10 CFR 50.73

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

Ralph G. Bird
Senior Vice President — Nuclear

March 12, 1990
BECo Ltr. 90- 035

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) 90-001-00, "Two Reactor Coolart System Instrumentation Excess Flow Check Valves Inappropriately Verified Operable During Testing", 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

DWE/bal

Enclosure: LER 90-001-00

cc: Mr. William T. Russell
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

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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)
Pilgrim Nuclear Power Station	0 5 0 0 0 2 9 3	1 OF 0 6

TITLE (4) Two Reactor Coolant System Instrumentation Excess Flow Check Valves Inappropriately Verified Operable During Testing

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)
									N/A	0 5 0 0 0
0 2	0 9	9 0	9 0	0 0 1	0 0	0 3	1 2	9 0	N/A	0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0		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
Douglas W. Ellis - Senior Compliance Engineer	5 0 8 7 4 7 - 8 1 1 6 0

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	

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	0 1	9 0

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

On February 9, 1990 at 1830 hours, a 24 hour Limiting Condition for Operation (LCO) was entered because the operability of two (one-inch) reactor coolant system (RCS) instrument line excess flow check valves had been inappropriately verified during a Technical Specification required functional test on November 3, 1989. The other 80 RCS instrument line excess flow check valves were satisfactorily tested. The LCO was terminated at 2139 hours following NRC relief from Technical Specification 4.7.A.2.b.1.d for the two check valves.

The cause for this problem had not been determined when this report was prepared. A supplemental report will be submitted after the ongoing investigation for cause has been completed.

Corrective actions planned include the replacement of the two excess flow check valves during the mid-cycle outage that is scheduled to begin March 9, 1990. Interim compensatory measures being taken include increased controls for access and work in the vicinity of the instrument lines, routine visual operator inspections of the instrumentation lines, and issuing a radiation work permit to promptly allow the closing of the related manual isolation valves upstream of the check valves if necessary.

The LCO was entered during power operation with the reactor mode selector switch in the RUN position. The reactor power level was 100 percent. The Reactor Vessel (RV) pressure was 1035 psig with the RV water temperature at 549 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) and the problem posed no threat to the public health and safety.

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

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

EVENT DESCRIPTION

On February 9, 1990 at 1830 hours, a 24 hour Limiting Condition for Operation (LCO) was entered because the operability of two (one-inch) reactor coolant system (RCS) instrument line excess flow check valves had been inappropriately verified during a functional test on November 3, 1989. Technical Specification 4.7.A.2.b.1.d specifies that the operability of RCS instrument line (excess) flow check valves shall be verified at least once per operating cycle. The (82) RCS excess flow check valves are functionally tested via procedure 8.M.3-2, "Instrument Line Flow Check Valve Test".

The excess flow check valves, CK-125A and CK-125B, were installed new in September 1987. The test procedure was inappropriately signed as completed on November 4, 1989 based on a previously written memorandum that indicated the two check valves were not required to be functionally tested until the next refueling outage. The memorandum was written because sufficient flow (i.e., greater than two GPM), needed to actuate the check valves (CK-125A/B), could not be achieved during post installation testing due to instrument line configuration.

Failure and Malfunction Report 90-32 was written to document the problem. The NRC Operations Center was notified in accordance with 10 CFR 50.72 on February 9, 1990 at 1925 hours.

The LCO was entered during power operation with the reactor mode selector switch in the RUN position. The reactor power level was approximately 100 percent. The Reactor Vessel (RV) pressure was 1035 psig with the RV water temperature at 549 degrees Fahrenheit.

The LCO was terminated as a result of a formal relief request made by Boston Edison Company from Technical Specification 4.7.A.2.b.1.d for the excess flow check valves (CK-125A/B). The request was subsequently discussed with the NRC (offices of Region I and NRR) via a teleconference call that began on February 9, 1990 at approximately 1925 hours. The request was granted by the NRC at approximately 2115 hours. The 24 hour LCO was terminated on February 9, 1990 at 2119 hours. The relief extends to the mid-cycle outage that is scheduled to begin on March 9, 1990.

BACKGROUND

The two (one-inch) excess flow check valves, CK-125A and CK-125B, are installed in reactor coolant system instrument (high side) lines that extend from primary containment penetrations X-82A and X-82B, respectively. Upstream of these penetrations, and within primary containment, each instrument line includes a restricting orifice. Downstream of these penetrations, and outside primary containment, each line includes a manual isolation valve, HO-126A or HO-126B, and in-series excess flow check valve, CK-125A or CK-125B. From the flow check valve, the one-half inch instrument line extends to racks that house sensors (transmitters) and local indicators that function to monitor Reactor Vessel pressure and water level.

The sensors provide signals to the circuitry of systems that include the following: Reactor Protection, High Pressure Coolant Injection, Reactor Core Isolation Cooling, Automatic Depressurization, Core Spray, Residual Heat Removal, Primary Containment Isolation Control, and Reactor Building Isolation Control.

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

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

Technical Specification 3.7.A.2.a.(5) specifies that at least one isolation valve (e.g., HO-126A) be secured in the isolated position for each instrument line containing an inoperable flow check valve (e.g., CK-125A). If the isolation valves upstream of the excess flow check valves (CK-125A/B) were to be closed, the number of operable instrument channels and trip systems would be less than the minimum specified by the Technical Specification(s) for the related systems.

CAUSE

A Human Performance Evaluation System (HPES) investigation is being conducted to determine the cause(s) for the problem and is expected to be completed during the week of March 12, 1990.

A supplemental report will be submitted after the investigation is completed and is expected to be submitted by June 1, 1990.

CORRECTIVE ACTION

A modification (PDC 90-13) has been prepared for the replacement of the excess flow check valves (CK-125A/B). The replacement (one-inch) excess flow check valves have a (flow) actuation (i.e., two GPM or less) that is lower than the currently installed check valves that actuate at a flow of greater than two GPM. The modification will be implemented during the mid-cycle outage that is scheduled to begin on March 9, 1990. The currently installed check valves (CK-125A/B) were manufactured by Dragon Valve Incorporated, part number 1596ON-1; the replacement check valves are similar, but are part number 1596ON-2.

Other corrective actions are indeterminate at this time in that the investigation for cause has not been completed.

INTERIM COMPENSATORY MEASURES

The following compensatory measures were initiated on February 9, 1990 at approximately 2057 hours and are being taken during continued operation:

- Access controls were increased for areas in the vicinity of the instrument lines from the penetrations (X-82A/B) upstream of the check valves (CK-125A/B) to the related downstream instrumentation racks. The increased controls include roping and the posting of appropriate notices in the areas.

- Controls were increased for work or maintenance in the vicinity of the instrument lines from the penetrations (X-82A/B) upstream of the check valves (CK-125A/B) to the related downstream instrumentation racks. The increased controls include authorization by the shift Watch Engineer for work or maintenance in the vicinity of the instrument lines.

- A Radiation Work Permit (RWP 90-161) was issued to promptly allow Operations personnel to close the related upstream isolation valve(s), HO-126A/B, if an instrumentation line break were to occur downstream of a flow check valve(s).

- The areas in the vicinity of the instrumentation lines from the penetrations (X-82A/B) upstream of the flow check valves to the related downstream instrumentation racks are being visually inspected for leakage at least once-per-shift.

SAFETY CONSEQUENCES

This condition posed no threat to the public health and safety.

The RCS instrument line excess flow check valves, including excess flow check valves CK-125A/B, provide two functions:

- The active function is part of Technical Specifications 3/4.7.A.2 because the check valves function to reduce an RCS leak into the Reactor Building (secondary containment) if an instrument line break were to occur downstream of a check valve (e.g., CK-125A). The safety analysis for a potential instrument line break is provided in the Pilgrim Station Updated Final Safety Analysis Report (FSAR) section 5.2.3.5.3. This section describes the instrument line containment boundary as an upstream orifice located inside primary containment and a downstream instrument line flow check valve (e.g., CK-125A) located outside primary containment.

- The passive function is not specifically a part of the Pilgrim Station Technical Specifications because the instrument lines, including flow check valves CK-125A/B, function to provide a passive pressure boundary as part of the pathway for sensing Reactor Vessel pressure and water level.

The active function of primary containment instrument line excess flow check valves is tested in accordance with procedure 8.M.3-2, "Instrument Line Flow Check Valve Test". The other 80 instrument line flow check valves were functionally tested with satisfactory results during the October-November 1989 outage.

Routine and periodic assurance of the excess flow check valves' (CK-125A/B) passive function is demonstrated as follows:

- Routine once-per-shift operator tours within the Reactor Building are performed in accordance with procedure 2.1.16, "Nuclear Power Plant Operator Tour", Attachment 2 (OPER-8). These tours include various checks in the vicinity of the instrument lines from the penetrations (X-82A/B) upstream of the check valves (CK-125A/B) to the related downstream instrumentation racks. These tours also include a check for Reactor Vessel pressure and water level indications at the applicable instrument.

- Routine once-per-shift checks of Reactor Vessel pressure and water level indications in the Control Room are performed in accordance with procedure 2.1.15, "Daily Surveillance Log", Attachment 1. The indications in the Control Room are derived from instrumentation including transmitters downstream of flow check valves CK-125A/B.

- Routine trending of Reactor Vessel pressure and water level transmitters, by the Systems Engineering Division, includes monitoring the performance and response of transmitters downstream of the flow check valves (CK-125A/B).

- Periodic surveillance testing of instrumentation downstream of the flow check valves (CK-125A/B) is performed in accordance with procedures. The procedures include: 8.M.1-32.1 (typical), "Analog Trip System - Trip Unit Calibration Cabinet C2228-A1", 8.M.1-32.5 (typical), "Analog Trip System - Trip Unit Calibration - Cabinet C2233A, Section A", 8.M.2-6.1, "Reactor Pressure Readout", 8.M.2-6.3, "Reactor Level Readout", 8.M.2-8.1 (typical), "Calibration of ATS Transmitters Rack C2205", and 8.M.2-8.6, "Calibration of ATS Transmitters Rack C2251 and C2252".

  These routine and periodic activities provide assurance that the passive function of the check valves (CK-125A/B) is functional.

The failure of an excess flow check valve body, or the instrument line upstream of the check valve, could result in a maximum leakage of 20 GPM into the Reactor Building. The leakage, limited by the upstream orifice, is within the makeup capacity of the Control Rod Drive or Feedwater Systems. The amount of steam resulting from a 20 GPM leak into the Reactor Building does not endanger the integrity of the Reactor Building.

- If a leak were to occur and the Reactor Building is not isolated, a significant pressure increase would not occur because of the relatively high Reactor Building ventilation exhaust rate.

- If a leak were to occur and the Reactor Building is isolated, the operation of either one of the two Standby Gas Treatment System trains would prevent the Reactor Building from exceeding its design value for internal (positive) pressure.

  The total radiological dose at the site boundary resulting from a 20 GPM leak with either of these two Reactor Building configurations would be substantially below the guidelines of 10 CFR Part 100.

Excess flow check valves CK-125A/B and the related upstream flow restricting orifices were installed new in 1987. Except for periodic surveillance testing, the flow restricting orifices and check valves are not subjected to a fluid flow environment. Therefore, it is reasonable to assume that these flow restricting orifices have not degraded in any way which could result in an increase in the limiting flow of 20 GPM.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because the operability of excess flow check valves CK-125A/B was not verified as specified by Technical Specification 4.7.A.2.b.1.d.

This report is also submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because the related manual isolation valves HO-126A/B, located upstream of excess flow check valves CK-125A/B, were not secured in the isolated position as specified by Technical Specification 3.7.A.2.a.(5). This action was not taken because of the relief granted from Technical Specification 4.7.A.2.b.1.d for excess flow check valves CK-125A/B.

SIMILARITY TO PREVIOUS EVENTS

Similarity to previous events is indeterminate at this time in that the investigation for cause has not been completed.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Valve (CK-125A/B)	V
Valve, Control, Flow (CK-125A/B)	FCV

SYSTEMS	
Containment Leakage Control System	BD
Incore Monitoring System	IC