Boston Edison
A BEC ENERGY COMPANY

10 CFR 50.73

J.F. Alexander
Nuclear Assessment Group Manager

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

Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 99-003-00, "Local Leak Rate Test Results Exceeding Allowable Technical Specification Leakage Rates," is submitted in accordance with 10 CFR 50.73.

Except for submitting a supplement to this report, this letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

J.F. Alexander

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RLC/nb
ler/9900300
Enclosure: LER 99-003-00

cc: Mr. Hubert J. Miller
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

INPO Records
700 Galleria Parkway
Atlanta, GA 30339-5957

Standard BECo LER Distribution

NRC Form 366
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)
Local Leak Rate Test Results Exceeding Allowable Technical Specification Leakage Rates.

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
									N/A	05000
05	16	99	1999	003	00	06	15	99	N/A	05000

OPERATING MODE (9): N

POWER LEVEL (10): 0

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

	20.2201 (b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
	22.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(B)		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)(C)		Specify in Abstract below
	20.2203(a)(2)(iv)		50.36(c)(2)	X	50.73(a)(2)(vii)(C)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert L. Cannon - Regulatory Affairs Senior Engineer	(508) 830-8321

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SB	ISV	A585	Y	X	BN	SHV	A391	Y
X	SB	ISV	A585	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		10	31	99

Abstract (16)

On May 16, 1999, the Control Room was informed that the inboard and outboard containment isolation valves for the 'C' main steam line exceeded the Technical Specifications (T.S.) allowable leakage rates during Appendix 'J' Local Leak Rate Testing (LLRT) that was being performed during Refueling Outage (RFO) No. 12. The in-series main steam line 'C' isolation valves were found to have leakage of 280 standard liters per minute (SLM) and 255 SLM. A leakage rate of 210.5 SLM corresponds to a one percent per day total allowable leak rate from all primary containment leakage sources as specified by the T.S. A leakage rate of 21.7 SLM (46 scfh) as specified by T.S. is the maximum leakage allowed for all four main steam lines combined. Leakage less than one percent per day is necessary to limit radioactive releases to less than 10 CFR 100 limits during a design basis accident. Immediate action was taken to verify the integrity of the secondary containment. Work documents were prepared to make necessary repairs to the subject containment isolation valves.

On May 24, 1999, a check valve in the Reactor Core Isolation Cooling (RCIC) System turbine exhaust piping did not meet the acceptance criteria during the performance of Appendix J leakage testing. At the time of submittal of this LER the root causes had not yet been completed. The root causes and any additional corrective actions will be documented in a supplement to this LER following completion of the root cause analysis.

The conditions were discovered during reactor refueling operations with the reactor mode selector switch in the REFUEL position. The reactor vessel pressure was zero psig and reactor coolant temperature was 84 degrees Fahrenheit. The condition posed no threat to public health and safety.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

EVENT DESCRIPTION

On May 16, 1999, at 0805 hours, the Control Room was notified that the inboard and outboard containment isolation valves for the 'C' main steam line (penetration X-7C) exceeded Technical Specifications (T.S.) 4.7.A.2.a.3 and 4.7.A.2.a.4 allowable leakage rates during Appendix 'J' Local Leak Rate Testing (LLRT) that was being performed during the 1999 refueling outage (RFO-12). The in-series main steam line 'C' isolation valves AO-203-1C and AO-203-2C were found to have leakage of 280 standard liters per minute (SLM) and 255 SLM, respectively. The MSIVs are Atwood & Morrill Y - Pattern, 20 inch MSIV's, model number 20849-H. A leakage rate of 210.5 SLM corresponds to a one percent per day total allowable leak rate from all primary containment leakage sources as specified by T.S. 4.7.A.2.a.3(1). A leakage rate of 21.7 SLM (46 scfh) as specified by T.S.4.7.A.2.a.4 is the maximum leakage allowed for all four main steam lines combined. T.S. 3/4.7.A bases indicates that a Primary Containment leakage rate of less than one percent per day is necessary to limit radioactive releases to less than 10 CFR 100 limits during a design basis accident. This condition was documented by Problem Report 99.9216. At 0835 hours on May 16, 1999, the NRC Operations Center was notified via the Emergency Notification System in accordance with 10 CFR 50.72(b)(2)(i).

On May 24, 1999, at 0515 hours, PR 99.9266 was written to document that check valve CK-1301-64 (globe stop type check valve) failed its seat leakage test because the piping could not be pressurized between it and a blank flange (installed for local leak rate testing). Check valve CK-1301-64 is located in the steam exhaust piping between the Reactor Core Isolation Cooling (RCIC) System turbine and its termination in the Torus (Suppression Pool). The acceptance criteria for check valves located in piping that terminates below the surface of the Suppression Pool is 10.1 gpm (total combined leakage). The criteria for CK-1301-64 is 4 gpm (or less). The criteria is based on a constant primary containment pressure (Torus atmosphere pressure above water level) of 45 psig for 30 days (primary containment mission time) that could result in a decrease in the Suppression Pool water volume to a level that would result in applicable piping terminating in the pool (water seal) to become no longer submerged. The problem identified in PR 99.9266 indicates the design basis limit for the Torus water inventory (Suppression Pool volume/level) could have been exceeded and, therefore, primary containment may not have been operable because the inventory would not have been maintained for 30 days following a design basis accident. The 8 inch stop check valve was manufactured by Anchor-Darling Company, model E9982. On May 25, 1999 at 2129 hours, the NRC Operations Center was notified via the Emergency Notification System in accordance with 10 CFR 50.72(b)(2)(iii).

The requirement to submit the results of 10 CFR 50 Appendix J reports (including local leak rate tests) has been changed to a requirement to retain records of the Appendix J tests for NRC review upon request. NUREG-1022 (Rev. 1), Section 5.1.8 does not reflect the change to Appendix J. The guidance in Section 5.1.8, however, indicates that the results of tests should be included in one report. Therefore, this LER also includes RCIC check valve CK-1301-64.

The conditions were discovered during reactor refueling operations with the reactor mode selector switch in the REFUEL position. The reactor vessel pressure was zero psig and the coolant temperature was approximately 84 degrees Fahrenheit.

CAUSE

At the time of submittal of this LER, the root cause analyses for the MSIVs and check valve had not been completed. The results of the analyses will be identified in a supplement to this LER.

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CORRECTIVE ACTION

Immediate corrective action was taken to verify the integrity of the secondary containment. Work documents were initiated to repair MSIVs AO-203-1C, AO-203-2C and CK-1301-64 prior to plant startup from RFO-12. Valves exceeding their design leak rate were disassembled and inspected. Damaged or worn parts were reworked or replaced as appropriate. Following repair, the valves will be retested in accordance with the work control program to confirm that all design leak rates are met.

Any additional corrective action identified during the root cause investigation for AO-203-1C and 2C, 64 will be identified in the supplement to this LER.

SAFETY CONSEQUENCES AO-203-1C and 2C

A preliminary radiological assessment of the as-found test results for AO-203-1C and 2C indicated that 10CFR100 offsite dose limits and GDC19 control room dose limits would not be compromised using the design basis accident LOCA source term as specified in the PNPS appendix R.3 of the FSAR.

SAFETY CONSEQUENCES CK-1301-64

Although the accident analyses indicate the primary containment pressure would not remain constant at 45 psig (rather, a rapid increase and then gradual decrease to 27 psig within a couple of hours and less thereafter); the Suppression Pool volume/level would increase substantially for a large break loss of coolant accident (LOCA) but would not increase substantially for a small break LOCA. The Emergency Operating Procedures (EOPs) utilize Suppression Pool parameters including volume/level for operator actions.

Although the problem with the RCIC turbine steam exhaust check valve could have represented a significant degradation, the problem would not reasonably represent an unanalyzed condition because the primary containment pressure profile during a design basis accident is not a constant 45 psig for 30 days. In addition, Emergency Operating Procedures would likely control Suppression Pool volume/level such that it would not decrease to less than the level corresponding to the level of the exhaust lines that terminates in the Suppression Pool. Since level is maintained as directed by the EOPs there is no consequence to the health and safety of the public.

REPORTABILITY FOR AO-203-1C AND A0-203-2C:

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because the leakage through the in-series MSIVs AO-203-1C and AO-203-2C exceeded T.S. 4.7.A.2.a.4. The leakage is assumed to have occurred as a result of gradual degradation and therefore, not just at the time of the test. Therefore, although discovered while shut down, the 24 hour LCO specified by T.S. 3.7.A.6 is assumed to have been exceeded.

This report is also being submitted in accordance with 10 CFR 50.73(a)(2)(ii). NUREG-1022, (Rev 1) section 3.2.4 discussion criterion (1), subitems (f)(i) and (f)(iii) provide guidance that the problem is reportable via 10 CFR 50.73(a)(2)(ii), as significant degradation. For criterion (3): outside the design basis (a)(2)(ii)(B) applies because, assuming the problem had existed during power operation, the design function identified in the Updated Final Safety Analysis Report (UFSAR) section 5.2.1 (maintain radioactive releases less than 10 CFR 100 limits) and T.S. Bases 3/4.7.A that relates Part 100 limits to one percent per day primary containment leakage could have been compromised if a design basis accident had occurred with the as-found leak rates of valves AO-203-1C and AO-203-2C.

This report is also submitted in accordance with 10 CFR 50.73 subparts (a)(2)(v)(C) and (a)(2)(vii)(C) because both trains (inboard and outboard) of the primary containment system penetration X-7C were inoperable.

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REPORTABILITY FOR CK-1301-64:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B). The fact that check valve CK-1301-64 failed the leak rate test indicates the problem did not likely occur at the time of the surveillance and therefore, the problem likely existed for a period of time greater than the 24 hour LCO specified by Tech Spec 3.7.A.6. The design basis limit of 4 gpm (for CK-1301-64) was exceeded and therefore, is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(B).

SIMILARITY TO PREVIOUS EVENTS

A review for similarity was conducted of Pilgrim Station LERs. The review focused on LERs documenting MSIVs or RCIC valves that failed their LLRT. The review identified LER 86-011-00, "Leakage Past MSIV's in Excess of LLRT Criteria."

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

Components	Codes
Valve, Isolation	ISV
Valve, Check	SHV

Systems

Containment Leakage Control System	BD
Main Steam System	SB
Reactor Core Isolation Cooling System	BN