10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

George W. Davis
Senior Vice President — Nuclear

July 11, 1991
BECo Ltr. 91- 090

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) 91-013-00, "Local Leak Rate Test Results of 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.

G W Davis

G. W. Davis

GJB/bal

Enclosure: LER 91-013-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
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Standard BECo LER Distribution

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Pilgrim Nuclear Power Station	0 5 0 0 0 2 9 3	1 OF 0 4

TITLE (4) Local Leak Rate Test Results of Feedwater Check Valves in Excess of Limits

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 6	1 3	9 1	9 1	0 1 3	0 0	0 7	1 1	9 1	N/A	0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9)	N			
POWER LEVEL (10)	0 0 0			
	[ ] 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) (B)	[ ] 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 AREA CODE	
Gary Basilesco - Senior Compliance Engineer	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)		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)

During Refueling Outage No. 8 between June 8 and June 13, 1991, 10 CFR 50 Appendix 'J' local leak rate testing of the 18 inch Anchor Darling, tilting disc, feedwater check valves was conducted. The as-found test results showed leakage for the four feedwater check valves to be in excess of the limits established in Pilgrim Nuclear Power Station Technical Specifications.

The Local Leak Rate Test Failure Analysis Team investigated root cause and recommended corrective actions. Results of the investigation indicate damaged soft seats and improper disc alignment caused the failures. Corrective action taken includes rebuilding the valves to the latest factory tolerances. This involved installing new discs using a new hinge pin design, replacing the soft seat material and rechecking alignment to ensure proper tolerances. The four valves were retested with satisfactory results between July 2 and July 4, 1991.

The leakages were identified during a refueling outage with the reactor mode selector switch in the REFUEL position. The Reactor Vessel (RV) was completely defueled with no fuel movement in progress. The RV/Refuel Cavity was flooded and the RV water temperature was approximately 82 degrees Fahrenheit. This event posed no threat to the health and safety of the public.

EVENT DESCRIPTION

During Refueling Outage No. 8 between June 8, 1991 and June 13, 1991, 10 CFR 50 Appendix 'J' local leak rate testing of the 18 inch Anchor Darling, tilting disc, feedwater check valves (FWCVs) was conducted. The results of the tests were as follows:

Outboard	FWCV	261-62-A	22.2 SLM*
Inboard	FWCV	261-58-A	394 SLM*
Outboard	FWCV	261-62-B	493 SLM*
Inboard	FWCV	261-58-B	218 SLM*

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

These results indicate leakage in excess of the limit of 7.89 SLM at 45 psig for "any one penetration or isolation valve" as specified in Technical Specification Section 4.7.A.2.a. The as-found leakage for the "B" feedwater line was also in excess of the limit of 126.24 SLM at 45 psig for "all testable penetrations and isolation valves" as specified in Technical Specification Section 4.7.A.2.a. In addition, when the "B" FWCVs were totaled using the minimum path leakage methodology (reference NRC Information Notice 85-71), the total penetration leakage of 218 SLM was greater than the maximum allowable test leak rate of one percent/day at a pressure of 45 psig or 210.41 SLM.

Failure and Malfunction Reports 91-236, 241, 253, and 254 were written to document the as-found leakage rates.

These leakage rates were identified during a refueling outage with the reactor mode selector switch in the REFUEL position. The Reactor Vessel (RV) was completely defueled with no fuel movement in progress. The RV/Refuel Cavity was flooded and the RV water temperature was approximately 82 degrees Fahrenheit.

CAUSE

The Local Leak Rate Test Failure Analysis Team investigated root cause and recommended corrective actions. Results of the investigation indicate the as-found alignment of the discs was not allowing for proper valve seating. The misalignment caused the soft seats on three of the four valve discs (58A, 62A, 62B) to bind on the hard seats as the valves closed. Inspection of the soft seat material showed evidence of degradation. In addition, the hinge pin-to-bushing tolerance was outside allowable limits thereby contributing to the disc misalignment and poor seating.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Pilgrim Nuclear Power Station	0 5 0 0 0 2 9 3	9 1	— 0 1 3	— 0 0	0 3 OF 0 4

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

The soft seat on the disc of the 58B valve showed no sign of binding or degradation. However, the disc was found misaligned such that the soft seat was traveling beyond the hard seat which resulted in unacceptable seating. Prior maintenance activities involving lapping of the hard seat resulted in contact between the hard seat and the disc retaining ring as the valve closed. This configuration allowed the soft seat to make good contact until the retaining ring eventually wore down. Although the soft seat material was in good condition, the disc and hard seats contained marks indicative of misalignment.

It should be noted that the valve discs have never been replaced in any of the four feedwater check valves. The valves have been reworked and reassembled in the past, in attempts to achieve proper alignment and tolerances. The reworking could be a contributor to the excess leakage as precise disc alignment becomes more difficult to achieve during each rework.

CORRECTIVE ACTION

The valve internals were rebuilt to the latest factory tolerances. This involved lapping the in-body seats, fitting new discs in each valve, matchmarking the hinge pin locations, boring new hinge pin holes in the discs, rechecking alignment, tightening tolerances on the soft seat protrusion and installing the discs using a new hinge pin design with a new flexitallic gasket-type hinge pin cover. With the exception of the in-body seats, valve internals have been replaced with new components. The valves were retested with satisfactory results between July 2 and July 4, 1991.

In addition, the Failure Analysis Team is investigating potential long term recommendations with respect to the feedwater check valves. This report will be updated if significant new corrective actions are taken.

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 10 CFR 100. Although individual valves had leak rates in excess of Technical Specification limits, the estimated dose consequences based on type B and type C tests were well within 10 CFR 100 limits. In order to determine the true as-found containment leak rate, the results of the overall integrated leak rate (type A) test are required. This report will be updated if the dose consequences change significantly after performance of the type A test.

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Pilgrim Nuclear Power Station	0 5 0 0 0 2 9 3	9 1	— 0 1 3	— 0 0	0 4 OF 0 4

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

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. In this case, no offsite dose will result from containment atmosphere leakage through the feedwater lines. In addition to feedwater flow, the High Pressure Coolant Injection System injects water into the reactor vessel through the 'B' feedwater line. Similarly, the Reactor Core Isolation Cooling System and Reactor Water Cleanup 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 10 CFR 50.73(a)(2)(i)(B) because the leakages identified were in excess of the limits established in the Pilgrim Station Technical Specifications.

SIMILARITY TO PREVIOUS EVENTS

A review of Pilgrim Station Licensee Event Reports issued since January 1984 was conducted. The review revealed two similar events in LER 86-017-01 and 90-004-00.

LER 86-017-01 describes local leak rate test failures of the four feedwater check valves. Corrective action included a redesign of the hinge pins, bushings and seat material. The leak tightness of the feedwater check valves improved after the corrective action was implemented as indicated by the test results reported in LER 90-004-00. The leak rate failures were discovered while shut down for repairs in June of 1986 with the reactor mode selector switch in the SHUTDOWN position.

LER 90-004-00 describes local leak rate failures of two of the feedwater check valves during the mid-cycle outage in March of 1990. The cause of the failures was damaged soft seats and tolerance inconsistencies in the valves. The soft seat material was replaced. The failures were discovered with the reactor mode selector switch in the SHUTDOWN position.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

Component	Code
Valves, Isolation	ISV

System	Code
Containment Leakage Control System	BD
Feedwater System	SJ