CP&L

Carolina Power & Light Company

Brunswick Nuclear Plant
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MAY 23 1994

SERIAL: BSEP-94-0188
10CFR50.73

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

BRUNSWICK NUCLEAR PLANT UNIT 2
DOCKET NO. 50-324/LICENSE NO. DRP-62
LICENSEE EVENT REPORT 2-94-006

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 and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

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

Very truly yours,

J. Cowan, Director-Site Operations
Brunswick Nuclear Plant

SFT/

Enclosures
1. Licensee Event Report
2. Summary of Commitments

cc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. P. D. Milano, NRR Project Manager - Brunswick Units 1 and 2
Mr. R. L. Prevatte, Brunswick NRC Senior Resident Inspector

IE22 1/1

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.NONE	

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 2	05000324	1 of 5

TITLE (4)
PENETRATION LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION ALLOWABLE LIMITS

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
04	21	94	94	- 06 -	00	05	23	94		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 4
POWER LEVEL (10): 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)	X	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
Steve F. Tabor, Regulatory Affairs Specialist	(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
X	JM	ISV	A391	Y					

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 April 21, 1994, during the Unit 2 B211R1 refuel outage, local leak rate testing (LLRT) of the Reactor Water Cleanup (RWCU) system suction line inboard and outboard isolation valves had been performed. Although the initial LLRT of the valves obtained acceptable results, subsequent testing indicated that the valves would not pressurize. Consequently, the overall test was considered a failure in that the limits established in the Technical Specification for allowable primary containment leakage were exceeded.

The RWCU suction line isolation valves are double-disc gate valves manufactured by Anchor Darling. The failure of the valves to minimize leakage within the allowable limits is attributed to non-uniform contact of the valve upper and lower wedges. This failure mechanism was previously documented in LER 1-92-023. The valve wedges were replaced and two post maintenance leakage tests performed. Both post maintenance leakage tests on each valve verified that leakage rates were well within the allowable limits.

The failure of both of the RWCU suction line isolation valves represents a potential loss of the primary containment isolation capability for that penetration had a design basis accident occurred. Previous similar events were reported in LERs 1-91-016 and 1-92-023.

The cause classification of this event per the criteria of NUREG-1022 is B, failures reasonably attributed to the manufacture of a component.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant Unit 2	05000324	94	- 06 -	00	2 of 5

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

TITLE

PENETRATION LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION ALLOWABLE LIMITS

INITIAL CONDITIONS

On April 21, 1994, the Unit 2 reactor was shutdown in day 27 of refueling outage B211R1. Local leak rate testing (LLRT) of the Reactor Water Clean Up (RWCU) system suction line inboard and outboard isolation valves had been performed to satisfy Technical Specification primary containment leakage rate surveillance requirements. Technical Specification 3.6.1.2.b requires that the overall primary containment leakage rate be limited to less than or equal to 0.60 L_a for primary containment isolation valves and associated penetrations when pressurized to P_a, 49 psig.

EVENT NARRATIVE

On April 20, 1994, leakage testing of the RWCU system suction line inboard and outboard isolation valves commenced. To ensure repeatability as committed in LER 1-92-023, the performance of two leakage tests on each of the valves was planned. The initial test data indicated that the associated penetration leakage rate was well within acceptable limits. The outboard valve passed an initial leak rate test with a measured leakage of 0.51 scfh demonstrating that the valve sealing surfaces were acceptable. The initial leak rate of the inboard valve was approximately 8 scfh indicating that no significant degradation of the valve's sealing surfaces existed.

A second leakage rate test was performed on each valve prior to the performance of maintenance on the valves. In each of the second tests the valves failed to pressurize. A third leakage test was performed on each of the valves. During the third tests the valves again failed to pressurize. The results of the second and third tests indicate that the valves were not capable of repeatedly controlling primary containment leakage. This event is being reported in accordance with the requirements of 10CFR50.73 (a) (2) (v) in that the leakage of the RWCU suction line isolation valves in excess of the overall limits established by Technical Specification represents a condition that alone could have prevented the fulfillment of the primary containment isolation system to control the release of radioactive material. As discussed below, however, the safety significance of this event is considered minimal.

CAUSE OF EVENT

The failure of the RWCU suction line isolation valves to repeatedly control leakage within the allowable limits is attributed to non-uniform contact of the valves' upper and lower wedges. This failure mechanism was previously documented in LER 1-92-023.

CORRECTIVE ACTIONS

The RWCU suction line isolation valves which failed to meet the LLRT requirements contained wedges with hand-ground mating surfaces. Matched wedges with machined mating surfaces were installed in both of the RWCU suction line isolation valves. Following replacement of the wedges two post maintenance leakage tests were performed. Both post maintenance leakage tests on each valve verified that leakage rates were well within the allowable limits.

SAFETY ASSESSMENT

The LLRT failure of the RWCU suction line isolation valves resulted in a leakage in excess of the overall limits established in the Technical Specification. An analysis was performed to determine the safety significance of this condition during the High Energy Line Break (HELB) and the Loss Of Coolant Accident (LOCA) scenarios. The following provides the results of this analysis:

HELB:

The RWCU suction line isolation valves are designed to close in response to a sensed break in the downstream piping. Although the valves exhibited significant leakage during LLRTs, it is expected that they would have closed and been capable of sealing in response to a line break. Subsequent valve inspection indicates that the valve seats were in acceptable condition and the high differential pressure expected during a HELB would have provided adequate seating force on the discs. The reasons for this conclusion are as follows:

Seat Condition:

The outboard valve passed an initial leak rate test with a measured leakage of 0.51 scfh demonstrating that the valve sealing surfaces were acceptable. The initial leak rate of the inboard valve was approximately 8 scfh indicating that no significant degradation of the valve's sealing surfaces existed. When the valves were disassembled for repair the "as found" seat conditions were determined to be acceptable.

Sealing Performance:

The RWCU suction line isolation valves are Anchor/Darling double disc gate valves. As documented in LER 1-92-023 there are concerns about the seating/sealing repeatability of these type valves. The concerns are attributed to variations in wedge position and seating force when the valves are closed under zero pressure conditions. Sealing problems are not expected when high differential pressure is applied across the valves because significant additional force is applied to the valve disc.

Based upon this, the leakage rate identified during LLRT of the RWCU suction line isolation valves is not considered safety-significant during a High Energy Line Break of the RWCU piping.

LOCA:

During a LOCA, the RWCU isolation valves may be required to close against low differential pressure. In this case, if both valves had failed to seat, they would have been subject to a potential leak rate in excess of 1,000 scfh. A review of the potential leakage paths determined that RWCU flow is normally returned to the Reactor Vessel via the Feedwater System and may also be directed to the Main Condenser or the Radwaste System. The other potential leak path is through failures or component leaks in the RWCU system. Each potential path was evaluated to determine its safety significance.

Piping & Component Leaks

The RWCU equipment outboard of 2-G31-F004 is not safety-related. However, it normally operates at pressures in excess of 1,000 psig and is unlikely to fail when system pressure is reduced to 49 psig or less, assuming that the postulated LOCA does not occur concurrently with a seismic event. BNP is not required to be designed to withstand this combination of events, so the assumption is valid.

It should also be noted that, if both valves had failed to seat and post-accident leakage had been released through RWCU components into the secondary containment, flow would have been less than 1% of the Standby Gas Treatment System capacity. Therefore, component leakage or post-LOCA failures and subsequent leakage to the secondary containment are not considered to be significant safety concerns.

Normal Return Path

The primary RWCU return path is through the "B" Feedwater line to the Reactor Vessel. According to ISI test results, the outboard Feedwater isolation valve passed LLRT. Consequently, leakage to the Feedwater System via the RWCU System could not have exceeded allowable limits and is not a safety concern. However, because the RWCU reject to main condenser isolation valves are not subject to the requirements of Appendix J leak testing, a potential leak path to the environment existed. For primary containment isolation valve leakage to reach the condenser after an accident, the leakage would first have to displace the water in the RWCU system. Since the water leakage rate to the condenser is believed to be low at a system pressure of approximately 1100 psig, leakage at an accident pressure of 49 psig is expected to be much lower. Although not quantified, the time delay before containment leakage would have reached the condenser is believed to be significant. As such, the safety significance of this event is considered minimal.

Main Condenser & Radwaste Return Paths

During infrequent operations, RWCU can be lined up to reject water to the Main Condenser or the Radwaste System. Either of these flow paths, had they been open at the initiation of a LOCA, had the potential to release RWCU leakage outside the primary and secondary containments. The Main Condenser has been previously analyzed to be a known potential leakage path to the environment. The offsite dose effects of 1,128 scfh leakage to the condenser were evaluated by General Electric and documented in letter OG92-079-09, dated January 28, 1992. This document specified that leakage above 1,128 scfh could result in an offsite dose in excess of 10CFR100 limits. Since the potential leakage through the RWCU suction line isolation valves is estimated to have been in excess of 1,000 scfh, the potential existed to exceed 10CFR100 limits had both of the valves failed to properly seat while RWCU was operating in either the Condenser Blowdown or Radwaste Reject modes concurrent with a design basis LOCA. Although not quantified, the probability of occurrence of this sequence of events is believed to be extremely low. Therefore, a potential leak path in excess of the allowable offsite dose limits existed but due to the low frequency of operation in modes that could have permitted high post-LOCA leak rates, the safety significance of this event is considered minimal.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant Unit 2	05000324	94	- 06 -	00	5 of 5

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

PREVIOUS SIMILAR EVENTS

Previous similar events were reported in LERs 1-91-016 and 1-92-023.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
JM	Primary Containment Isolation System
ISV	Isolation Valve