10 CFR 50.73

BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President — Nuclear

July 7, 1989
BECo Ltr. 89-095

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) 89-019-00, "Reactor Core Isolation Cooling System Made Inoperable per Technical Specifications due to Inoperable Area Temperature Switch", 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.

K. L. Highfill
for R. G. Bird

DWE/bal

Enclosure: LER 89-019-00

cc: Mr. William 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

Cert No P082897521

IE22
1/1

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): Reactor Core Isolation Cooling System Made Inoperable Per Technical Specifications Due to Inoperable Area Temperature Switch

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	0 9	8 9	8 9	0 1 9	0 0	0 7	0 7	8 9	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 2 5

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	
Douglas W. Ellis - Senior Compliance Engineer	5 0 8	7 4 7 - 8 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
X	BN	BLK	W172	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)

On June 9, 1989 at 0500 hours, the Reactor Core Isolation Cooling (RCIC) System was made inoperable and a 24 hour Limiting Condition for Operation (LCO) began in accordance with a Technical Specification. The system was made inoperable because a lifted lead for an area temperature switch was discovered during a monthly functional test.

The primary cause for the lifted lead was a surveillance procedure that was signed as completed on May 14, 1989 with unsatisfactory results. Factors contributing to the lifted lead included a terminal block (Weidmuller six point type SAK6N) compression fitting that was broken during the previous surveillance (on May 14, 1989), and Instrumentation and Control (I&C) Supervisor failure to identify the lifted lead in accordance with the approved procedure for lifted leads and jumpers.

The terminal block was replaced and the RCIC System was returned to service on June 9, 1989 at 2150 hours. The responsible I&C Supervisor has since received specific training on the applicable procedure. Additional I&C Supervisor training regarding Technical Specifications has been planned. Approximately 379 other completed surveillance procedures have since been reviewed with no similar problems identified. Applicable surveillance procedures will be revised to clarify the acceptance criteria during the systematic two year procedure review process. The High Pressure Coolant Injection System was operable during the period of time when the RCIC System was required to be operable.

This condition was discovered during power operation at 25 percent reactor power. The reactor mode selector switch was in the RUN position. This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) and this condition posed no threat to the public health and safety.

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	8 9	— 0 1 9	— 0 0	0 2 OF 0 6

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

EVENT DESCRIPTION

On June 9, 1989 at 0500 hours, the Reactor Core Isolation Cooling (RCIC) System was made inoperable and a 24 hour Limiting Condition for Operation (LCO) began in accordance with Technical Specification Table 3.2.B Note 1 (one). The system was made inoperable in accordance with Technical Specification Table 3.2.B Note 2 (two) that specified the closing of the system's isolation valves (MO-1301-16 and MO-1301-17). This action was taken because a taped, disconnected lead for an area temperature switch (TS-1360-14D) was discovered during a functional test of the RCIC System area temperature switches. The (monthly) functional test began on June 9, 1989 at approximately 0330 hours and was being conducted in accordance with Procedure 8.M.2-2.6.3 (Rev. 13) Attachment A. "RCIC Steam Line High Temperature Instrument Functional Test".

A portion of the steam supply piping to the RCIC System turbine is located in a room that is within the Reactor Building (at elevation 23 feet). Located within the room are four area temperature switches that provide a detection function if a RCIC System turbine steam supply pipe break were to occur in the room. The temperature switches are part of the Primary Containment Isolation Control System (PCIS)/RCIC System logic circuitry that automatically controls the Primary Containment System (PCS)/RCIC System turbine steam supply isolation valves. The logic circuitry is arranged into two channels ('A' and 'B'). Each channel is capable of automatically closing the turbine steam supply isolation valves (MO-1301-16 and -17). The Channel 'A' area temperature switches TS-1360-14C and -16C are connected in-series. Similarly, the Channel 'B' temperature switches TS-1360-14D and -16D are connected in-series. The lifted lead effectively eliminated the Channel 'B' area temperature switches from providing a pipe break detection function in the room and resulted in less than the minimum number (two) of operable RCIC System instrument Channels ('A' and 'B') for a pipe break in the room.

Failure and Malfunction Report 89-221 was written to document the discovery. The NRC Operations Center was notified on June 9, 1989 at 0645 hours. Operability testing of the High Pressure Coolant Injection System began in accordance with Technical Specification 3.5.D.2 on June 9, 1989 at 0555 hours and was satisfactorily completed on June 9, 1989 at 0755 hours.

The discovery occurred during power operation with the reactor mode selector switch in the RUN position. The Reactor Vessel (RV) pressure was 950 psig with the RV water temperature at approximately 538 degrees Fahrenheit. The reactor power level was 25 percent.

CAUSE

The primary cause for the condition was a surveillance procedure (8.M.2-2.6.3 Attachment B) that was signed as completed on May 14, 1989 with unsatisfactory results. Factors contributing to the cause were a combination of: a terminal block compression fitting that was broken during the previous surveillance performed while shutdown (on May 14, 1989); less than adequate communication; and utility I&C Supervisor failure to follow the approved lifted lead and jumper procedure (1.5.9.1).

The cause for the lifted lead was an inadequate electrical connection of the temperature switch to its terminal block that occurred during the May 14, 1989 calibration process. The temperature switch (TS-1360-14D) was routinely replaced with another calibrated temperature switch at step 56 of the procedure (8.M.2-2.6.3 Attachment B). The replacement included inserting the leads of TS-1360-14D into its terminal block and torquing the two terminal screws to the prescribed value. After the in-series temperature switch (TS-1360-16D) was similarly replaced, TS-1360-14D was jumpered (per procedure) for a functional test of TS-1360-16D. The functional test consists of heating the temperature switch and verifying that the related Channel 'B' tripping relay (13A-K32) is energized and is de-energized after the heat is removed from the temperature switch. After the satisfactory completion of the functional test of TS-1360-16D, the jumper was removed (per procedure) from TS-1360-14D and TS-1360-16D was then jumpered (per procedure) for a similar functional test of TS-1360-14D. After TS-1360-14D was sufficiently heated, the related tripping relay (13A-K32) did not energize (step 65.a). Subsequent investigation revealed an inadequate electrical connection for one of the leads of TS-1360-14D. The inadequate connection was due to a broken compression fitting of the terminal block (Weidmuller Terminations Incorporated six point type SAK6N).

The shift I&C Supervisor was notified of the problem with the terminal block connection. The use of a spare connection on the same terminal block was considered (I&C Supervisor and I&C technicians stationed at TS-1360-14D) but was not used because the resulting configuration would not have been consistent with approved drawings. Consequently, the lead of TS-1360-14D was lifted and taped and the steps for functional testing of TS-1360-14D were marked with an asterisk. The corresponding entry in the discrepancy section of the procedure (Attachment B) indicated, "terminal broken, did not test, put in MR (Maintenance Request) to replace terminal block". The as-left condition of TS-1360-14D (lifted and taped lead) was not identified in the procedure (Attachment B steps 64 through 67) or discrepancy section.

After the other area temperature switches were satisfactorily replaced and functionally tested, the completed surveillance procedure (8.M.2-2.6.3 Attachment B) was signed by the shift I&C Supervisor and presented to the senior shift licensed operator (Nuclear Watch Engineer) for signature. The I&C Supervisor took this action believing that the Maintenance Request would be appropriately prioritized because the problem involved the RCIC System. The I&C Supervisor verbally indicated to the shift Nuclear Watch Engineer (NWE) that the temperature switch was not tested. The shift NWE signed the procedure because the procedure and discrepancy section did not indicate a lifted or taped lead, the plant was then in the cold shutdown condition, and a Maintenance Request had been written. The I&C Supervisor failed to identify the lifted lead to the NWE and consequently, the lifted lead was not controlled and entered into the lifted lead and jumper log in accordance with procedure 1.5.9.1, "Lifted Leads and Jumpers".

The Maintenance Request (MR 89-45-178) indicated that a terminal block was accidently broken while performing the activity (8.M.2-2.6.3) and that the terminal block needed to be replaced. The initial information entered on the Maintenance Request did not indicate a problem with the RCIC System or a temperature switch, and did not indicate that a lead was lifted or taped. Consequently, the Maintenance Request was assigned a priority (three) that did not provide for appropriate action, i.e. terminal block replacement prior to startup (May 25, 1989). The initial information identified on the Maintenance Request form was entered by one of the I&C Technicians assigned to the activity (8.M.2-2.6.3) but not at the location of TS-1360-14D.

CORRECTIVE ACTION

The following corrective actions have been taken or planned:

The terminal block was replaced and a post work functional test of the affected Channel 'B' area temperature switches (TS-1360-14D and -16D) was performed per procedure 8.M.2-2.6.3 (Attachment A). The functional test was completed with satisfactory results on June 9, 1989 at 2100 hours. The RCIC System was declared operable on June 9, 1989 at 2150 hours. A functional test of the Channel 'A' and Channel 'B' area temperature switches (including TS-1360-14C, -16C, -14D, and -16D) was performed per procedure 8.M.2-2.6.3 (Attachment A). This functional test was completed with satisfactory results on June 9, 1989 at 2220 hours.

The responsible I&C Supervisor has since received specific training regarding the administrative controls of procedure 1.5.9.1, "Lifted Leads and Jumpers". The other I&C personnel (Supervisors, technicians, and apprentices) and Electrical Maintenance personnel (Supervisors and engineers) have received or will receive similar training. Moreover, the I&C Supervisors and Electrical Maintenance Supervisors will receive additional training, including applicable Technical Specifications training, that has been planned for this year (fourth quarter).

Approximately 379 other completed surveillance test procedures have since been reviewed. The completed test procedures were reviewed for discrepancies that could have possibly affected the operability of the tested system(s). The review identified no discrepancies that had an adverse affect on the operability of the tested system(s).

Revision of surveillance procedures is part of the systematic two year procedure review program. The program includes a checklist for evaluation of acceptance criteria. The long term revision of the surveillance procedures, together with the interim corrective measures being taken, provides reasonable assurance that a similar event will not occur.

Interim corrective measures are being taken until the revision(s) of the surveillance procedures have been completed. The measures include the use of a checklist. The checklist is used by the shift Nuclear Watch Engineer (NWE) when a completed surveillance test procedure is submitted to the NWE for signature. The checklist provides for a documented followup review by the shift Nuclear Operating Supervisor (NOS) or Shift Technical Advisor (STA) if any test exceptions, notes or discrepancies are identified. The NOS or STA followup review includes documented disposition of the exceptions, notes or discrepancies regarding possible impact on operability and applicable Technical Specification(s).

SAFETY CONSEQUENCES

The lifted lead posed no threat to the public health and safety.

The lead of the temperature switch TS-1360-14D was lifted and taped on May 14, 1989 while the plant was in a shutdown condition and when the operability of the RCIC System was not necessary. The lifted lead impacted the operability of a portion of the RCIC System Channel 'B' logic circuitry only. The RCIC System Channel 'A' logic circuitry, including redundant area temperature switches TS-1360-14C and -16C, was not affected and was capable of providing the same temperature detection and isolation function.

The High Pressure Coolant Injection (HPCI) System was operable during the period of operation that the lead was lifted while the RCIC System was required to be operable (May 25, 1989 to discovery) and during the period that the RCIC System was made inoperable.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because Technical Specification Table 3.2.B Notes 1 (one) and 2 (two) were (unknowingly) not complied with from May 25, 1989 to June 9, 1989 (discovery).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review was focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(i) involving the RCIC System or a similar condition(s) that was contrary to a Technical Specification(s).

The review identified somewhat similar conditions reported in LERs 50-293/85-020-00, 87-004-00, and 89-014-00.

For LER 85-020-00, unacceptable results of an offsite laboratory analysis of the Emergency Diesel Generator (Fuel) Oil Storage Tank 'A' were not identified to Operations personnel for three days. The fuel analysis results for water or sediment were slightly greater than the acceptance criteria of less than 0.05 percent. Because of the notification delay compensatory measures specified by a Technical Specification (4.5.F.1) for an inoperable EDG ('A') were not initiated immediately after the analysis results were received. The cause was attributed to (utility non-licensed) Chemistry technician error due to an inadequate (Chemistry) procedure. The procedure (7.1.36, "Diesel Generator Fuel Oil Sampling & Quality Analyses") did not indicate that the Nuclear Watch Engineer must be notified if the fuel oil analysis is not within the applicable Technical Specification (4.9.A.1.e). The condition existed during operation at 100 percent reactor power.

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	8 9	— 0 1 9	— 0 0	0 6 OF 0 6

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

For LER 87-004-00, the pressure in the Diesel Generator 'A' trench portion of the Dry Chemical Fire Protection System was found by a utility non-licensed operator to be 5 (five) psig below the minimum pressure (275 psig) during a daily surveillance on December 21, 1986. A Maintenance Request was written at that time to correct the problem but was assigned an inappropriate system number (i.e. Diesel Generator instead of the Fire Protection System). On February 18, 1987 it was discovered that appropriate Fire Protection compensatory measures were not taken in accordance with a Technical Specification (3.12) for the low pressure condition during the period from February 6, 1987 (at 1900 hours) to February 13, 1987 (at 0845 hours) when the Dry Chemical System was required to be operable. The cause was attributed to utility personnel error. A contributing factor was the assignment of the inappropriate system number to the Maintenance Request. The condition existed while shutdown.

For LER 89-014-00, the inboard PCS/RCIC System turbine steam supply isolation valve (MO-1301-16) was in the open position for approximately 40 minutes while the in-series outboard isolation valve was inoperable (valve MO-1301-17 open with its breaker open). While valve MO-1301-17 was inoperable, the in-series valve MO-1301-16 should have been deactivated in the closed position per Technical Specification 3.7.A.2.b. The cause was attributed to a combination of RCIC System logic system functional test procedure (8.M.2-2.10.11.1) error, utility operator (licensed and non-licensed) error, and tagging procedure (1.4.5) deficiencies. The condition existed during operation at 25 percent reactor power.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Block	BLK
Block, Disconnect	DBLK
Switch, Temperature	TS

SYSTEMS	
Engineered Safety Features Actuation System (PCIS)	JE
Leak Monitoring System	IJ
Reactor Core Isolation Cooling (RCIC) System	BN
Temperature Monitoring System	IM