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

10 CFR 50.73

**E. T. Boulette, PhD**
Senior Vice President — Nuclear

May 26, 1994
BECo Ltr. 94-065

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

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

The enclosed Licensee Event Report (LER) 94-003-00, "False Low Reactor Vessel Water Level Signal While Shut Down During Control Rod Drive System Venting", 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.

E T Boulette

E. T. Boulette, PhD

RAG/lam/9400300

Enclosure: LER 94-003-00

cc: Mr. Thomas T. Martin
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Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution



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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

# LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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TITLE (4)

False Low Reactor Vessel Water Level Signal While Shut Down During Control Rod Drive System Venting

| 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 NAME N/A | DOCKET NUMBER 05000 |
| 04 | 27 | 94 | 94 | --003-- | 00 | 05 | 26 | 94 | FACILITY NAME N/A | DOCKET NUMBER 05000 |

| | | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| OPERATING MODE (9) | N | | 20.402(b) | | 20.405(c) | X | 50.73(a)(2)(iv) | | 73.71(b) |
| POWER LEVEL (10) | 000 | | 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 |
| | | | 20.405(a)(1)(iii) | | 50.73(a)(2)(i)B | | 50.73(a)(2)(viii)(A) | | (Specify in Abstract below and in Text, NRC Form 366A) |
| | | | 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 (Include Area Code) |
|---|---|
| Robert A. Gay - Senior Compliance Engineer | (508) 830-8047 |

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 |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | |
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| 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 15 single-spaced typewritten lines) (16)

On April 27, 1994, at 1410 hours, a false low reactor vessel water level signal occurred while shut down. The trip signal resulted in designed automatic responses that included a full scram signal, closing of applicable Primary Containment System isolation valves that were open, closing of the Secondary Containment System ventilation dampers and start of the Standby Gas Treatment System. The affected systems were returned to normal service by 1427 hours on April 27, 1994. The event occurred while venting a portion of the Reference Leg Backfill System as part of returning the Control Rod Drive System to service following scram solenoid pilot valve diaphragm replacements.

The cause of the false low water level signal was a deficiency in a procedure being used for venting and filling the Control Rod Drive System charging water header. Corrective action taken included filling, venting and repressurizing the CRD System charging water header. Corrective action planned includes revisions to the related vent and fill procedures.

The event occurred while shut down with the reactor mode switch in the REFUEL position for pre-startup testing. The Reactor Vessel (RV) pressure was zero psig with the RV water temperature at approximately 135 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv). The event posed no threat to the public health and safety.

BACKGROUND

The Reactor Vessel (RV) instrumentation associated with this event includes RV water level, and RV internal pressure. By design, this safety-related instrumentation is arranged into separate, redundant Channels 'A' and 'B'. Each instrument channel is connected to piping extending into primary containment. The Analog Trip System (ATS) is part of the instrumentation monitoring RV parameters. The ATS consists of transmitters, master trip units, slave trip units, and trip relays. The transmitters are housed on instrument racks located outside primary containment. The transmitters convert the parameter being monitored into signals. The signals are converted into indications and/or trip functions to related systems. The systems include the Reactor Protection System (RPS), Anticipated Transient Without Scram (ATWS) System, Primary Containment Isolation Control System (PCIS)/Reactor Building Isolation Control System (RBIS), Core Standby Cooling Systems (CSCS), and Reactor Core Isolation Cooling (RCIC) System.

The RPS, PCIS/RBIS, ATWS, RCIC and CSCS water level signals are derived from instruments associated with Condensing Chambers 12A/12B. A low reactor vessel water level trip signal is generated by analog trip system trip units connected to level transmitters LT-263-57A, LT-263-57B, LT-263-58A and LT-263-58B. LT-263-57A/B are associated with Condensing Chamber 12A. LT-263-58A/B are associated with Condensing Chamber 12B. The Feedwater Control System water level signals are derived from instruments associated with Condensing Chambers 13A/13B. The installation of a continuous reference leg backfill system in July 1993, under Plant Design Change (PDC) 93-24, precludes the buildup of non-condensable gases in the reference lines associated with Condensing Chambers 12A/12B. The design is intended to prevent the build up of non-condensable gases inside the reference lines, thus eliminating the possibility of inaccurate level indications created by notching during depressurization. Non-condensable gases in the reference lines associated with Condensing Chambers 13A/13B can be removed by performance of the backfilling procedures 3.M.2-12.3, "Backfilling Condensed Chambers 12B and 13B, Active Leg and Instrument Lines from Racks 2206, 2276, 2252", and 3.M.2-12.4, "Backfilling Reference Lines for Rack 2205, 2275, and 2251 Instruments (12A & 13A Condensing Chambers)".

Prior to the event, Operations personnel were preparing the plant for startup. The plant had been shut down on April 22, 1994, for replacement of scram solenoid pilot valves' diaphragms. Maintenance personnel had completed the diaphragm replacements. All control rods were in the fully inserted position. The reactor mode selector switch was in the REFUEL position. Operations personnel had commenced systematic venting of the Control Rod Drive (CRD) System discharge piping at the reactor vessel water level reference leg backfill Rack C2208.

Rack C2208 provides, via separate headers, backfill flow to the Reactor Water Level instrumentation connected to Condensing Chambers 12A and 12B. The venting was being accomplished in accordance with Procedure 2.1.11.1 (Rev. 6), "System Fill Vent and Drain Instructions". The venting was necessary because the CRD System, normally in service, was removed from service and had been depressurized and vented.

# LICENSEE EVENT REPORT (LER)
## TEXT CONTINUATION





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

The Residual Heat Removal (RHR) System was in the Shutdown Cooling (SDC) mode of operation with the Loop 'B' pumps in service and with the RHR valves MO-1001-47, -50, -28B and -29B valves in the open position. The Reactor Water Cleanup (RWCU) System was in service.

EVENT DESCRIPTION

On April 27, 1994, at 1410 hours, a false low reactor vessel water level signal occurred while shut down. The signal resulted in a full Reactor Protection System (RPS) scram signal, isolation signals to applicable portions of the Primary Containment Isolation Control System (PCIS), and the Reactor Building Isolation Control System (RBIS). The RPS scram signal resulted in no control rod motion because the control rods were in the inserted position at the time of the event.

The PCIS isolation signal resulted in the following designed responses:

- The inboard and outboard Primary Containment System (PCS) Group 2 (two)/sampling system isolation valves, that were in the open position, closed automatically.
- The PCS Group 3 (three)/RHR System SDC suction line isolation valves MO-1001-50, -47, and the return isolation valve MO-1001-29B closed automatically. The RHR Loop 'B' pumps ('B' & 'D') tripped automatically.
- The PCS Group 6 (six)/Reactor Water Cleanup (RWCU) System isolation valves MO-1201-2, -5 and -80, in the open position, closed automatically. The RWCU pump in service tripped automatically.

The RBIS isolation signal resulted in the following responses:

- The Secondary Containment System (SCS) Trains 'A' and 'B' supply and exhaust ventilation dampers closed automatically.
- The Standby Gas Treatment System (SBTS) Trains 'A' and 'B' started automatically.

After initial investigation, licensed operator actions included the following activities. The RPS scram signal was reset. The PCIS and RBIS isolation signals were reset at 1425 hours on April 27, 1994. The RWCU system was returned to service at 1427 hours. The SGTS was returned to normal standby service. The RHR System was returned to service in the SDC mode at 1421 hours.

Problem Report 94.9200 was written to document the event. The NRC Operations Center was notified in accordance with 10 CFR 50.72 at 1510 hours, on April 27, 1994.

The event occurred while shut down. The reactor mode selector switch was in the REFUEL position for pre-startup testing. The RV pressure was zero psig with the RV water temperature at approximately 135 degrees Fahrenheit.

A critique was convened at 1515 hours on April 27, 1994. The critique was attended by applicable personnel including the operators who were performing the venting process.

## CAUSE

The root cause of the event was procedural deficiency. Procedure 2.2.87 (Rev. 48), "Control Rod Drive System", did not contain sufficient steps or instructions to alert the licensed operator that when the CRD system is to be shut down, the Reactor Water Level Reference Leg Backfill System shall be removed from service at reference leg backfill Rack C2208 and returned to service after the CRD system has been restored to normal (Procedure 2.2.80).

A false low reactor water level signal was produced when a vent path was established through rack C2208 flow element bypass valve HO-C2208-4, metering bypass valve HO-C2208-6, normally open inlet block valve HO-C2205A-1 and normally open valves HO-C2205A-4 and HO-C2205A-5. The Reactor Operator (RO) performing the venting process in accordance with Procedure 2.1.11.1 (Rev. 6) "System Fill, Vent And Drain Instructions", reported a surge of air and water was noticed when valve HO-C2208-6 was opened. This surge of air and water resulted in a momentary change in sensed level from +31" to +7" on the 'A' side level instrumentation. The venting process used in accordance with Procedure 2.1.11.1, contributed to this event by failing to consider the impact on equipment during venting and filling.

## CORRECTIVE ACTION

Corrective action taken included filling, venting and repressurizing the CRD charging water header in accordance with Procedure 2.2.87 (Rev. 48), "Control Rod Drive System". This action was taken to eliminate air in the CRD charging water header up to Rack C2208.

The reactor water level backfill system was returned to service in accordance with Procedure 2.2.80 (Rev. 12), "Reactor Vessel Level, Temperature and Internal Pressure Instrumentation". This action was taken to remove air from Rack C2208 to the reactor water level instrument racks.

The reactor level instruments were backfilled and returned to service in accordance with Procedure 3.M.2-12.4 (Rev. 2), "Backfilling Reference Lines for Rack 2205, 2275 and 2251 Instruments (12A and 13A Condensing Chambers)", and Procedure 3.M.2-12.3 (Rev. 5), "Backfilling Condensing Chambers 12B and 13B, Active Leg and Instrument Lines from Racks 2206, 2276, 2252.

A walkdown was performed on the instrument racks for possible external leakage. No indication of external leakage was identified during the walkdown.

Corrective action planned to preclude recurrence includes revising Procedure 2.2.87 (currently, Rev. 48), "Control Rod Drive System". The purpose of the revision is to add steps in the procedure section that will identify actions to be taken when the CRD System is to be removed from or returned to service. Essentially, the Reactor Water Level Reference Leg Backfill System will be removed from service at Rack C2208 and returned to service after the CRD System has been restored to normal (Procedure 2.2.80).

Procedure 2.1.11.1 (currently, Rev. 6), "System Fill, Vent, and Drain Instructions", will be revised to include a caution statement in Attachment 2, "System Fill Checklist", and Attachment 3, "System Vent Checklist". The enhancement is to ensure a vent path, if established, does not expose instrumentation to air that could subsequently affect level instrumentation performance.

SAFETY CONSEQUENCES

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

The event was the designed response to a false reactor vessel low water level condition sensed by level transmitters LT-263-57A/B. The setpoint for a reactor vessel low water level trip signal is calibrated at approximately +12 inches. In this event, reactor vessel water level did not decrease. The Technical Specification 2.1.3 safety limit for reactor vessel water level, while in cold shutdown, is 12 inches above the top of the active fuel zone. The level corresponding to the top of the active fuel zone is negative 126 inches. The false low reactor vessel level signal that occurred was approximately +7 inches. A sensed level of +7 inches is approximately 53 inches above the CSCS trip setting (calibrated at approximately -46 inches).

The RHRS/SDC mode of operation has a power generation design basis only. The SDC mode of operation functions to reduce the RV water temperature to 125 degrees Fahrenheit approximately 20 hours after a shutdown for refueling or servicing activities.

After the false low reactor vessel water level signal occurred, shutdown cooling (SDC) was re-established following a slight RV temperature increase of approximately 2 degrees Fahrenheit. However, alternate means for heat removal are available and described in Procedure 2.4.25, "Loss of Shutdown Cooling", including methods for feed and letdown using the Condensate System, RWCU System, and the Main Condenser.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the PCIS Group 2 and 3 isolation signals were not planned and are not exempted from reporting in accordance with 10 CFR 50.73(a)(2)(iv).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) written since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73 (a)(2)(iv) involving a similar event or cause, since installation of the continuous reference leg backfill system in July 1993. The review identified a similar event reported in LER 50-293/93-026-00.

For LER 93-026-00, a low reactor water level signal (approximately +12 inches) occurred during hot shut down conditions. The reactor mode selector switch was in the REFUEL position for startup checks that were in progress. The RV water Temperature was 253°F and the RV pressure was approximately 10 psig. The signal resulted in a full scram signal, and PCIS and RBIS isolations similar to this event. The cause of the event was utility licensed operator error in that the operator did not refer to all reactor vessel water level indications while maintaining RV water level in the desired band. A contributing factor was the feedwater level instruments did not accurately reflect reactor water level. The nonsafety related feedwater level instruments associated with Condensing Chambers 13A/13B were indicating higher than the safety-related reactor water level instruments associated with Condensing Chambers 12A/12B. The feedwater level indications were inaccurate due to the presence of non-condensable gases in the feedwater level instrumentation lines that are connected to Condensing Chambers 13A/13B. The non-condensable gases were present because there was/is no continuous backfill system for the nonsafety-related instrumentation connected to Condensing Chambers 13A/13B, and there was no procedural requirement to backfill the instrumentation lines connected to Condensing Chambers 13A/13B prior to plant startup. Corrective action taken included backfilling the instrumentation lines connected to Condensing Chambers 12A/13A and 12B/13B in accordance with procedure 3.M.2-12.3 and 3.M.2-12.4, and revising procedure 2.1.1, "Startup from Shut Down", to require backfilling the instrument lines associated with Condensing Chambers 13A/13B.

Evaluations for the installation of a continuous backfill system to the instrument lines associated with Condensing Chamber 13A/13B were previously identified and had not been completed when this report was prepared. This action is being tracked via action item RC94.0009.03.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

| COMPONENTS | CODES |
|---|---|
| Valve, Isolation (MO-1001-28B, -29B, -47 and -50) | ISV |
| Valve, Control, Hand (HO-C2208-4, -6, -13 and HO-C2205A-1) | HCV |

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

| SYSTEMS | CODES |
|---|---|
| Containment Isolation Control System (PCIS) | JM |
| Engineered Safety Features Actuation System (PCIS/RBIS/RPS) | JE |
| Plant Protection System (RPS) | JC |
| Reactor Building (SCS) | NG |
| Reactor Building Environmental Control System (RBIS) | VA |
| Reactor Water Cleanup (RWCU) System | CE |
| Standby Gas Treatment System (SGTS) | BH |