FEB 04 1993

Docket No. 50-293

Mr. Paul M. Blanch
135 Hyde Road
West Hartford, Connecticut 06117

Dear Mr. Blanch:

In response to your request for updated information on reactor vessel instrumentation issues at Pilgrim, enclosed is relevant information from our inspection conducted in December 1992. This is essentially the same information which was provided to you in our letter of January 7, 1993.

If we can be of further assistance, please contact Mr. Eugene Kelly of my staff at (215) 337-5183.

Sincerely,

Original Signed By:
James C. Linville

James C. Linville, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosure: NRC Inspection Report 50-293/92-28 (Section 8.3)

cc w/encl:
Ernest C. Hadley, Esquire

cc w/o encl:
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector

bcc w/o encl:
G. Kelly, DRP
J. Linville, DRP
J. Macdonald, SRI - Pilgrim

RI:DRP
Kelly/meo
2/2/93

RI:DRP
Linville
2/4/93

OFFICIAL RECORD COPY

A:PBLANCH.MEO

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

Docket No. 50-293

JAN 2 8 1993

Dr. E. Thomas Boulette
Acting Senior Vice President-Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

Dear Dr. Boulette:

SUBJECT: PILGRIM INSPECTION 92-28

This refers to the safety inspection conducted by Messrs. J. Macdonald, A. Cerne, and D. Kern of this office November 24 to December 31, 1992 at the Pilgrim Nuclear Power Station, Plymouth, Massachusetts. Areas relevant to the health and safety of the public examined during this inspection are described in the enclosed report. Our findings were based upon observations of performance and independent evaluations of safety systems and quality records. The preliminary results have been discussed with Mr. L. Schmeling and other members of your staff at the conclusion of the inspection period.

Good precautionary measures were taken during the storm on December 11-14, that included reduced reactor power operations. Your operating staff's response to the weather-induced automatic reactor trip was prompt, particularly the shift supervisory oversight of trip recovery activities and electrical distribution system status.

Based on the results of this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Notice). It appears that several opportunities existed (but were missed) as part of supervisory procedure review, incident critique, and post-trip report processes to identify the improperly established main steam line (MSL) high radiation alarm setpoint prior to reactor startup. Setting of the alarm was of minor safety significance in that the reactor protection system function is independent of the alarm.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room. The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96.511.

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JAN 28 1993

Your cooperation with us is appreciated.

Sincerely,

James Linville, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosures:
1. Notice of Violation
2. NRC Inspection Report No. 50-293/92-28

cc w/encls:
E. Kraft, Acting Vice President, Nuclear Operations and Station Director
L. Schmeling, Plant Manager
V. Oheim, Manager, Regulatory Affairs and Emergency Planning Department
D. Tarantino, Nuclear Information Manager
N. Desmond, Compliance Division Manager
R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
R. Adams, Department of Labor and Industries, Commonwealth of Massachusetts
The Honorable Edward M. Kennedy
The Honorable John F. Kerry
The Honorable Edward J. Markey
The Honorable Terese Murray
The Honorable Peter V. Forman
B. McIntyre, Chairman, Department of Public Utilities
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Plymouth Civil Defense Director
Paul W. Gromer, Massachusetts Secretary of Energy Resources
Sarah Woodhouse, Legislative Assistant
A. Nogee, MASSPIRG
Regional Administrator, FEMA
Office of the Commissioner, Massachusetts Department of Environmental Quality Engineering
Office of the Attorney General, Commonwealth of Massachusetts
T. Rapone, Massachusetts Executive Office of Public Safety
Chairman, Citizens Urging Responsible Energy
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
K. Abraham, PAO (2 copies)
NRC Resident Inspector
Commonwealth of Massachusetts, SLO Designee

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 92-28

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: November 24 - December 31, 1992

Inspectors: J. Macdonald, Senior Resident Inspector
A. Cerne, Resident Inspector
D. Kern, Resident Inspector

Approved by: Eugene M Kelly 1/28/93
E. Kelly, Chief, Reactor Projects Section 3A Date

Scope: Resident inspection addressed the areas of plant operations, radiological controls, maintenance and surveillance, emergency preparedness, security, safety assessment and quality verification, and engineering and technical support. Initiatives selected for inspection included: restoration from an electrical backfeed lineup; observation of an inplant emergency preparedness drill; control and testing of certain containment isolation valves; and, plant design changes associated with the reactor vessel head spray lines.

Inspections were performed on backshifts during November 30 and December 1-4, 7, 11, 13-18, and 21-31, 1992. "Deep" backshift inspections were performed on December 13 from 10:00 to 12:00 p.m. and December 14 from 00:01 to 05:45 a.m.

Findings: Inspection results are summarized in the Executive Summary.

Procedure 3.M.2-7.6, "NUMAC Log Radiation Monitor Setpoint Change Procedure" was not properly performed. Technicians established incorrect RPS protective setpoints and management reviews failed to identify the associated discrepancies (Violation 92-28-01, see Section 4.4).

The technical basis for the deactivation of a head spray line remains unresolved (Unresolved Item 92-28-02, see Section 8.2).

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EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/92-28

Plant Operations Operations Section preparation for and response to the effects of a northeaster storm were comprehensive. Decisions to maintain reduced reactor power at the 80% rod pattern line and to separate the safety-related buses from the distribution system demonstrated a strong safety perspective.

The immediate response by operators to two automatic reactor trips was appropriate. Communications, use of procedures, and supervisory oversight of control room operations were excellent during post-trip recovery activities and subsequent reactor startups. Also, the identification of loose or missing bolts on motor operated valve actuator limit switch covers during routine rounds indicated good questioning attitudes and attention to detail by plant operators.

Maintenance and Surveillance Actions taken to verify the presence of and trend the effect of steam leakage past safety relief valve (SRV) RV 203-3A were thorough. Although not required by Technical Specifications (TS), the decision to establish cold shutdown and replace the leaking SRV pilot valve following an unrelated plant shutdown demonstrated sound safety judgement. In addition, coordination between the materials & component engineering section, maintenance personnel, and system engineers to complete the repair during this unscheduled maintenance period was outstanding. Restoration from the backfeed electrical lineup following post trip corrective maintenance was performed. Maintenance and operations personnel demonstrated excellent procedural knowledge and communications.

An automatic reactor trip on December 20 was caused by procedural weaknesses and poor work practices by technicians changing the main steam line (MSL) high radiation protective setpoints. Also, the technicians failed to lower the MSL high radiation alarm setpoints following the reactor trip. As a result, the MSL high radiation alarm was not available to control room operators upon the subsequent plant restart. Failure to properly reestablish the MSL high radiation protective setpoints and associated failure of the management review process on two occasions indicates a need for greater management attention.

Emergency Preparedness The capability to draw, analyze, and provide real time post-accident sampling system data under simulated emergency conditions was successfully demonstrated in a December drill.

Safety Assessment and Quality Verification Implementation of Phase II of a planned three phase structural reorganization, to become effective January 1, 1993, was announced on December 16, 1992. Licensee event reports (LERs) were of good detail, accurate, and clearly identified root cause and corrective action, detailed and properly addressed the required reporting criteria.

(EXECUTIVE SUMMARY CONTINUED)

Engineering and Technical Support Deactivated head spray line containment isolation valves remain to be removed from the Type C local leak rate test program. Several questions regarding American Society of Mechanical Engineers (ASME) Code criteria and the technical basis of certain aspects of the head spray line deactivation plant design change remain unresolved.

Continuing NRC review of the licensee reactor vessel water level instrumentation spiking status determined the operability assessment was consistent with the guidance of NRC generic documentation for degraded or nonconforming conditions on operability.

The inspector also checked the compatibility of the Residual Heat Removal (RHR) system configuration, as-left after implementation of PDC 86-20 with the design intent of PDC 86-52B-196 and, in this regard, additionally reviewed drawings M100BC-282-1, M100-38-7, MIN 40-12, M241 and an earlier FRN 191 to PDC 86-52B. During a plant inspection tour, the inspector noted that the electrical supply breaker (B20B3) for the head spray valve (MO-1001-63) which had been removed during RFO 8 was still danger-tagged open. The inspector reviewed the PNPS tagout sheet T90-10-21 and determined that this open tag status was inconsistent with the handling of the electrical supply to the other head spray CIV (MO-1001-60), where the breaker had been left open, but the tagout cleared. The inspector discussed this inconsistency with cognizant plant personnel who initiated action to query the nuclear engineering department as to whether the entire tagout T90-10-21 could be closed and cleared.

In review of PDC 86-52B-196 design change criteria, the inspector identified a statement which implied that the abandoned piping in the reactor head spray system outside containment would remain vented. However, since PDC 86-20 installed a pipe cap on one side of this piping and PDC 86-52B-196 capped the piping at penetration X-17 on the other side, the reviewed licensee documentation provided no indication how such venting was implemented in that the valves identified in the PDC relating to this piping appeared to have been left in the closed position. Furthermore, since valve MO-1001-60, as identified in tagout T90-10-21 was chained closed, two vent paths, one on either side of the closed valve, would have to have been provided for the as-left piping to be consistent with the PDC design criteria.

Additionally, the inspector confirmed that the pipe cap for the penetration X-17 piping had been procured to ASME Code, Section III, Class 2 criteria as "impact-tested material." However, it appeared that an impact-tested weld procedure had not been used to install the pipe cap as would be required by the ASME Code, Section IX, unless certain conditions of exemption allowed by Section III of the code were satisfied. Given that the PNPS FSAR documents the containment drywell shell material to be fabricated of impact tested plate and forgings, the licensee issued problem report 93-9005 to resolve this question regarding the weld procedure qualification.

The inspector determined through the review of PDC 86-52B-196 and related supporting documentation that the current configuration of the reactor head spray piping was acceptable, in that the continued safe operation of PNPS had not been adversely affected by the design modification. However, as noted above, certain questions, regarding the existing pipe venting and the containment penetration weld qualification criteria, remain open. Pending the licensee presentation of evidence that the installed configuration is in compliance with the intended PDC design criteria, these issues remain unresolved **(92-28-02)**.

8.3 **Reactor Vessel Water Level Instrumentation Update**

NRC Inspection Report 50-293/92-23, Section 8.1, provided a detailed status of licensee activities in response to reactor vessel water level instrumentation spiking experienced during recent reactor shutdown evolutions. Specifically, the issue has been addressed in terms of the generic concern for level instrumentation inaccuracies during rapid depressurization events due to the evolution of noncondensible gases from the reference legs.

The NRC conducted review of BECo operability determination of the level instrumentation, with particular attention on the instrumentation associated with the two-thirds (2/3) core height containment spray interlock. The safety function of this instrumentation is to provide level signals and indication such that adequate core cooling can be achieved for certain classes of accidents. The NRC staff independently concluded that this safety function would be satisfied at Pilgrim based upon the following:

- If flow is diverted to containment spray after 2/3 core coverage is achieved, one core spray pump alone is adequate to maintain 2/3 level and core cooling. Thus, even the diversion of all available low pressure coolant injection (LPCI) would not preclude adequate core cooling.

- It is unlikely that significant diversion of flow would occur prior to reflooding the vessel to 2/3 core height, because:

  -- the interlock does not cause any automatic actuations; that is, satisfying the interlock does not automatically divert LPCI flow to containment spray.

  -- according to the Pilgrim Reload Analysis (SAFER/GESTR Report NEDC-31852), for design basis loss of coolant accidents, the core is reflooded to 2/3 core height within approximately 60 to 150 seconds; therefore, the operator would have to immediately divert LPCI for such erroneous action to occur prior to reflooding the vessel. Moreover, operators are directed by procedure to assure adequate core cooling prior to initiation of containment spray, and operators have been sensitized to potential errors in level indication. Station Emergency Operating Procedures (ie, EOP-03, "Primary Containment Control") which govern the decision to divert LPCI flow and spray the containment would require the presence of a high drywell pressure above 2.5 psig. Also, the EOPs direct that only "those RHR pumps not required to assure adequate core cooling by continuous operation in the LPCI mode" be used for containment spray diversion.

  -- over 20 linear feet of reference leg volume, including both horizontal and vertical sections, must be voided and not recovered at Pilgrim to cause a continuous 14 inch level error, and an error of this amplitude is already considered in the interlock setpoint.

  -- it is expected that the magnitude of error in the level indication following an actual depressurization event would be significantly less than that estimated by conservative assumptions used in the calculations performed by the General Electric Company and the BECo consultants.

  -- the potential for level errors has likely been lessened by actions taken by the licensee to reduce external reference leg leakage (ie, tighten fittings and packing at the instrument racks).

- If it was postulated that LPCI flow was prematurely diverted to containment spray, the safety function of the interlock would still be fulfilled, because:

-- at Pilgrim the diversion of LPCI flow to containment spray (both drywell and torus) represents only approximately 25% of the capacity of one LPCI/RHR pump.

-- Appendix K analysis from the current Pilgrim Reload Analysis for the most limiting case for which LPCI flow is credited (ie, battery failure case), indicate a 1694 degrees F peak clad temperature (PCT), which represents a 506 degrees F margin to the 2200 degrees F PCT limit.

-- NRC staff reviewed analysis for a similar BWR/3 plant in which 100% of the flow of one RHR pump was assumed to be diverted from the core from the onset of the accident. These analysis support the staff judgement that, using Appendix K analysis assumptions, diversion of 25% of the flow of one RHR pump would not result in exceeding 2200 degrees F PCT.

Based on the above, the NRC staff concluded that any manual actuation of containment spray which is based upon an erroneous level signal to this interlock is both highly unlikely and of low safety significance, and that the safety function of the level instrumentation system at Pilgrim would therefore be fulfilled. The NRC staff also concluded that the BECo operability determination was performed consistent with the guidance of NRC Generic Letter No. 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," dated November 7, 1991.

During reactor depressurization following the two automatic trips that occurred during this inspection period, no reactor vessel water level instrumentation "spiking" was observed. The reactor tripped on December 13, 1992, after 20 days of operation and again on December 20,1992, shortly after startup from the previous trip. The NRC will continue to monitor BECo's progress in resolving the problem of noncondensible gas accumulation in the level instrumentation system at Pilgrim.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

9.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on January 7, 1993 with licensee management, summarizing inspection activity and preliminary findings for this report period. No proprietary information was identified as being included in the report.

9.2 Other NRC Activities

During the weeks of November 30 - December 4, 1992 and December 14-18, 1992 a Probabilistic Risk Assessment team inspection was conducted. Inspection results will be documented in NRC Inspection Report 50-293/92-81.