CP&L

Carolina Power & Light Company

Brunswick Nuclear Project
P. O. Box 10429
Southport, NC 28461-0429

December 27, 1989

FILE: B09-13510C
SERIAL: BSEP/89-1119

10CFR50.73

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-89-024

Gentlemen:

In accordance with Title 10 to 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,

J. L. Harness, General Manager
Brunswick Nuclear Project

TMJ/jlh

Enclosure

cc: Mr. S. D. Ebneter
Mr. E. G. Tourigny
BSEP NRC Resident Office

IE22
1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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.

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 5

TITLE (4) Failure to Test Seventeen Primary Containment Isolation Valves Per Tech Spec 4.6.1.1.a Due to Failure to Recognize Testing Applicability

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)
1 1	2 7	8 9	8 9	— 0 2 4 —	0 0	1 2	2 7	8 9	BSEP Unit 2	0 5 0 0 0 3 2 4
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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)	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)	X 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)
Theresa M. Jones, Regulatory Compliance Specialist	9 1 9 4 5 7 - 2 0 3 9

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
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	3 0	9 0

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On November 27, 1989, it was determined as reportable that seventeen primary containment isolation valves were not being tested in accordance with Technical Specification (T/S) Surveillance Requirement 4.6.1.1.a. Sixteen of the seventeen valves were installed by Plant Modifications (PMs) and the seventeenth was removed as a temporary repair and then reinstalled. The root cause of the failure to identify appropriate testing requirements for the valves installed by PMs was determined to be a failure to identify applicable design criteria and regulatory commitments in the design basis documentation. The root cause for the failure to incorporate the seventeenth valve back into the appropriate test has not yet been determined. An investigation is still underway into the root cause for the failure to develop appropriate design basis documentation and failure to reincorporate the resinstalled valve back into the test. A revision to incorporate these valves into the appropriate test has been completed. A review of each primary containment penetration is underway to ensure appropriate testing is being carried out on the remainder of the valves. A supplement to this report will be issued by June 30, 1990. This event had minimal safety significance.

Event

On November 27, 1989, it was determined as reportable that seventeen primary containment isolation valves were not being tested in accordance with Technical Specification (T/S) Surveillance Requirement 4.6.1.1.a.

Initial Conditions/Event Description

Unit 1 reactor was at 100% power and Unit 2 was shut down in Refuel Outage No. 8. During a review of common (i.e., for both Unit 1 and Unit 2) Periodic Test 02.2.4a, Primary Containment Integrity Verification-Containment External, Operations personnel determined that seventeen valves were not being verified by the PT that appeared to meet the requirements for verification in accordance with Technical Specification Surveillance Requirement 4.6.1.1.a. A self-identified Significant Condition Adverse to Quality (SCATQ), 89-045, was initiated on October 12, 1989, in accordance with Plant Procedure (PLP) 04, Corrective Action Program.

Event Investigation

As a result of SCATQ 89-045, Nonconformance Report (NCR) S-89-102 was initiated on October 25, 1989. The NCR is based on the fact that contrary to the design criteria for code "D1" valves, listed in System Description (SD) 12, Primary Containment Isolation System (PCIS) (EIIS/JM), the subject valves (see Attachment A) were not being checked per PT-02.2.4a. An investigation determined that sixteen of the seventeen valves had been installed via Plant Modifications (PMs) (see Attachment A) without identification of the design criteria or applicable T/S. The PMs were installed by the Nuclear Engineering Department (NED) and the former Brunswick Engineering Support Unit (BESU). (BESU responsibilities have now been divided between NED and plant Outage Management and Modification (O&M) group.) In addition, the PM review by Operations and Technical Support Group personnel failed to identify the surveillance requirements applicable to the involved valves.

The remaining valve, 2-E11-V127 (the body drain isolation valve to the Residual Heat Removal B loop Suppression Pool Cooling Isolation valve), had been removed as a temporary repair in accordance Engineering Evaluation Request 86-0224 and the requirement to test the valve had consequently been removed from PT-02.2.4a. However, the valve was replaced by Direct Replacement 87-195 during the following outage without revising PT-02.2.4a to reincoporate the necessity to test the valve.

The root cause of the failure to identify the appropriate surveillance requirements and include the valves in PT-02.2.4a was determined to be the failure of the PMs design basis documentation to identify the applicable design criteria and regulatory related commitments per Engineering Procedure (ENP) 03, Plant Modification Procedure. Contributing to the failure of Operations and Technical Support personnel to identify the applicable surveillance requirements was a 1985 Technical Specification Interpretation (TSI), 85-01, that defined PCIS valves as those listed in ENP-16, Procedure for Administrative Control of Inservice Inspection Activities, which lists valves requiring Local Leak Rate Testing (LLRT). This TSI indicated that if a valve did not require LLRT it was not a primary containment concern. It is felt this contributed to the failure of personnel to identify the testing requirements while performing associated procedure revision requirement reviews. On November 6, 1987, the TSI was reissued to correctly refer to SD-12.

Root Cause

The root cause of the failure of the PMs design basis documentation to identify the design criteria and applicable regulatory commitments has not yet been determined. An investigation is continuing and a supplement to this report will be issued by June 30, 1990.

Corrective Actions

Revision 16, of PT-02.2.4a, implemented on October 17, 1989, incorporated the subject valves.

The Technical Support group is currently reviewing each primary containment penetration against SD-12, PT-02.2.4 and 02.2.4a, and the Final Safety Analysis Report (FSAR) to determine and initiate any additional revisions to these documents. This review is scheduled to be complete by June 1, 1990.

NED is currently investigating this event in accordance with its Design Deficiency Program, NED procedure 3.18. This item is scheduled to be resolved by February 1, 1990.

An investigation into the cause of the failure to reincorporate 2-E11-V127 into PT-2.2.4a is continuing. The results will be included in the supplement.

Event Assessment

This event had minimal safety significance. The containment isolation system is designed to prevent or limit the release of radioactive material that may result from postulated accidents. During this event personnel would have been able to identify a PCIS problem and take appropriate actions to mitigate a release to the Reactor Building. Secondary containment is designed to prevent a release to the environment. In addition, the Unit 1 and 2 CAC (EIIS/BB)

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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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	0 5 0 0 0 3 2 5	8 9	— 0 2 4 —	0 0	0 4 OF 0 5

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

valves, CAC-V164, V166, and V169 each have inboard redundant isolation valves which are and were verified by PT-02.2.4a. The reactor vessel head inner seal lies between the reactor and the Unit 1 or 2 B21-V83. If both the inner seal and the B21-V83 were to leak, the leakage would be identifiable by condensation and sound associated with steam escaping to the Reactor Building fifty-foot atmosphere. The 2-E11-V127 is located on the outboard side of the 2-E11-F024B disc. The E11-F024B acts as a redundant isolation and is tested in accordance with LLRT procedures. In addition, the line associated with the E11-F024B discharges below suppression pool water level which results in a water seal normally being present. The remaining valves, Unit 1 and 2 B21-V160, V161, V162, and V163, exist on fluid filled lines supplying the N026A and B reactor level transmitters. Leakage associated with these valves would be apparent by level indications problems and associated water leakage would have been identified by personnel in the Reactor Building.

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 9	— 0 2 4 —	0 0	0 5 OF 0 5

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

ATTACHMENT A

PM No.	
80-133	1-CAC-V164, V166, V169
80-134	2-CAC-V164, V166, V169
82-287C	1-B21-V83
82-288C	2-B21-V83
86-007	1-B21-V160, V161, V162, V163
86-008	2-B21-V160, V161, V162, V163

Direct Replacement

87-195	2-E11-V127

Valve No.	Valve Name
1(2)-CAC-V164	CAD N_2 Injection Line Vent Valve (EIIS/BB/VTV)
1(2)-CAC-V169	CAD N_2 Injection Line Vent Valve (EIIS/BB/VTV)
1(2)-CAC-V166	Suppression Pool Purge Exhaust Line Vent Valve (EIIS/BB/VTV)
1(2)-B21-V83	Test Valve for Excess Flow Check Valve (EFCV) B21-F008 (EIIS/IJ/TV)
1(2)-B21-V160	Test Valve for EFCV B21-IV-2456 (EIIS/IG/TV)
1(2)-B21-V162	Test Valve for EFCV B21-IV-2456 (EIIS/IG/TV)
1(2)-B21-V161	Test Valve for EFCV B21-IV-2455 (EIIS/IG/TV)
1(2)-B21-V163	Test Valve for EFCV B21-IV-2455 (EIIS/IG/TV)
2-E11-V127	Valve E11-F024B Body Drain Isolation Valve (EIIS/BO/*/ISV)

*Component identifier not found.