10 CFR 50.73

BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President — Nuclear

April 13, 1990
BECo Ltr. 90-056

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

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

Dear Sir:

The enclosed Licensee Event Report (LER) 90-004-00, "Local Leak Rate Test Results of Two Feedwater Check Valves in Excess of Limits", is submitted in accordance with 10 CFR Part 50.73.

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

R. G. Bird

GJB/bal

Enclosure: LER 90-004-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

Cert No P082894057

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Local Leak Rate Test Results of Two Feedwater Check Valves in Excess of 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 NAMES	DOCKET NUMBER(S)
0 3	1 6	9 0	9 0	— 0 0 4 —	0 0	0 4	1 3	9 0	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 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
Gary J. Basilesco - Senior Compliance Engineer	AREA CODE 5 0 8 7 4 7 - 8 5 3 4

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	B D	I S V	A 3 9 1	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

During the 1990 midcycle outage at Pilgrim Station, 10CFR50 Appendix 'J' local leak rate testing of the 18 inch Anchor/Darling, tilting disc, feedwater check valves (FWCVs) was conducted from 3/13/90 to 3/16/90. The "As-Found" test results showed leakage for two of the four feedwater check valves in excess of the limits of Pilgrim Nuclear Power Station Technical Specification Section 4.7.A.2.a.

A failure analysis team has been formed to investigate the root cause. Preliminary results of the investigation indicate a damaged soft seat and tolerance inconsistencies in the valves. Corrective action to date included replacing the soft seat material and realigning the hinge pins.

The event occurred during a midcycle outage while in cold shutdown. The Reactor Mode Selector Switch was in SHUTDOWN and reactor power level was zero percent. The reactor pressure was zero psig and the reactor coolant temperature was 105 degrees Fahrenheit. This event posed no threat to the health and safety of the public.

EVENT DESCRIPTION

During the 1990 midcycle outage at Pilgrim Station, 10CFR50 Appendix 'J' local leak rate testing of the 18 inch Anchor/Darling, tilting disc, feedwater check valves (FWCVs) was conducted from 3/13/90 to 3/16/90. The 'B' side FWCVs passed the local leak rate test with satisfactory results. The 'A' side FWCVs were tested with the following results:

Outboard	FWCV	261-62-A	16.4	SLM*
Inboard	FWCV	261-58-A	460.6	SLM*

* Values are listed as standard liters per minute (SLM) at 45 pounds per square inch gauge (psig).

These results indicate that for valves 261-62-A and 261-58-A, leakage was in excess of the limit (7.89 SLM at 45 psig or five (5) percent of the allowable operational leak rate) "for any one penetration or isolation valve", as set forth in Technical Specification Section 4.7.A.2.a(3). In addition, the "As Found" leakage for the "A" feedwater line was in excess of the limit (126.24 SLM at 45 psig or 60 percent of the maximum allowable test leak rate) for "all testable penetrations and isolation valves" as set forth in Technical Specification Section 4.7.A.2.a(2). The maximum allowable test leak rate (L_a) is one (1) percent/day at a pressure of 45 psig or 210.41 SLM. The allowable operational leak rate (L_t) is 75 percent of L_a or 157.8 SLM. Although the individual valves had leak rates in excess of Technical Specification limits, if the minimum path leakage methodology were applied (Ref. NRC Information Notice 85-71), the total penetration leakage (16.4 SLM) was well within the primary containment integrity limits (i.e. 60 percent of L_a).

Appendix 'J' testing is being conducted as outlined in BECo letters dated July 7, 1989 and March 14, 1990 regarding exemptions to 10CFR50 Appendix 'J' testing. To date, no other reportable conditions have been identified.

This event occurred during a midcycle outage while in cold shutdown. The Reactor Mode Selector Switch was in SHUTDOWN and the reactor power level was zero percent. The Reactor pressure was zero psig and the reactor coolant temperature was 105 degrees Fahrenheit. The event posed no threat to the health and safety of the public.

CAUSE

A failure analysis team has been established to investigate root cause. Preliminary results of the investigation indicate the following items contributed to the cause:

- Insufficient hinge pin to bushing tolerance prevented proper seating.
- Soft seat (on disc) was binding on hard seat when valve closed.
- Hinge pin cover gasket installation was not compensated for during disc alignment.

A supplemental LER will be submitted if the final root cause is significantly different than the preliminary cause or other reportable conditions are identified.

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

CORRECTIVE ACTION

The above items were addressed during the maintenance work performed on the valves in accordance with Procedure 3.M.4-49, "Feedwater Check Valve Maintenance". The soft seat material and bushings in both valves were replaced. The existing hinge pins were reworked and reinstalled. The valves were reassembled with the proper alignment and tolerances. The inboard feedwater check valve, 261-58-A, was retested with satisfactory results on March 30, 1990. The outboard valve, 261-62-A was retested with satisfactory results on April 3, 1990. Procedure 3.M.4-49 is being revised to improve tolerance and alignment information.

SAFETY CONSEQUENCES

This event posed no threat to the health and safety of the public.

The feedwater check valves are required to limit leakage from the primary containment during postulated Design Basis Accidents (DBAs) such that offsite radiation doses do not exceed the guideline values set forth in 10CFR100. Although individual valves had leak rates in excess of Technical Specification limits, the estimated dose consequences were well within 10CFR100 limits. In the event of a DBA with core damage, containment atmosphere could leak from the primary containment into the feedwater piping (located in the Turbine Building); however, this is unlikely because the feedwater system piping is expected to be intact and is likely to contain water seals formed where the piping elevation dips and rises. These water seals will minimize leakage from the primary containment and the resulting offsite radiation dose.

If cooling water flow into the reactor vessel through the feedwater lines is maintained during a DBA (with core damage), containment atmosphere will not leak from the primary containment through the feedwater lines and no offsite dose will result. In addition to feedwater flow, the High Pressure Coolant Injection (HPCI) System injects water into the reactor vessel through the 'B' feedwater line. Similarly, the Reactor Core Isolation Cooling (RCIC) System and Reactor Water Cleanup (RWCU) System inject water into the 'A' feedwater line. Therefore, operation of any combination of these systems during a DBA will minimize the leakage of containment atmosphere through the feedwater lines.

This report is submitted in accordance with 10CFR50.73(a)(2)(i)(B) because the leakages identified were in excess of the limit for any one penetration or isolation valve as set forth in the PNPS technical specifications.

SIMILARITY TO PREVIOUS EVENTS

A review of Pilgrim Station Licensee Event Reports issued since January 1984 was conducted. The review revealed one similar LER, 86-017-01.

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

LER 86-017-01 describes local leak rate test failures of the feedwater check valves. Corrective action included a redesign of the hinge pins, bushings and seat material. The leak tightness of the feedwater check valves has improved since this corrective action was implemented with the as-found leak rate of three of the four valves improving significantly, when compared to the 1986 test results.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

Component	Code
Valves, Isolation	ISV
System	
Containment Leakage Control System	BD
Feedwater System	SJ