10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
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George W. Davis
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July 5, 1991
BECo Ltr. 91-085

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 supplemental Licensee Event Report (LER) 90-001-01, "Two Reactor Coolant 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.

G. W. Davis

RLC/bal

Enclosure: LER 90-001-01

cc: Mr. Thomas T. Martin
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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 WITH 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 1 0

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)
0 2	0 9	9 0	9 0	— 0 0 1 —	0 1	0 7	0 5	9 1	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 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) 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	
Robert L. Cannon - Senior Compliance Engineer	5 0 8	7 4 7 - 8 3 2 1

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

CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS

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)

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 2119 hours following NRC relief from Technical Specification 4.7.A.2.b.1.d for the two check valves.

The cause of this event included nomenclature errors in the 1987 excess flow check valve surveillance procedure used as a post modification test, and an inappropriate sign-off of the November 1989 surveillance.

Interim compensatory measures taken included 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. Corrective actions included the replacement of the two excess flow check valves during the mid-cycle outage that began on March 11, 1990.

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.

REASON FOR SUPPLEMENT

This supplement is submitted to meet our commitment to submit a supplemental report for the cause and associated corrective action which had not been fully determined when the initial report was submitted.

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 (F&MR) 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 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 extended to the mid-cycle outage that began on March 11, 1990.

BACKGROUND

Pilgrim Station has 82 installed excess flow check valves. Eighty (80) were manufactured by Chemiquip and two (2) were manufactured by Dragon. The Dragon valves which are the subject of this report were installed in one inch instrument lines outside of containment. The Dragon valves were installed on the instrument reference legs as part of a modification in 1987.

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

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.

The operability of the excess flow check valves is verified by performing surveillance testing in-situ with the plant in a cold pressurized condition by venting the instruments downstream of the check valves to cause them to check.

Following installation of the two Dragon valves in 1987, a post modification surveillance test was conducted to prove operability of the 82 excess flow check valves. In July of 1988, BECo reviewed its past surveillances in preparation for startup. This review identified that the two new Dragon valves had not actually been tested in-situ because achievable test flow was limited to approximately 2 gpm while the design actuation flow was 5-6 gpm. The 80 Chemiquip valves were successfully tested.

The surveillance test was approved based on a July 1988 BECo memorandum that justified use of the manufacturer's test to prove operability until RFO-8 when the two Dragon valves would be replaced. At the time, RFO-8 was scheduled to begin in December 1990. The definition of operating cycle in effect during July 1988 required the Dragon valves to be tested before restart from RFO-8. Within this context, the recommendation to replace the valves in RFO-8 was sound.

In November of 1988, BECo's definition of the surveillance interval for operating cycle was revised to 18 months + 25%. This revision changed the due date of the next excess flow check valve surveillance from December 1990 to October 1989. Because of this change, the plant was shut down in October of 1989 to conduct the check valve surveillances as well as others.

During the October 1989 outage the 80 Chemiquip excess flow check valves were tested and an unsuccessful attempt to check the Dragon excess flow check valves occurred. Questions concerning seat leakage criterion for the Chemiquip valves resulted in an engineering service request being written. Revised criterion was conveyed to the Station on November 3, 1989 with the July 1988 memo attached. The memo stated that operation until RFO-8 was acceptable without Dragon valve testing. It was misconstrued to be an adequate basis for waiving of the surveillance test. The test was signed-off referencing the engineering memo as a basis for not testing the 2 Dragon valves. The minor safety significance of the issue drew attention away from a Technical Specification compliance issue.

In response to an NRC question, on January 12, 1990, BECo initiated a records review to find justifying documents for waiving the November 3, 1989 Dragon valve operability test.

The review of the applicable documents on February 2, 1990 concluded that while no safety issue existed, the approach used to waive the test was not valid. A clarification to the Technical Specifications was proposed. On February 9, 1990 the Operations Review Committee (ORC) reviewed the proposed Technical Specification clarification and ORC agreed that no safety issue existed, but that waiver of the test was contrary to Technical Specification compliance and not within the scope of a clarification. The ORC chairman promptly notified the Station Director.

IMMEDIATE CORRECTIVE ACTION

Immediate corrective steps were initiated including entering a 24 hour LCO and requesting Technical Specification (TS) relief from TS 4.7.A.2.b.1.d. Compensatory measures were established to assure integrity of the two (2) valves and a night order entry was made to ensure appropriate operations personnel understood the issue and the compensatory measures. These measures included:

- 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 included 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 were visually inspected for leakage at least once-per-shift.

These compensatory measures continued until the plant was shutdown on March 11, 1990 for a planned mid-cycle outage.

CAUSE AND CORRECTIVE ACTION

In the longer term, the two (2) excess flow check valves were replaced with testable valves during the mid-cycle outage that began March 11, 1990.

To bound the issue a review of completed surveillance procedures was initiated to ensure similar problems did not exist elsewhere. Concurrent with this review, the Systems Engineering Division identified a related Technical Specification compliance issue associated with the Technical Specification requirements for Primary Containment isolation valves MO-1001-60 and MO-1001-63. A detailed discussion of that compliance issue was provided in Licensee Event Report (LER) 90-006-00.

An investigation was also conducted using the Human Performance Evaluation System (HPES). The HPES report was completed on March 23, 1990.

Each of the processes that were involved in this issue were reviewed for adequacy. The four processes reviewed were:

- Modification process.

- Surveillance process.

- Failure and Malfunction (F&MR) process.

- Technical Specification clarification process.

MODIFICATION PROCESS

Within the modification process, several issues were examined. The design check flow rate was specified to be greater than the system would produce. Although the valves would perform their function in the event of an instrument line break, the valves were not testable as installed. This was an isolated error in 1985. The responsible design engineer was counselled.

The applicable surveillance test procedure was not identified for revision when the design change package was reviewed in 1985. In 1989 as part of an ongoing Quality Assurance (QA) Audit program, the QA Department identified a need for improvement in the identification of procedures affected by design changes. The modification process was revised to require two reviews in this area.

In 1987, the post-modification test did not identify the inability to perform the in-situ test. This was a problem with the specific surveillance test procedure used, and not a modification process issue. The procedure was corrected.

SURVEILLANCE PROCESS

Review of the surveillance process raised two issues:

- Nomenclature errors in the 1987 excess flow check valve surveillance procedure used as a post modification test.
- An inappropriate sign-off of the November 1989 surveillance.

The nomenclature errors were corrected shortly after discovery in the Summer of 1988. During this same time, a strengthened procedure validation process was established that would identify problems of this nature prior to procedure implementation. In addition, during this period procedure walkdowns were being conducted to verify surveillance procedure nomenclature. With today's improved procedure writers' guide we have corrected this programmatic issue with procedure review.

The "Conduct of Operations" Procedure (PNPS 1.3.34) required Technical Specification surveillances that have exceptions shall be independently reviewed by the Nuclear Operations Supervisor or Shift Technical Advisor prior to sign-off. This instruction was and is adequate, and had the question on the review form been addressed, this may have prevented the sign-off of the surveillance. This point was reviewed with the operating staff as part of the procedure compliance/attention to detail upgrade effort that was ongoing throughout the latter half of 1989. Extensive management review has shown this effort to be effective.

F&MR PROCESS

A review of the F&MR focused on why the F&MR was closed out without identifying the Technical Specification compliance issue. The F&MR form itself requires a reportability review. However, the work instruction which the compliance engineer used to accomplish this review required strengthening. This has been accomplished. A review of approximately 150 F&MRs indicated that similar situations do not exist.

TECHNICAL SPECIFICATION CLARIFICATION PROCESS

The TS clarification process was also reviewed. This review indicated the process works but needs strengthening. The inappropriately proposed clarification was identified by the ORC. The Regulatory Affairs Manager counselled Licensing and Compliance Division personnel regarding the need for scrupulous, independent review of regulatory guidance to ensure the requirements of the Technical Specification are met. A review by the Licensing Division Manager verified that current Technical Specification Clarifications were appropriate as written. The clarification process was strengthened by obtaining an SRO (or equivalent) review of the proposed clarification prior to submittal for ORC review.

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 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 had not degraded in any way that 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

A review of Pilgrim Station Licensee Event Reports (LER) issued since 1984 in accordance with 10 CFR 50.73(a)(2)(i) which were caused by personnel error was performed. The following events were determined to be similar because the cause was a misunderstanding or incorrect judgment regarding the applicability of T.S. surveillance requirements.

- LER 90-007-00

  On May 7, 1990, as a result of a performance review of surveillance 8.A.2 "Drywell to Suppression Chamber Vacuum Breaker Leakage Rate Test (1.25 psig)", it was determined that when restarting from Refueling Outage No. 7 (RFO 7) in December 1988, surveillance 8.A.2 was not performed at the appropriate point in the startup process. The surveillance had expired during the outage for RFO-7 and should have been performed prior to reactor criticality in accordance with Technical Specifications. The surveillance is required to be performed once per refueling outage and quarterly. The surveillance was performed during the outage in December 1987. The quarterly surveillance was satisfactorily performed shortly after Drywell inerting in March 1989.

  The cause was determined to be a misunderstanding of Technical Specification surveillance requirements and an incorrectly scheduled surveillance in the Master Surveillance Tracking Program (MSTP).

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Pilgrim Nuclear Power Station	0 5 0 0 0 2 9 3	9 0	— 0 0 1	— 0 1	1 0 OF 1 1

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

- LER 90-006-00

  On March 30, 1990, the position of one of the two inoperable, in-series, Primary Containment System isolation valves in the Head Spray line had not been recorded daily as required by Technical Specification (T.S.) 4.7.A.2.b.2 Valves MO-1001-60 and MO-1001-63 were placed in the closed position with their respective circuit breakers (72-844 and 52-2053) opened (de-energized) in conjunction with work associated with Plant Design Change 86-20 which cut and capped the Head Spray piping in March 1986. The cause was a misunderstanding (personnel error) of actions to be taken to implement PDC 86-20.

- LER 87-004-00

  On February 18, 1987, at approximately 1610 hours, it was determined that the dry chemical fire suppression system associated with a piping trench below the "A" emergency diesel generator had been inoperable since December 21, 1986. Contrary to the plant Technical Specifications, the appropriate limiting condition for operation was not complied with from approximately 1900 hours on February 6, to 0845 hours on February 13, 1987, during which time the suppression system was required to be operable. The root cause of this event was cognitive personnel error in the failure of the individuals involved to identify the Technical Specification requirements associated with the system.

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