10 CFR 50.73

BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

January 27, 1995
BECo Ltr. 95-007

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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) 94-007-00, "Primary Containment Pressure Boundary Degraded due to Uninstalled Instrument Plug", 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

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DWE/lam/9400700

Enclosure: LER 94-007-00

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NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) (See reverse for number of digits/characters for each block)		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.

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

Primary Containment Pressure Boundary Degraded due to Uninstalled Instrument Plug

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
12	28	94	94	007	00	01	27	95	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)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10): 100		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)	X	50.73(a)(2)(vii)(C)		OTHER
		20.405(a)(1)(iii)	X	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)
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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
A	BD	PTD	F180	N						

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 December 28, 1994, at approximately 1040 hours, a plug for the Drywell-to-Torus atmosphere differential pressure transmitter that is part of the Torus atmosphere portion of the Primary Containment System pressure boundary was discovered to be not installed during a calibration of the transmitter while at 100 percent power. Immediate action taken included the installation of a plug for the transmitter and the senior utility shift Licensed Operator was notified of the discovery. A 24 hour limiting condition for operation was entered and later terminated at 1646 hours on December 28, 1994. Preliminary engineering evaluation concluded potential PCS leakage due to the uninstalled plug at the PCS integrated leak rate test pressure of 45 psi could have exceeded 1 (one) percent. The NRC Operations Center was notified of the condition at 1756 hours on December 28, 1994.

The root cause was utility non-licensed I&C Technician and Supervisor error during the previous calibration of the transmitter on November 22, 1994. Contributing factors included deficiencies in the calibration procedure, physical location of the plug, and transmitter design and installation (lack of calibration valves). Corrective action taken included Maintenance Manager direction to Maintenance personnel regarding procedure signoffs and verification, Senior Management briefings to Maintenance personnel regarding the uninstalled plug and management expectations, strengthening the calibration procedure and tagging of all instruments that are part of the Primary Containment pressure boundary. Corrective action planned includes the installation of isolation valves to selected instruments.

Evaluation of the potential radiological consequences of the uninstalled plug concluded 10 CFR Part 100 and 10 CFR 50 Appendix A (GDC 19) exposure guidelines would not have been exceeded if a design basis accident had occurred while the plug was not installed.

BACKGROUND

The safety objective of the Primary containment System (PCS) is to provide the capability in conjunction with other safeguard features, to limit the releases of fission products in the event of a postulated design basis accident so that offsite doses would not exceed the guideline values set forth in 10 CFR Part 100. The PCS design employs a low leakage pressure suppression containment system which houses the Reactor Vessel, the Reactor Recirculation System loops, and other branch connections of the Reactor Primary System.

The PCS consists of a Drywell, a pressure suppression chamber (Torus) that stores a large volume of water, a connecting vent system between the Drywell and the Torus, isolation valves, vacuum relief system, containment cooling systems, and other service equipment.

In the event of a process piping failure within the Drywell, reactor water and steam would be released into the Drywell atmosphere. The resulting increased Drywell pressure would force a mixture of gas, steam, and water through the vent system into the Suppression Pool. The steam would condense rapidly in the Suppression Pool and result in rapid Drywell pressure reduction. Non-condensible gas transferred during blowdown would pressurize the Torus atmosphere. The resulting Torus atmosphere pressure would subsequently vent to the Drywell through the vacuum relief system as the Drywell pressure decreases to less than the Torus atmosphere pressure.

Service equipment, including instrumentation, is provided to maintain primary containment within its design parameters during normal operation. The instrumentation includes Drywell atmosphere temperature and pressure, Suppression Pool water temperature and level, Drywell-to-Torus atmosphere differential pressure, and Reactor Building-to-Torus atmosphere differential pressure.

The instrumentation that monitors the Drywell atmosphere pressure includes PTD/PID-5067A. The instrumentation that monitors the Torus atmosphere pressure includes PTD/PID-5067B. The instrumentation that monitors the Drywell-to-Torus atmosphere differential pressure includes PTD/PID-5021. Transmitters PTD-5021, PTD-5067A and PTD-5067B are located on Panel C-88 and were manufactured by the Foxboro Company (Model E13DM). The E13 series transmitters operate on the force-balance principle. In operation, a difference in pressure between the high pressure side and low pressure side of the transmitter is sensed and converted to a signal sent to a pressure indicating instrument (e.g., PID-5021). The sensing cell is mounted in the transmitter's manifold that is equipped with high and low pressure connections, vent and drain plugs. For PTD-5021, the Drywell atmosphere is connected to the high pressure side of the transmitter, and the Torus atmosphere is connected to the low pressure side of the transmitter. For PTD-5067A, the Drywell atmosphere is connected to the high pressure side of the transmitter, and the Reactor Building atmosphere is introduced into the low pressure side of the transmitter. For PTD-5067B, the Torus atmosphere is connected to the high pressure side of the transmitter, and the Reactor Building atmosphere is introduced into the low pressure side of the transmitter. Located at the end of this report is a simplified sketch depicting the Drywell, Torus, and transmitters PTD-5067A, PTD-5067, and PTD-5021.

Instruments PID-5067A (from PTD-5067A), PID-5067B (from PTD-5067B), and PID-5021 (from PTD-5021) provide indication in the Main Control Room, provide no automatic control function, and are governed by Technical Specification 3/4.2.F (surveillance instrumentation). Table 3.2.F specifies the minimum number of operable instrument channels for these instruments and actions and timeframes for less than the minimum number of operable instrument channels. Table 4.2.F specifies these instrument channels be checked once per shift and calibrated once per 6 (six) months. Moreover, instruments PTD/PID-5067A, PTD/PID-5067B, and PTD/PID-5021 are used to ensure the Drywell-Torus atmosphere differential pressure is maintained as specified by Technical Specification 3.7.A.1.i ($\geq$ 1.17 psid). The instrument checks are performed via procedure 2.1.15, "Daily Surveillance Log". The instrument calibrations are performed via procedure 8.M.2-6.2, "Drywell and Torus Pressure/Temperature Readout".

Transmitters PTD-5067A, PTD-5067B and PTD-5021 were calibrated while shut down on November 22, 1994. Procedure 8.M.2-6.2 (Rev. 21) Attachment 2 was used to perform the calibrations. A plant startup was initiated on November 29, 1994 and the reactor core was brought to criticality at 0657 hours on that date and at which time primary containment integrity was required as specified by Technical Specification 3.7.A.2.a.

EVENT DESCRIPTION

On December 28, 1994, at approximately 1040 hours, a plug on the Torus atmosphere portion of the Drywell-to-Torus atmosphere differential pressure transmitter DPT-5021 was discovered to be not installed.

The discovery was made by utility non-licensed Instrumentation & Control (I&C) Technicians during the performance of Procedure 8.M.2-6.2 (Rev. 21) Attachment 2, "Drywell and Torus Pressure Readout Field Transmitters", at step [4]. This step is for the calibration of PTD-5021 and contains steps for isolating the instrument, connecting test equipment, calibrating the instrument, removing the test equipment, and returning the instrument to service. The discovery was made at substep (a)(4).

The shift utility non-licensed I&C Supervisor was notified and a temporary, non-qualified, plug was installed. The I&C Supervisor then notified the shift senior utility Licensed Operator (NWE) regarding the discovery, initiated a work document (priority 1 MR 19403991) for replacement of the temporary plug, and wrote a Corrective Action Program document (PR94.9568) to document the discovery.

The NWE initiated a 24 hour Limiting Condition for Operation (A94-236) at 1130 hours pending the replacement of the temporary plug. The NWE also requested Nuclear Engineering Services Department (NESD) evaluation of the discovery relative to potential PCS leakage. Preliminary calculations indicated that potential leakage would have been greater than the Technical Specification 4.7.A.2.a (10 CFR 50 Appendix J) test limit of 1.0% per day at 45 psi (test pressure). Based on the results of the preliminary calculations, the discovery of the uninstalled plug was determined to be reportable per 10 CFR 50.72(b)(2)(iii)(C) and the NRC Operations Center was notified in accordance with 10 CFR 50.72 at 1756 hours on December 28, 1994. Meanwhile, Maintenance personnel and Systems Engineering personnel

identified and walked down other similar instrumentation for a similar problem. The walk-downs were completed by 0300 on December 29, 1994, with no deficiencies identified.

A critique was held on December 28, 1994, and was attended by applicable personnel including the I&C Technicians and Supervisor on-shift at the time of the discovery of the uninstalled plug. A follow-up critique was held on January 5, 1995, and was attended by the I&C Technicians and Supervisor involved with the previous performance of Procedure 8.M.2-6.2 on November 22, 1994.

The discovery occurred while at 100 percent reactor power with the reactor mode selector switch in the RUN position. The Reactor Vessel (RV) pressure was 1037 psig with the RV water temperature at approximately 547°F.

CAUSE

The root cause was utility non-licensed I&C Technician and Supervisor error. Specifically, inattention to detail by the I&C Technicians who performed and the Supervisor who reviewed the calibration of PTD-5021 on November 22, 1994. The I&C Technicians and Supervisor indicated there was no sense of urgency or distractions that influenced their work. To his best recollection, the Technician who initialed the procedure substep recalled the re-installation of the plug. To his best recollection, the other technician who did not initial the double verification block for the substep recalled the re-installation of the plug. The re-installation of the plug could not be proved or disproved during the investigation. The I&C Supervisor focus during review of the calibration procedure was on the numerical values recorded during the calibration. The missing initials in the double verification block were not noticed by the Supervisor during the review.

There were several factors contributing to the uninstalled plug. The factors include procedure deficiencies, and less than optimum instrument design and installation.

- Procedure 8.M.2-6.2 deficiencies:

  - Specific information cautioning technicians the plug is a primary containment boundary was not included in the procedure.

  - Attachment 2 step [4](e) includes five substeps for returning the transmitter to service. Substep (2) directs the technician to: "Install test plugs on transmitter". This requires two actions that may have resulted in the re-installation of only one plug and not two.

  - The space for the initials of the Technician who is to verify substep (2) is located at the bottom of procedure page and immediately above the page number and revision number, where it could be overlooked.

- Procedure 8.M.1-33, "Instrument Walkdown", deficiency. The procedure is performed to verify instruments are properly aligned prior to startup from extended outages. Transmitters PTD-5021, PTD-5067A and PTD-5067B were not included in the procedure which was performed prior to startup (i.e., after November 22, 1994).

- Instrument design and installation:

  - Transmitter PTD-5021 is located on Panel C-88 approximately 18"-24" above floor level. The low pressure side of PTD-5021 has two test plugs one of which is removed from the low pressure side of the transmitter for a calibration. The plug that is removed for a calibration is located on the bottom surface of the transmitter's manifold and, as such, is not readily visible without crouching or significant body adjustment.

  - Transmitters PTD-5067A and PTD-5067B that are also calibrated as part of Procedure 8.M.2-6.2 Attachment 2 are also located on Panel C-88. Transmitters PTD-5067A and PTD-5067B are located side-by-side and approximately five feet above floor level. Since these transmitters measure Drywell atmosphere pressure and Torus atmosphere pressure relative to Reactor Building atmosphere pressure, one of the low pressure side plugs is not installed during operation. This configuration is different from that of PTD-5021 which requires both low pressure side (Torus atmosphere) plugs to be installed during operation.

  - Transmitter PTD-5021 does not have calibration valves consistent with most other transmitters and pressure switches installed in other applications.

CORRECTIVE ACTION

Immediate corrective action taken consisted of the installation of a temporary, non-qualified, plug where the plug was not installed on PTD-5021. The temporary plug installed on PTD-5021 was replaced with a qualified plug on December 28, 1994 (via MR19403991) with PTD-5021 isolated for the replacement and returned to service after the replacement. The LCO (A94-236) was terminated at 1646 hours on December 28, 1994, following the installation of the qualified plug.

Maintenance personnel (Technicians, Supervisors and Division Managers), including I&C Technicians and Supervisors, received direction from the Maintenance Section Manager on January 4, 1995, regarding the requirements for procedure signoffs and verifications.

In response to a Plant Manager request, the Quality Assurance Department (QAD) began an administrative review of 30 surveillance procedures performed by I&C personnel in 1994. Collectively, the review identified seven discrepancies. The discrepancies included three missing check marks, three missing initials, and one missed double verification. The review concluded the discrepancies were a failure to adhere to approved procedures. Procedure 1.5.17, "Conduct of Maintenance", Section 6.9 (Adherence to Procedures) states, in part, "approved written procedures and instructions shall be strictly adhered to". The review resulted in the issuance of a Deficiency Report (DR 2064) on January 9, 1995. The discrepancies were subsequently evaluated and determined to have had no impact to plant

safety. The Deficiency Report (DR 2064), in turn, led to plant staff review of 34 additional completed surveillance procedures. Collectively, the staff reviews identified 23 administrative discrepancies out of approximately 7500 procedurally provided spaces that require annotation during performance of these 34 procedures. None of the discrepancies prevented acceptance criteria signoff. The corrective actions currently being taken and planned due to the QAD and plant staff reviews include the following:

- Procedure 1.5.17 (Rev. 1), "Conduct of Maintenance", is being revised. The focus of the revision is to clearly define double verification and independent verification.

- Review of completed surveillances procedures by two supervisors. The focus of these second reviews is to provide additional assurance the completed surveillance procedure contains the applicable data entries, check marks, initials, and signatures prior to licensed operator signoff. Based on experience, the second supervisor reviews may be modified or eliminated as performance indicates.

- A monthly audit will be performed of 21 completed surveillance procedures, seven each from the I&C Division, Mechanical Division and Electrical Division. The audits will be performed by senior Nuclear Organization managers. Based on standards being finalized when this report was prepared, these audits may be modified or eliminated as performance indicates.

- The formal response to DR 2064 had not been finalized when this report was prepared. This report will be supplemented if the response identifies additional significant corrective action.

On January 13, 1995, the Station Director, Plant Manager, and Maintenance Manager briefed Maintenance Section personnel on the significance of the uninstalled plug for PTD-5021 and expectations for adhering to procedures.

A modification document (FRN 95-04-05) was approved on January 20, 1995. The modification includes provision for installing additional tubing, fittings, and valves to the drain/calibration ports of transmitters PTD-5067A, PTD-5067B, and PTD-5021. The changes have been scheduled for implementation beginning in the first week of February 1995.

Primary Containment pressure boundary related instruments that, if breached, would not be detected by normal means of monitoring have been tagged.

Procedure 8.M.2-6.2 was revised (to Rev. 23). The focus of the revision was to strengthen the step for returning PTD-5021 to service relative to the re-installation of the plug. The revision included similar strengthening relative to returning PTD-5067A and PTD-5067B to service.

Corrective action planned to preclude recurrence includes the following:

- Procedure 8.M.1-33 (currently Rev. 7) is being revised. The focus of the revision is to include PTD-5021, PTD-5067A and PTD-5067B.

- PTD-5021 and similar instruments have been evaluated for possible modification. The focus of the evaluation was the installation of isolation valves between the plugs and instruments with a primary containment pressure boundary that are required to be opened and/or closed for calibration purposes. Isolation valves will be similarly installed on other instruments that are not yet so equipped.

- The frequency of I&C Division self-assessment type audits of completed I&C procedures has been increased. The results of these audits will be reviewed at the weekly meetings of the I&C Technicians and Supervisors. The frequency may be modified as performance indicates.

DISCIPLINARY ACTION

The I&C Technicians who performed the calibration and Supervisor who reviewed the calibration of PTD-5021 on November 22, 1994, received disciplinary action.

SAFETY CONSEQUENCES

The most severe nuclear system effects and the greatest release of radioactive material to primary containment results from a complete circumferential break of one of the recirculation loop pipelines. The accident is described in the Final Safety Analysis Report (FSAR) section 14.5.3 and was established as the design basis LOCA. The radiological consequences of a design basis LOCA are assessed in the FSAR (section 14.5.3.2 and Appendix R section R.6) and are part of the bases for Technical Specification 4.7.A for primary containment testing (ILRT). As evaluated in the FSAR, the radiological consequences of a design basis LOCA with a 1.50% per day leak rate, no holdup, fission product release fractions stated in the AEC/NRC Technical Information Document TID-14844, 95 percent SGTS filter efficiency, and Main Stack release would be significantly less than the limits of 10 CFR 100.

Site evaluation factors, discussed in 10 CFR Part 100, include the safety features and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Pilgrim Station is designed to keep the thermal response of the core below fuel clad melt conditions.

The likelihood of a design basis LOCA occurring during the period of time the plug of PTD-5021 was not installed and during a six-month period (Technical Specification required surveillance frequency for PTD/PID-5021) was evaluated. The evaluation used probabilistic risk input from the Pilgrim Station specific Individual Plant Examination (IPE). The evaluation concluded the risk of reactor core damage during a 37 day period (i.e., a period the plug was not installed) was approximately 1.7E-08. The risk of core damage during a six-month period would be approximately 5.9E-08. Therefore, the likelihood of a design basis LOCA occurring during the period the plug was not installed and during a six month period (Technical Specification surveillance frequency) was very low.

The effect of the uninstalled plug of PTD-5021 was evaluated for potential PCS leakage and potential radiological consequences. The calculation included potential leakage at 45 psig and at the Torus atmosphere pressures postulated during and following a design basis LOCA. The calculation also considered the results of the most recent as-left PCS Integrated Leak Rate Test (ILRT). The calculation indicated the potential PCS leakage due solely to the uninstalled plug could have been approximately 1.82% per day at 45 psig. The total potential PCS leakage at 45 psig, i.e. the leakage at 45 psig due to the uninstalled plug and the most recent as-left ILRT results (at 45 psig) could have been approximately 2.1% per day. Although the design basis pressure of 45 psi is used for ILRT, Torus atmosphere pressure during a design basis LOCA would peak less rapidly (than the Drywell) at 27 psig and decrease thereafter. Therefore, the calculation also considered potential leakage at predicted Torus atmosphere pressure during and following a design basis LOCA. The calculation indicated the potential PCS leakage due solely to the uninstalled plug could have been approximately 0.53% per day using the Torus atmosphere pressure profile during and following a design basis LOCA. The potential total PCS leakage, i.e. the potential leakage due to the uninstalled plug using the Torus atmosphere pressure profile during and following a design basis LOCA and the most recent as-left ILRT results (at 45 psig), could have been approximately 0.88% per day. The most recent as-found ILRT results (at 45 psig) were also considered to account for potential PCS degradation since the most recent as-left ILRT. This consideration resulted in a potential total PCS leakage, i.e. the potential leakage due to the uninstalled plug using the Torus atmosphere pressure profile during and following a design basis LOCA and the most recent as-found ILRT results (at 45 psig), that could have been approximately 0.89% per day.

The potential radiological consequences of PCS leakage was evaluated regarding the radiation exposure guidelines contained in 10 CFR Part 100 and 10 CFR Part 50 Appendix A (GDC 19). The evaluation included consideration of the maximum PCS leakage allowable to not exceed 10 CFR Part 100 limits using the TID-14844 source term is approximately 6.97% per day. The evaluation also considered the PCS calculated leakage and included use of the TID-14844 source term. The evaluation concluded the radiation exposure guidelines of 10 CFR Part 100 and 10 CFR Part 50 Appendix A (GDC 19) would not have been exceeded using the total PCS leakage, i.e. the ILRT results at 45 psig and potential leakage due to the uninstalled plug using the Torus atmosphere pressure profile during and following a design basis LOCA, and TID-14844 source term. Moreover, if the FSAR assumed PCS leakage of 1.5% per day is substituted as the total PCS leakage, the exposure guidelines of 10 CFR Part 100 and 10 CFR Part 50 Appendix A (GDC 19) would not have been exceeded.

The Drywell-to-Torus atmosphere differential pressure during the period of time the plug was not installed was evaluated. The evaluation utilized information including available plant computer (EPIC) historical data records; daily readings from PTD-5067A, PTD-5067B and PTD-5021 obtained via Procedure 2.1.35, "Control Room Readings"; and Drywell nitrogen makeup data. The evaluation concluded there was reasonable assurance the Drywell-to-Torus atmosphere differential pressure was maintained at greater than the 1.17 psid specified by Technical Specification 3.7.A.1.i.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) because primary containment integrity, specified by Technical Specification 3.7.A.2.a, was not maintained at all times from November 29, 1994, at 0657 hours (when the reactor was brought to criticality) until a plug was installed on December 28, 1994.

This report is also submitted in accordance with 10 CFR 50.73(a)(2)(vii)(C) because the PCS (i.e., pressure boundary integrity) was inoperable due to the uninstalled plug for PTD-5021.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs). The review focused on LERs involving the primary containment pressure boundary. The review identified LERs 89-008-00, 89-037-01 and 91-023-00.

For LER 89-008-00, the Drywell personnel airlock portion of primary containment was inadvertently breached for approximately 5 (five) seconds when both the inner and outer doors of the airlock were simultaneously open. Secondary Containment was operable at the time of the event. The event occurred at 0230 hours on February 16, 1989, with the reactor mode selector switch in the STARTUP position, the reactor power level at approximately 2.5 percent and the RV pressure at approximately 740 psig with the RV water temperature at approximately 505°F. The event occurred when personnel were attempting to enter the Drywell airlock through the outer door from the Reactor Building while others were attempting to enter the Drywell airlock through the inner door from the Drywell. The cause was slack in the tie rod roller chain assembly of the interlock mechanism on the inner door. The most probable cause of the slack in the chain assembly was improper tightening of the roller chain assembly jamb nuts during previous maintenance. The loose jamb nuts and increased slack in the roller chain assembly allowed the mechanical interlock for the doors to be satisfied without the inner door being closed. Immediate actions taken included the closing of the Drywell air lock doors, the termination of all Drywell entries until controls were established to prevent recurrence. Corrective action taken included adjustment of the tie rod roller chain assembly and including the airlock doors in the Preventive Maintenance program.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

For LER 89-037-01, one of the Traversing In-Core Probe (TIP) solenoid operated ball valves (45-300A) that is part of the PCS was discovered to be almost in the full open position when it was thought to be in the closed position. Secondary Containment and the (explosive type) shear valve, in-series with the ball valve, were operable during the period of time the valve was open. The discovery occurred at approximately 1200 hours on November 30, 1989, with the reactor mode selector switch in the RUN position, the reactor power level at 94 percent, and the RV pressure at 1025 psig with the RV water temperature at approximately 540°F. The discovery occurred when the ball valve was being manually operated as part of troubleshooting the ball valve. A significant rotation (approximately 80 degrees) of the ball valve's roll pin from the expected and indicated full closed position occurred when the ball valve was manually operated. [Leading to the discovery was a previous problem on November 15, 1989. On that date, the TIP System mechanism 'A' (C-730A) related to ball valve 45-300A became stuck with its TIP in the TIP indexer while being withdrawn during a surveillance performed per Procedure 9.5, "LPRM Calibration".] The cause of the ball valve being open was damage to the ball valve stem. The damage was believed to have occurred during manual manipulation of the valve on November 15, 1989, when the valve was manually opened to allow retraction of the TIP. This action was taken because the valve's actuator solenoid de-energized while the TIP was being retracted. After the TIP was retracted, the ball valve was allowed to self-close (spring action) to what was believed to be the closed position. A Temporary Procedure (TP89-112) was written to troubleshoot the problem. The troubleshooting, however, on November 30, 1989, revealed permanent distortion that resulted in an offset condition of the ball valve position such that the valve was (unknowingly) not fully closed by the actuator spring. Initial action taken included closing the ball valve, verifying the valve was closed, removing the valve actuator and tagging the valve. Corrective action taken included replacement of the ball valve and actuator while shut down on December 11, 1989, and I&C Technician training.

For LER 91-023-00, 11 of 76 Drywell head bolts were discovered to be loose at approximately 2300 hours on July 28, 1991. The discovery occurred while shut down and during a required 10CFR50 Appendix J integrated leak rate test. The test was being performed at full pressure (45 psig), near the end of a refueling outage. The cause of the loose Drywell head bolts was the failure of some of the spherical type washer sets that are part of Drywell head flange connection. The primary cause of the failed washers was the use of washers made of case hardened AISI 8620 material instead of hardened and tempered AISI 4140 material. Factors possibly contributing to the failure of some of the washers were the inverted installation of some of the washers, corrosion, and crack propagation over time. Corrective action taken included the replacement of all Drywell head washers. The replacement washers were made of AISI 4140 material hardened and tempered to the specified hardness. Additional corrective actions taken included attempts to determine the source of the washers (installed during original construction c.1971) and revision of the procedure used to remove and install the Drywell head. The revision included specific detail regarding how the Drywell head washer sets are installed.

A review was also conducted of Pilgrim Station LERs submitted since 1984 involving utility non-licensed I&C personnel error. The review identified LER 92-018-00.

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

For LER 92-018-00, an automatic scram occurred on December 20, 1994, at 0233 hours during power ascension while at 75 percent reactor power. Automatic responses included a transfer of electrical loads, turbine-generator trip, and closing of applicable isolation valves. The root cause was utility non-licensed I&C Technician error. The Technician incorrectly adjusted the trip settings of the Main Steam Radiation Monitors during power ascension on December 19, 1992. Hydrogen injection began on December 20, 1992, and the resulting increase in main steam radiation exceeded the trip settings that were too low for hydrogen injection radiation levels. Contributing to the error was the format of the settings identified in the trip setting procedure (3.M.2-7.6). The required settings were identified in whole number format (e.g., 14,900) while the radiation monitors display the settings in scientific notation format (e.g., 1.49E4). Corrective action taken included revising the trip setting procedure to identify the trip settings in scientific notation format.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

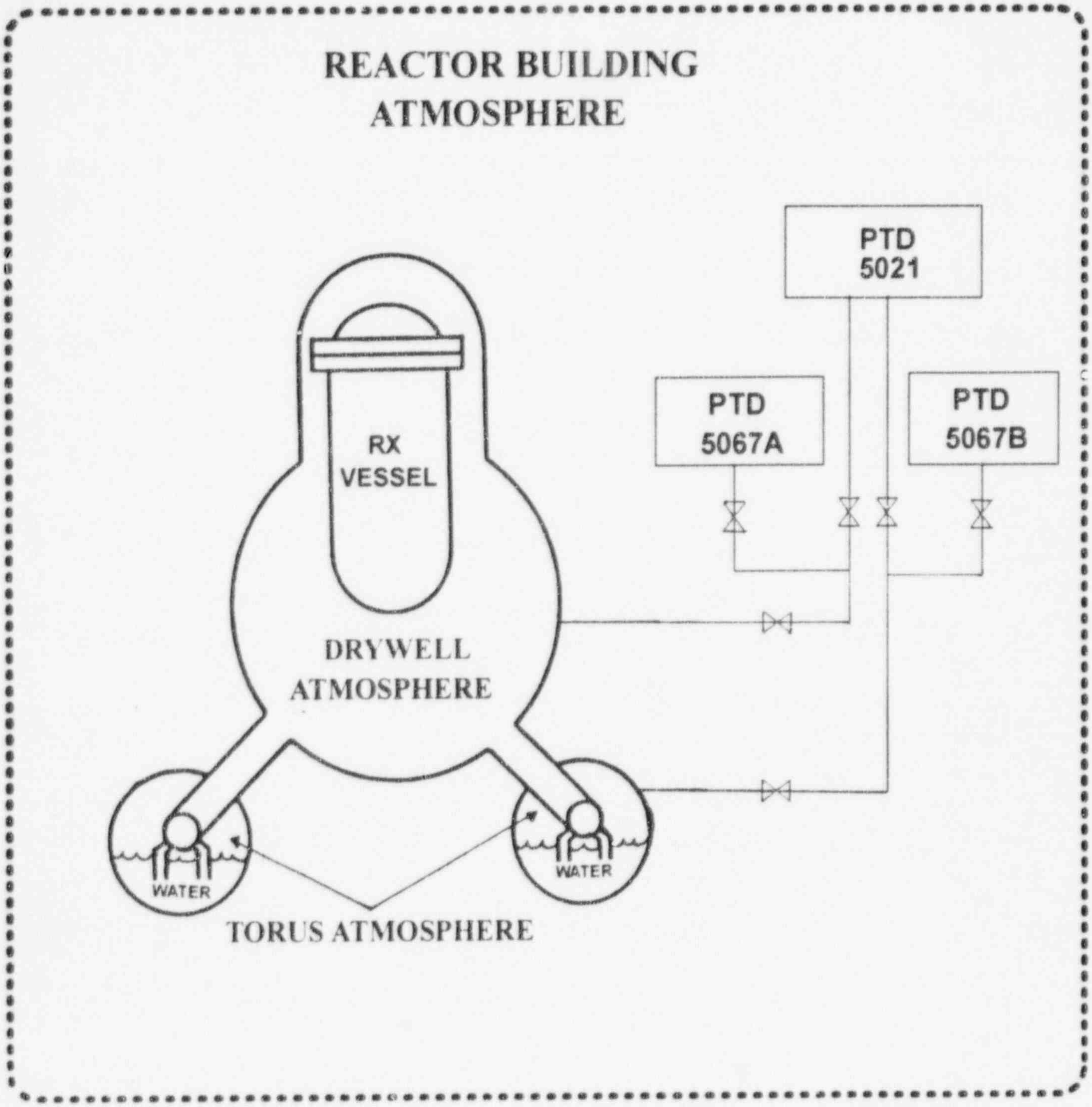
The EIIS codes for this report are as follows:

COMPONENTS	CODES
Transmitter, Differential, Pressure (PTD-5021)	PTD
Vessel (Primary Containment Vessel/Torus)	VSL
SYSTEMS	
Containment Leakage Control System	BD

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
PILGRIM NUCLEAR POWER STATION	05000-293	94	--007--	00	12 of 12

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



SIMPLIFIED SKETCH OF DRYWELL-TORUS ATMOSPHERE PRESSURE TRANSMITTERS