CP&L

Carolina Power & Light Company
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DEC 13 1996

SERIAL: BSEP 96-0471
10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO.1
DOCKET NO. 50-325/LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-96-007, SUPPLEMENT 1

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits Supplement 1 to Licensee Event Report 1-96-007, High Pressure Coolant Injection System Inoperable Due to Steam Supply Valve Packing Leak. This supplement provides additional information regarding the cause of the High Pressure Coolant Injection System steam supply valve stem galling. Please refer any questions regarding this submittal to Mr. Mark Turkal at (910) 457-3066.

Sincerely,

W. Levis

W. Levis
Director — Site Operations
Brunswick Nuclear Plant

MAT/wrm

Enclosures
1. Licensee Event Report 1-96-007, Supplement 1
2. Summary of Commitments

cc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, Jr., NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, NRC Senior Resident Inspector - Brunswick Units 1 and 2
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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

TITLE (4)

High Pressure Coolant Injection System Inoperable Due to Steam Supply Valve Packing Leak

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 05000
05	09	96	96	-- 07 --	01	12	13	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor, Sr. Analyst, Regulatory Affairs	(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	BO	VLV	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 15 single-spaced typewritten lines) (16)

On May 9, 1996, with Unit 1 operating at rated power, the High Pressure Coolant Injection (HPCI) system valves were being stroke time tested. After discovering steam in the HPCI room, Operations manually isolated HPCI. HPCI was declared inoperable at 2033 hours.

A steam leak from the packing of the HPCI Steam Supply Valve, 1-E41-F001 was the cause. On May 11, 1996, valve repairs and testing were completed. On May 12, 1996, at 0342 hours, the HPCI Operability Test was performed with no leakage from the 1-E41-F001.

During the plant refueling outage following the event, it was found that the valve body provisions for bonnet and yoke fit up did not ensure that the these items were concentric. This caused the stem to be off center in the bonnet bore and eventually resulted in stem galling. The valve was fully refurbished and the off center condition corrected. Additional corrective actions will include enhancement of the maintenance procedure to inspect for off center stems when assembling Anchor Darling pressure seal valves.

This event has minimal safety significance in that at the time of the event the Automatic Depressurization, Core Spray, Low Pressure Coolant Injection and the Reactor Core Isolation Cooling systems were operable.

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

TITLE

High Pressure Coolant Injection System Inoperable Due to Steam Supply Valve Packing Leak

INITIAL CONDITIONS

Unit 1 was operating at rated power. Stroke time testing of the High Pressure Coolant Injection (HPCI) system (BQ) isolation valves was in progress. The Automatic Depressurization, Core Spray, Low Pressure Coolant Injection and the Reactor Core Isolation Cooling systems were operable.

EVENT NARRATIVE

On May 9, 1996, while stroke time testing the HPCI system isolation valves, a small steam leak developed in the HPCI room. An Auxiliary Operator discovered steam in the HPCI room and notified the Control Room. Operations manually isolated the HPCI steam line and depressurized the line to minimize the release of steam into the HPCI room. HPCI was declared inoperable at 2033 hours. In accordance with the requirements of 10 CFR 50.72(b)(2)(iii)(D) an event notification was initiated at 2351 hours.

A leak from the HPCI Steam Supply Valve, 1-E41-F001 (BQ/ISV) stem packing was determined to be the source of the steam. On May 10, 1996, at 0015 hours, Maintenance personnel discovered the packing was damaged due to stem galling.

On May 11, 1996, following completion of repairs to the 1-E41-F001, testing to satisfy stroke time testing and packing leakage requirements was completed satisfactorily. On May 12, 1996, at 0342 hours, 0PT-09.2, HPCI System Operability Test, was satisfactorily performed with no leakage observed at the 1-E41-F001 valve stem.

This event is being reported in accordance with the requirements of 10 CFR 50.73(a)(2)(v)(D) in that the manual isolation of the HPCI system due to steam leakage resulted in the manual isolation of a single train system and thereby prevented the fulfillment of the safety function of that system.

CAUSE OF EVENT

The 1-E41-F001 steam leak was caused by degradation of the valve stem packing. Valve inspection determined that the valve stem was not centered in the stuffing box with the packing removed. This condition caused wear of the packing and eventual contact of the stem with either the gland follower or the bore of the valve bonnet, resulting in galling of the valve stem. The galling of the stem caused the damage to the valve packing. The 1-E41-F001 valve is a double-disc gate valve manufactured by the Anchor/Darling Company.

The inspection performed during the plant refueling outage found that the off center stem condition was caused by the bonnet and bonnet bore not being concentric with the yoke extension, yoke, and actuator. Previous maintenance on this valve had performed machining of the valve body bore to correct a defect in the pressure seal surface. Although the misalignment problem could have been caused by an original manufacturing error, the challenges of in-place machining make the previous corrective maintenance the most likely cause of the initial misalignment problem. Had this maintenance activity included a stem to bonnet bore alignment verification, the unacceptable condition would have been corrected. Although this verification would not be needed for a normal valve refurbishment, it is apparent that a stem alignment verification is needed when more extensive repairs are made.

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

CORRECTIVE ACTIONS

Initially the galling was removed from the stem surface and the valve packing was replaced. Following this repair, the valve was satisfactorily stroke tested, evaluated using VOTES testing equipment, and then inspected during and after turbine operation. The repaired valve performed as required with no problems noted. During the plant refueling outage following the event, the 1-E41-F001 valve was completely disassembled and refurbished. The portion of the stem that had been galled was replaced with a new component. The minor galling found on the inside surface of the bonnet bore was repaired. The disc and in body seats were resurfaced to ensure minimal leakage past the valve. Inspections of the valve internals revealed the bonnet bore was not concentric with the mounted yoke and actuator assembly. The actuator was removed and adjustments were made to allow the actuator to be mounted directly above the bonnet bore. The stem was then verified to be properly centered in the bonnet bore and new packing was installed. The repaired condition was reviewed with the vendor and confirmed that it was acceptable. Post maintenance testing was satisfactorily performed on the valve. HPCI was operated on low pressure steam for mechanical overspeed trip testing and low pressure surveillance testing requirements. HPCI was operated again during plant power ascension to meet high pressure surveillance testing requirements. This testing was completed satisfactorily with no identified problems.

The corrective maintenance procedure for the 1-E41-F001 valve and valves of similar design will be revised to inspect for correct stem to bonnet bore alignment during reassembly. As an interim measure, until the corrective maintenance procedure is revised, the procedure has been placed on restricted use.

SAFETY ASSESSMENT

This event has minimal safety significance in that at the time of the event the Automatic Depressurization, Core Spray, Low Pressure Coolant Injection and the Reactor Core Isolation Cooling systems were operable.

The limiting concern for a steam leak in the HPCI room is the automatic isolation of the HPCI system which by design would occur once HPCI room temperature reached 165 °F. During the leak, the room temperature reached equilibrium at approximately 130 °F. There was one Residual Heat Removal (RHR) system room cooler providing nominally 3980 scfm to the HPCI room during the leakage with the other RHR room cooler in standby. The normal Reactor Building Ventilation system was also in service with a design flow of 3400 scfm to the HPCI room.

Had an event occurred requiring HPCI injection concurrent with the packing leak, both of the RHR room coolers would have started once HPCI room temperature reached design actuation setpoints (120 and 145°F respectively). Engineering evaluation has determined that with two RHR room cooler fans operating at the maximum expected cooler outlet temperature, HPCI room temperature would have remained below the HPCI system room temperature isolation setpoint. Consequently, the observed leakage would not have been expected to cause the isolation of the HPCI system and thus the HPCI system would have remained capable of performing its intended safety function.

PREVIOUS SIMILAR EVENTS

Previous events involving the inoperability of the HPCI system due to equipment degradation were reported in Licensee Event Reports 1-95-022 (HPCI system discharge flow element gasket leak) and 2-95-02 (failed resistor in the HPCI governor speed control circuit power supply).

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
High Pressure Coolant Injection System	BQ
1-E41-F001	BQ/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
Maintenance procedure 0CM-VGT500 will be revised to include a step for verifying that the stem is centered in the bonnet bore.	This procedure revision will be effective by 5/1/97.