CP&L

Carolina Power & Light Company

Brunswick Nuclear Plant
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JUN 27 1994

SERIAL: BSEP-94-0228
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
SUPPLEMENTAL LICENSEE EVENT REPORT 2-94-004

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Supplemental Licensee Event Report. The original report fulfilled the requirement for a written report within thirty (30) days of a reportable occurrence and was 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. Supplemental 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

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, Unit 2	05000324	1 of 4

TITLE (4)
PENETRATION LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION ALLOWABLE LIMIT DURING LOCAL LEAK RATE TESTING

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR
03	29	94	94	- 04 -	01	06	23	94

OTHER FACILITIES INVOLVED (8)	
FACILITY NAME	DOCKET NUMBER
	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	SB	ISV	R344	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 March 29, 1994, the Unit 2 reactor was shutdown in day 4 of refueling outage B211R1. Local leak rate testing (LLRT) of the main steam isolation valves (MSIVs) had been performed. The results of the LLRT indicated that the leakage on main steam line (MSL) A inboard and outboard isolation valves was approximately 135 and 38.6 standard cubic feet per hour (scfh) respectively, exceeding the technical specification limit of 11.5 scfh. The cause of the LLRT failures is attributed to indications identified on the valve seats and discs which resulted from foreign material intrusion. Additionally, a contributing factor to the leakage of the inboard isolation valve is attributed to actuator-to-valve misalignment. The valves' seat and disc surfaces have been repaired to ensure proper sealing surfaces and modified junk rings installed to prevent misalignment problems. The inboard valve actuator guide tube assembly was disassembled and remachined to improve actuator alignment. Following valve maintenance the valves successfully passed LLRTs. The safety significance of this event is considered minimal. Previous events involving MSIVs not meeting technical specification leakage requirements have been reported in LERs 1-88-025, 2-91-019, and 2-92-10.

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

TITLE

PENETRATION LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATION ALLOWABLE LIMIT DURING LOCAL LEAK RATE TESTING

INITIAL CONDITIONS

On March 29, 1994, the Unit 2 reactor was shutdown in day 4 of refueling outage B211R1. Local leak rate testing (LLRT) of the main steam isolation valves (MSIVs) had been performed to satisfy technical specification primary containment leakage rate surveillance requirements. Technical specifications require that the primary containment leakage rate be limited to less than or equal to 11.5 standard cubic feet per hour (scfh) for any one main steam line (MSL) isolation valve when tested at 25 psig.

The MSIVs are Rockwell Model 1612, 24", Y-type globe valves.

EVENT NARRATIVE

The LLRT is accomplished by draining the MSL, closing the inboard and outboard MSIVs, pressurizing the space between the valves to 25 psig with air while providing a vent path for any leakage, and measuring the leakage rate. In the event of a leakage rate equal to or greater than 11.5 scfh, the test procedure directs that the associated MSL be filled with water between the reactor vessel and the inboard MSIV to seal any leakage past the inboard MSIV. The space between the MSIVs is again air pressurized and the leakage rate resulting from leakage past the outboard MSIV is measured. This method of testing indicated the total MSL leakage, the leakage associated with the outboard MSIV, and by subtraction of the outboard MSIV leakage from the total MSL leakage, the leakage associated with the inboard MSIV.

On March 27, 1994, leakage testing of MSL A MSIVs commenced. Testing was initially performed by closing the inboard and outboard MSIVs and pressurizing the space between the valves. The total line leakage was determined to be approximately 173 scfh. Subsequent testing was accomplished by raising reactor vessel level and filling the MSL until a static head pressure of 25 psig was reached at the inboard MSIV. Leakage past the outboard MSIV was determined to be approximately 38.6 scfh. By subtracting the outboard MSIV leakage from the total MSL leakage, the inboard MSIV leakage was determined to be approximately 135 scfh.

This event is being reported in accordance with the requirements of 10CFR50.73 (a) (2) (v) in that the leakage of MSL A MSIVs in excess of the 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

An internal inspection of the MSL A inboard and outboard isolation valves was performed to support identification of the cause of the LLRT failures. Based on this inspection the primary cause of the valve leakage failures is attributed to scratches and pitting identified in the sealing area of the in-body seats and discs. These indications appear to have resulted from closure of the valve on material on the seats although this material is indeterminate and was not identified during the Unit 2 outage inspection and repair activity. Galling of the valve body bore resulting from excessive friction between the valve disc and the wall of the valve is believed to be the source of the foreign material. The valve vendor has recognized that excessive valve internal friction may occur as a result of normal valve wear. To correct this concern the vendor has produced an improved valve disc design with less valve internal friction.

Additionally, a potential contributing factor to the leakage of the inboard isolation valve was actuator-to-valve misalignment. Actuator-to-valve misalignment is attributed to the deflection of the actuator guide tubes that results from the cantilevered loading imposed by the actuator on the bonnet assembly over time. The MSL A inboard and outboard isolation valves have performed reliably in the past and have not required mechanical repair as a result of misalignment.

CORRECTIVE ACTIONS

The following repairs to the MSL A isolation valves have been completed:

The actuator guide tube assemblies were removed from the inboard valve actuator and disassembled. These assemblies were verified to be straight and the guide tube feet were machined flat to align the actuator.

The in-body seats on both MSL A isolation valves were lapped and narrowed to remove the pitting and scratches and ensure a proper sealing surface.

Taller junk rings were installed in the packing chambers of both MSL A isolation valves. The taller junk rings are designed to improve disk centering and reduce friction between the packing chamber components and the valve stem.

The disk piston assemblies and the stem disks on both MSL A isolation valves were replaced with the latest design. The replacement disk pistons are designed to reduce the potential for disk misalignment. Additionally, the redesigned disc piston weighs less and has a different center of gravity resulting in less friction between the valve internal surfaces. The bore of the valves were flapped to remove the galling. With the new lighter disc, the new junk ring, and the re-alignment of the valve components, future valve seat damage by particles resulting from galling of the body bore is unlikely.

Prior to restoring the valves to service the valves successfully passed LLRTs.

SAFETY ASSESSMENT

Primary containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitations, will limit the site boundary radiation doses to within the limits of 10CFR100 during accident conditions. Technical specifications require that primary containment leakage rates be limited to less than or equal to 11.5 scfh for any one MSL isolation valve tested at 25 psig.

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

The safety significance of this event is considered minimal. MSL A leakage was limited to approximately 38.6 scfh. Although the technical specification limits were exceeded, a MSL leakage rate of 38.6 scfh would not have resulted in the 10CFR100 off-site dose limits being exceeded during a design basis accident (LOCA). This assessment is based upon dose calculations completed for Brunswick Units 1 and 2 by General Electric (GE letter OG-91-1063-09, dated December 17, 1991), using the Boiling Water Reactor Owners Group methodology as currently approved by the NRC. This calculation determined that with 100 scfh leakage per MSL, the off-site dose would be less than 10CFR100 limits.

PREVIOUS SIMILAR EVENTS

Past failures of MSIVs to meet the technical specification required 11.5 scfh leakage requirements were reported in LERs 1-88-025, 2-91-019, and 2-92-010. LER 1-88-025 involved the outboard MSIV on two separate MSLs. LER 2-91-019 reported the failure of both MSLs C and D. LER 2-92-010 involved the failure of MSL D inboard and outboard MSIVs.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
MSIV	SB/ISV

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
NONE	N/A