November 6, 2001

Mr. Robert M. Bellamy
Site Vice President
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, Massachusetts 02360-5599

SUBJECT: PILGRIM NUCLEAR POWER STATION - INSPECTION REPORT 50-293/01-06

Dear Mr. Bellamy:

On September 29, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report documents the inspection findings which were discussed on October 9, 2001, with yourself and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified one issue of very low significance. This finding was determined to be a violation of NRC requirements. However, because of its very low safety significance and because the finding was entered into your corrective action program, the NRC is treating the issue as a Non-Cited Violation, in accordance with Section VI.A.I of the NRC's Enforcement Policy. If you contest this noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Pilgrim facility. With the exception of the above finding, the team concluded that problems were properly identified, evaluated and resolved within the problem identification and resolution programs.

Since September 11, 2001, Pilgrim Nuclear Power Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Entergy Nuclear Generation Company. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Curtis J. Cowgill, Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 50-293/01-06
Attachment: Supplemental Information

cc w/encl: M. Krupa, Director, Nuclear Safety & Licensing
J. Alexander, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
S. Brennion, Regulatory Affairs Department Manager
J. Fulton, Assistant General Counsel
R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
Office of the Commissioner, Massachusetts Department of Environmental Quality Engineering
Office of the Attorney General, Commonwealth of Massachusetts
Chairman, Citizens Urging Responsible Energy
S. McGrail, Director, Commonwealth of Massachusetts, SLO Designee
Electric Power Division
J. Perlov, Secretary at the Executive Office of Public Safety
R. Shadis, New England Coalition Staff

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	50-293/01-06
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	August 19, 2001 through September 29, 2001
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector J. H. Williams, Senior Operations Engineer
Approved By:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-06, on 08/19-09/29/2001; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station, Personnel Performance During Non-Routine Evolutions.

The inspection was conducted by resident inspectors and a senior operations engineer. The inspection identified one Green finding, which was a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP) . The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/oversight/index.html Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspector identified a Non-Cited Violation for inadequate corrective actions associated with the reactor vessel water level instrumentation reference leg back fill system.

  The finding was of very low safety significance because the frequency of a rapid depressurization event is very low, there are multiple reactor level instruments some of which were unaffected, operator training and use of symptom based emergency operating procedures. (Section 4OA3)

B. Licensee Identified Findings

No findings of significance were identified.

Report Details

SUMMARY OF PLANT STATUS

On August 19, 2001, the reactor was taken to 100 percent core power following the August 13, 2001, reactor trip. On September 14, 2001, the plant experienced a recirculation pump run back to 67 percent core flow resulting in a power reduction to 79 percent power. This unplanned power reduction resulted during corrective maintenance to the feedwater level control system. Power was returned to 100 percent later the same day. On September 17, 2001, the unit was brought to 50 percent power to perform rod time testing an a thermal backwash of the main condenser. The unit returned to 100 percent power on September 18, 2001.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1RO5 Fire protection

a. Inspection Scope

Three plant areas important to reactor safety were toured to observe conditions related to: (1) transient combustibles and ignition sources; (2) the material condition and readiness of fire protection systems and equipment; and (3) the condition and status of fire barriers used to prevent fire damage or fire propagation. The areas toured included the reactor core isolation cooling (RCIC) turbine room, salt service water system and the emergency diesel generator building. The inspector verified that any degraded conditions were properly compensated for until appropriate corrective actions could be taken.

b. Findings

No findings of significance were identified.

1RO7 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the emergency diesel generator (EDG) radiator performance data per procedure TP98-050, “Special Test for Emergency Diesel Generator Ventilation and Damper Testing,” and 3.M.3-61.4, “Emergency Diesel Generator Mid-Cycle Preventive Maintenance.” The inspector reviewed the test data for adverse trends and the test acceptance criteria against calculation M-991, the Updated Final Safety Analysis Report (UFSAR) and the EDG vendor manual.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector reviewed the performance of an operating crew in the simulator on August 27, 2001, and September 25, 2001. The inspector verified that the crews met the event scenario objectives and performed the critical tasks. The scenarios involved the loss of high pressure feedwater, and a Loss of Coolant Accident concurrent with a Loss of Offsite Power. The inspector verified proper use of the Emergency Plan and also verified that the post scenario critique discussed any relevant lessons learned. The inspector verified that the identified discrepancies during the scenario were discussed with the crew to enhance future performance.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Quarterly Maintenance Rule Inspection

a. Inspection Scope

The inspector reviewed the implementation of the maintenance rule (10 CFR 50.65) for selected systems and components. The review included a review of the applicable maintenance rule basis document and the UFSAR and included the following:

- Equipment failures for the standby liquid control system (SBLC) and accounting of equipment unavailability through a review of problem reports, and preventive and corrective maintenance requests issued since June 1999. The SLC is designated as an (a)(2) system. Problem reports (PR) reviewed included PR 00.9147 (Squib valve continuity failure) and PR 00.9376 (SBLC pump leaking oil from end cover).
- The failure of X56 transformer which supplies power to electrical panels Y4 and Y41. The failure was correctly determined to be a functional failure and was documented in PR 01.9776.
- The trip of the “B” motor-generator set in the reactor recirculation system. This trip was correctly determined to be a functional failure and was documented in PR 01.9775.
- Mechanical seal leakage on RCIC system isolation valve 1301-17. This deficiency was not considered a functional failure since the steam leakage did not affect operability. This issue was documented in PR 01.2188

b. Findings

No findings of significance were identified.

.2 Biannual Maintenance Rule Inspection

a. Inspection Scope

The inspector reviewed the periodic evaluation required by 10 CFR 50.65 (a)(3) for Pilgrim Nuclear Power Station to verify that structures, systems and components (SSCs) within the scope of the maintenance rule were included in the evaluation, and balancing of reliability and unavailability was given adequate consideration. The inspector reviewed the licensee's most recent periodic evaluation report dated August 16, 2001, for the period of July 7, 1999, to May 19, 2001. The inspector verified that the periodic evaluation was completed within the two year time period.

The inspector selected a sample of safety significant systems that were in a(1) status to verify that: (1) goals and performance criteria were appropriate, (2) industry operating experience was considered if appropriate, (3) corrective action plans were effective, and (4) performance was being effectively monitored. As of August 16, 2001, nine (9) SSCs were in a(1) status. Seven of these systems were risk significant. The inspector also reviewed the licensee's assessment of the balance between reliability and availability for these systems.

The inspector reviewed the following (a)(1) systems:

- Control Room High Efficiency Air Filtration System (CRHEAF)
- Primary Containment (PC)
- 125 VDC System
- Primary Containment Isolation System (PCIS)
- Reactor Protection System (RPS)

Additionally, the following risk significant (a)(2) systems were reviewed for balancing reliability and availability, performance monitoring and performance criteria.

- Residual Heat Removal (RHR)
- Salt Service Water (SSW)
- Neutron Monitoring (NM)

A sample of high safety significant systems going from (a)(1) to (a)(2) were also reviewed including "plant level performance criteria".

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspector reviewed several on-line maintenance work plans/activities to assess the adequacy of the licensee's risk assessment process. The inspector reviewed the plan against the criteria contained in licensee procedures 1.5.21, "Integrated Scheduling Guidelines," and 1.5.22, "Risk Assessment Process." The inspection included a review of the risk assessments and contingencies established, and verification that the increase in plant risk and protected equipment was conveyed during the licensee's morning meeting and that the plan was posted throughout the site.

The inspector reviewed the risk associated with performing corrective maintenance on the feedwater level control system, taking the residual heat removal system out-of-service for surveillance testing, and reviewed the licensee's work plan for the week of September 24, 2001.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed the following operability evaluation to verify that continued operability was justified. The Pilgrim UFSAR, technical specifications, and licensee procedure 1.3.34.5, "Operability Evaluations," were used as references to assess the adequacy of the operability evaluations. The inspector also verified that the identified corrective actions to correct the degraded condition were adequate and scheduled in the licensee's work control process.

- OE 01-042, Drywell-to-torus vent line expansion bellows

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed the results of the following post-maintenance tests (PMT) to ensure that the test activities were adequate to verify operability and functional capability of the component/system following maintenance:

- Feedwater level control amplifier replacement work in the feedwater system. The operation of the feedwater control system was tested and the PMT determined that further troubleshooting was required.

- Replacement of the accumulator for control rod drive no. 42-27. The PMT correctly included a scram time test to ensure proper operation.
- Repair of RCIC system isolation valve 1301-17 which exhibited mechanical leakage.
- Replacement of conductivity cell CE-C2 for the "B" condensate demineralizer. Chemistry technicians obtained a sample for analysis to confirm the new conductivity cell was operating correctly. Also, the mechanical joint was examined to ensure no leakage existed.
- Replacement of the master feedwater level controller.
- Overhaul of electrical breaker 52-405 for the service air compressor K-105A.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed the results of the following surveillance tests:

- Control Rod Scram Time Testing Per Procedure 9.9
- "A" Emergency Diesel Generator Initiation By RHR Logic Per 8.M.2-2.10.8.1
- RCIC Steam Line Isolation Per 8.M.2.6.1/6.4
- "A" Core Spray Loop Testing Per 8.Q.3-2 and 8.5.1.1
- "A" Emergency Diesel Generator Operability Per Procedure 8.9.1

The inspector verified that the system requirements were correctly incorporated into the test procedures and that the test acceptance criteria was consistent with the technical specifications, the licensee's Inservice Test program and the Pilgrim UFSAR. Also, any anomalies identified during testing were checked to determine proper resolution.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspector observed portions of the September 6, 2001, annual emergency planning exercise, to evaluate the drill and licensee critique. The inspector focused on event classification and notification, and communication of priorities among the emergency response organizations. The simulator, Technical Support Center, Operations Support Center, Media Center, and Emergency Operating Facility were activated. The Commonwealth of Massachusetts partially participated by sending Massachusetts Emergency Management Agency and Department of Public Health representatives to the EOF and Media Center. Response activities at the towns and other offsite agencies were simulated.

b. <u>Findings</u>

No findings of significance were identified.

40A3 <u>Event Follow-up</u>

<u>(Closed: URI 50-293/01-03-04) Reactor Vessel Water Level Instrumentation Spiking</u>

Background

On April 21, 2001, during a normal plant cool down, reactor vessel water level instrument spiking occurred which caused several actuations including: an automatic closure of the main steam isolation valves, reactor scram signal while rods were fully inserted, and isolations of the reactor water clean-up and reactor building ventilation systems. The licensee determined that air was inadvertently introduced into the reactor vessel water level reference leg several months earlier during control rod drive system pump maintenance. The licensee had not followed all of the necessary actions required to maintain the reference leg back fill system. This issue was left as an unresolved item in Section 4OA3 of NRC Inspection Report No. 50-293/2001-003.

Significance Determination

The introduction of non-condensible gases into the safety-related reactor water level instrument reference legs can cause inaccurate reactor vessel level indication following reactor coolant system depressurization events. Reactor vessel level indication errors of sufficient magnitude and duration could prevent automatic safety system functions or cause operators to take inappropriate actions. Therefore, this issue has a credible impact on safety.

A Phase 3 risk evaluation was conducted to assess the significance of this issue in accordance with the Significance Determination Process. The Phase 3 evaluation used both qualitative and quantitative risk insights to establish the safety significance of this condition. The Phase 3 risk evaluation concluded that these findings were both of very low safety significance (Green). An independent risk evaluation conducted by the licensee corroborated this conclusion. The following are several of the key factors that mitigate the risk significance of this issue:

1) The frequency of rapid depressurization of the reactor vessel which could cause level indication errors of substantial magnitude is very low. Slow depressurization events would result in a less significant challenge to operators because the magnitude of level errors would likely be smaller and more time would be available to diagnose and respond to the inaccurate level indications.

2) There are multiple reactor level instruments, some of which would not be affected by the level errors observed at Pilgrim. It is also very unlikely that level errors of similar magnitude would simultaneously occur on all level indication channels. In addition, the post accident monitoring level instruments would continue to provide useful level trend information to the operators.

3) The operators have symptom based emergency operating procedures (EOPs) that provide detailed instructions on necessary actions if inaccurate level indication is observed. The EOPs require that operators continuously monitor level indication for unreliable behavior.

4) Operators are trained to recognize the response of reactor vessel level instruments to non-condensible gas migration in the reference legs. The actual level deviations experienced at Pilgrim were very erratic. Therefore, it is extremely unlikely that the operators would mistakenly assume that the level indication was accurately reflecting vessel level.

5) For events initiated with the reactor at power, automatic functions such as low level reactor scram and emergency core cooling system (ECCS) initiation signals would not have been adversely affected by this condition. In the shutdown condition, the automatic emergency safeguards feature (ESF) isolations would remain functional.

6) The duration that non-condensible gasses were present in the reference legs of the level transmitters at Pilgrim was relatively short (~ 46 days).

Regulatory Compliance

10CFR50, Appendix B, Criterion XVI, “Corrective Actions”, requires that effective corrective actions be taken to preclude the recurrence of significant conditions adverse to quality. Entergy did not implement effective corrective actions from June 1993 to April 2001 to preclude recurrence of a significant condition adverse to quality. Several maintenance and operating provisions to prevent reactor vessel level notching were either not performed or were inadequate. For example, the following inadequacies were noted:

- Each refueling outage, operation of the back fill system was required to be secured for at least two days, and the reactor vessel level indication was intended to be checked for mismatches on redundant instrument channels. This activity was not scheduled in a preventive maintenance program and was not performed.

- The licensee deferred the back fill system flowmeter calibration which was recommended by the instrument vendor. The flowmeter had not been calibrated since January 1997.

- Procedures for back filling reference legs did not provide sufficient guidance concerning the need to purge the reference legs using a booster pump. A complete backfill was not performed since June 1995.

- Procedural guidance concerning control rod drive (CRD) charging header venting following maintenance was inadequate in that no action steps existed in procedure 2.2.87, “Control Rod Drive System.” PR 95.9316 identified a similar problem when a false low level scram signal occurred after placing certain level instruments back onto the back fill system. The PR incorrectly concluded that existing procedure guidance was adequate.

As stated above, this issue was reviewed with the SDP and determined to be of very low safety significance (Green). Therefore, this corrective action violation is being treated as a Non-Cited Violation, consistent with the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as a Non-Cited Violation (**NCV 01-06-01**). This issue is documented in the licensee's corrective action program as PR 01.9385. URI 01-03-04 is **closed**.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Bellamy, Site Vice President, Entergy Nuclear, and other members of licensee management at the conclusion of the inspection on October 9, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered propriety. No propriety information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

E. Almeida	Manager Engineering Design
S. Bethay	Director Engineering
S. Brennion	Superintendent Regulatory and Industry Relations
L. Darsney	System Engineer
P. Dietrick	General Manager Plant Operations
C. Dugger	Vice President Operations
V. Fallacara	Manager Operations
A. Felix	System Engineer
J. Gaedtke	System Engineer
R. Gay	System Engineer
S. Hudson	Maintenance Rule Coordinator
K. Kampschneider	System Engineer
K. Lane	System Engineer
W. Lobo	Regulatory Affairs
K. Mulligan	Manager Maintenance
W. Perks	Manager Technical Services
B. Riggs	Director Nuclear Assessment
C. Wend	Radiation Protection Manager
M. Williams	System Engineer
J. Yingling	System Engineer

b. List of Items Opened, Closed and Discussed

Opened and Closed

50-293/01-06-01	NCV	Ineffective Corrective Action for Reactor Vessel Level Spiking

Closed

50-293/01-03-04	URI	Ineffective Corrective Action for Reactor Vessel Level Spiking

c. List of Documents Reviewed

The Pilgrim Nuclear Power Station Maintenance Rule Periodic Assessment, dated August 16, 2001.
Monthly Update of Maintenance Rule Performance/System Status for July 2001
Monthly Update of Maintenance Rule Performance/System Status for August 2001
Heating, Ventilation & Air Conditioning System (System 24) Basis Document, dated June 29, 1998
Control Room High Efficiency Air Filtration System (a)(1) Corrective Action Plan, dated July 8, 1999

Heating, Ventilation & Air Conditioning System Performance History
Primary Containment System (System 50) Basis Document, dated 4/23/98
Feedwater Check Valves (a)(1) Corrective Action Plan, dated 9/13/01
Main Stem Isolation Valves (a)(1) Corrective Action Plan, dated 11/18/99
Primary Containment Performance History
DC Power Distribution System 250V/125V/24V (System 46G) Basis Document, dated September 5, 2001
125 VDC (a)(1) Corrective Action Plan, dated 7/19/01
Expert Panel Meeting Minutes, dated 9/5/01
Primary Containment Isolation System (System 45B) Basis Document, Rev. 2
Primary Containment Isolation System (a)(1) Corrective Action Plan, dated July 1, 1999
Primary Containment Isolation System Performance History
Expert Panel Meeting Minutes dated September 2, 1999
Reactor Protection System (System 45B) Basis Document, Rev. 1
Reactor Protection System (a)(1) Corrective Action Plan, dated July 7, 1999
Design Review Board Meeting Minutes, dated January 10, 2000
Reactor Protection System Performance History
Problem Report for the Reactor Protection System, dated August 31, 2001
Plant Level Performance Criteria Basis Document, dated April 6, 1998
Plant Level Performance History
Problem Report for Plant Level Performance Criteria dated November 8, 2000
Design Review Board Meeting Minutes dated February 20, 2001
Feedwater System (System 6) Basis Document, Rev. 1
Feedwater System (a)(1) Corrective Action Plan, dated April 16, 1998
Design Review Board Meeting Minutes dated January 17, 2000
Residual Heat Removal System (System 10) Basis Document, dated August 27, 1999
Residual Heat Removal System Performance History
Salt Service Water System (System 29) Basis Document, dated April 21, 1998
Salt Service Water System Performance History
Neutron Monitoring System (System 45A) Basis Document, Rev. 1
Neutron Monitoring System Performance History

d. <u>List of Acronyms</u>

CFR	Code of Federal Regulations
CRHEAF	Control Room High Efficiency Air Filtration System
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESF	Emergency Safeguards Feature
NM	Neutron Monitoring
PC	Primary Containment
PCIS	Primary Containment Isolation System
PMT	Post Maintenance Test
PR	Problem Report
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPS	Reactor Protection System
SBLC	Standby Liquid Control System

SSC	Structures, Systems and Components
SSW	Salt Service Water
UFSAR	Updated Final Safety Analysis Report