the error were a number of factors. This test is performed once each 18 months. Other MSTs, which perform the same functional type tests, have the demineralized water and communications hook up at the same instrument rack where the test valve is located. This resulted in the expectation that the test valve would be at panel H12-P009. In addition, noise in the area led to the pointing out of what was thought to be the valve, by the lead technician, and calling out of "V90" rather than repeating "1-E21-V90" during the verification of the valve by the electrician. The personnel performing the MST were on the first day of a twelve hour night shift rotation and were, consequently, fatigued. The interruption of the test caused by the need to move the test equi_ _nt for the final EFCV may also have contributed to the error.

Past similar events which resulted in Engineered Safety Actuations _e to personnel error (ie, wrong component, channel or system) include LERs 1-90-01_ 2-90-010, 1-89-021, 2-89-003 and 2-89-011.

CORRECTIVE ACTIONS

Involved personnel have been counselled about attention to detail and the need to utilize proper communications techniques when working with others.

Training will be provided to appropriate personnel emphasizing attention to detail and communications.

A procedure revision has been initiated to add the specific location of 1-E21-V90 valve to prevent a recurrence of this event.

Revising the MST to remove EFCV 1-E21-F017A and add it to another which is performed on the 20 foot, in the same area as the 1-E21-V90, will be evaluated to possibly alleviate the need to relocate the test equipment during the performance of the test.

EVENT ASSESSMENT

This event had minimal safety significance as the reactor was defueled, primary containment is not required and the PCIS valve, 1-E51-F043A, functioned as designed.

EIIS CODES

PCIS	JM	EFCV	JM/ISV	RCIC	BN