March 18, 2002

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 - NRC INSPECTION REPORT 50-293/01-12

Dear Mr. Bellamy:

On February 16, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report documents the inspection findings which were discussed on March 5, 2002, with Tom Trepanier 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.

No findings of significance were identified.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate Entergy Nuclear Generation Company compliance with these interim requirements.

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/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Clifford Anderson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 50-293/01-12

Attachment: Supplemental Information

cc w/encl:
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W. Riggs, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
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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
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	50-293/01-12
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	December 30, 2001, through February 16, 2002
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector E. Gray, Sr. Reactor Inspector
Approved By:	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-12; on 12/30/2001 - 02/16/2002; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station, Resident Inspection.

The inspection was conducted by resident inspectors, and a senior reactor inspector. This inspection identified no significant findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. 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/ASSESS/index.html.

A. Inspector Identified Findings

None

B. Licensee Identified Violations

None

Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period with the mode switch in shutdown and the plant in a cold shutdown condition following an automatic shutdown that had occurred prior to this inspection period on December 27, 2001. On December 30, 2001, the mode switch was placed in startup and the reactor was taken critical. The unit returned to 100 percent power on January 1, 2002. On January 30, 2002, power was temporarily reduced to 85 percent to perform control rod drive testing.

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

1R04 Equipment Alignment

b. Inspection Scope

The inspector conducted a partial system walkdown of the core spray and high pressure coolant injection systems. This included reviewing applicable plant and information drawings and normal operating procedures. The inspector reviewed valve static mimics in the control room and walked down accessible portions of the systems to ensure proper system alignment. The inspector confirmed that the systems were properly aligned to support normal and emergency plant operations.

c. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Fire Protection Inspection

a. Inspection Scope

Four 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 readiness of fire barriers used to prevent fire damage or fire propagation. The areas toured included: the intake structure, the “B” residual heat removal system area, the high pressure coolant injection system area, and the emergency diesel generator building. The inspector verified that adequate compensatory measures were in place for degraded or inoperable fire protection equipment.

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515/146, Hydrogen Storage Locations

a. Inspection Scope

The inspector conducted a tour of the protected area to verify that the licensee was providing greater than 50 feet separation between the bulk hydrogen storage and (1) ventilation intakes and, (2) risk significant tanks or Structures, Systems, or Components.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the licensee's programs and processes for assuring that safety-related heat exchangers were operationally maintained and capable of performing their design function. Specifically, the inspector verified that the licensee's maintenance, testing, inspection and evaluation of results were adequate to ensure proper performance of the following heat removal systems and heat exchangers:

- The salt service water (SSW) system, including trash rakes, traveling screens, pumps, and instrumentation.
- Reactor building closed cooling water (RBCCW) heat exchanger that interfaces with the SSW system.
- The residual heat removal (RHR) heat exchanger.

The inspector reviewed heat exchanger test methodology, frequency of testing, test conditions, acceptance criteria and trending of results. The inspection, cleaning and maintenance methods used to evaluate the SSW and RBCCW systems reliability were reviewed with design and system engineers. This was to verify the methods used for inspection and cleaning were consistent with expected degradation and that the final condition of the heat exchangers was acceptable. Selected test calculations of component performance data were reviewed to verify the test results reflected heat exchanger condition and that operation was consistent with design. The inspector assessed the trending of the measured data for the components inspected. The salt water intake conditions, including depth evaluation of internal bays, and the debris screens, were reviewed. The status and effectiveness of the chlorination system for the SSW system was reviewed. The inspector conducted walkdowns of the RBCCW and RHR heat exchangers, the SSW system components, and instrumentation available to plant operators to assess their material condition. Also, a sample of problem reports related to the extent of biofouling, debris fouling, and chlorination control were reviewed to verify the licensee entered the problems into their corrective action program and provided appropriate corrective actions.

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 control room simulator on February 7, 2002. The scenario involved a small steam leak in the steam tunnel, the failure of the main steam isolation valves to automatically close and multiple control rods which did not insert after a scram. Operators demonstrated proficiency using EOP-2, "RPV Control - Failure To Scram," and EOP-4, "Secondary Containment Control" procedures. The inspector witnessed good interaction between operating crew members and the training instructors. Formal communications were used between crew members. A post scenario discussion covered all operational aspects, including any opportunities for improvement.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed the implementation of the maintenance rule (10 CFR 50.65) for selected systems and components. The review included applicable maintenance rule basis documents and the Updated Final Safety Analysis Report (UFSAR) and included the following specific equipment issues:

- Proper classification of equipment failures for the high pressure coolant injection (HPCI) system. The inspector reviewed problems reports (PR) issued within the last two years and reviewed the HPCI maintenance rule basis document. Problem reports reviewed included PR 00.9165 (HPCI inverter failure alarm) and 91.9182 (blown fuse in HPCI controller).

- Proper classification of equipment failures for the primary containment isolation system (PCIS). The inspector reviewed problems reports issued within the last two years and reviewed the PCIS maintenance rule basis document.

- On February 8, 2002, operators noted an abnormal noise emanating from the "A" salt service water (SSW) pump motor bearing. The pump was secured and the control switch was danger tagged in the pull-to-lock position. The system engineer preliminarily determined that this equipment problem was a functional failure as defined by the maintenance rule; the final evaluation will be determined pending the results of the pump disassembly. The SSW system is currently classified as an a(2) system.

- During a routine stroke time surveillance test, valve MO-1001-28A, "A" loop residual heat removal (RHR) Outboard Injection Valve, exhibited erratic operation. Troubleshooting by the licensee determined that the valve was inoperable due to an internal motor operator problem. Specifically, a castle nut inside the motor actuator came loose and caused misalignment of the gearing. The control room operators issued problem report (PR) 02.9070 to document, evaluate and correct the problem. The system engineer indicated the component failure would be classified as a maintenance preventable functional failure under the maintenance rule criteria. Sufficient margin existed that this failure would not cause the RHR system to be classified as an a(1) system. The inspector determined that the licensee properly followed the criteria in the maintenance rule.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspector reviewed the risk management controls used during a significant emergent work activity on valve MO-1001-28A, "A" loop residual heat removal (RHR) Outboard Injection Valve. On February 8, 2002, valve MO-1001-28A failed its routine stroke time surveillance test, rendering it inoperable. The licensee reperformed the risk assessment to include the emergent work activity and determined that there was only a slight increase in core damage risk, and the overall risk remained low. The inspector determined that the licensee properly evaluated and managed risk during an emergent equipment problem related to valve MO-1001-28A.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed the following operability evaluations to verify that continued operability was justified. The Pilgrim Updated Final Safety Evaluation Report (UFSAR), technical specifications, and licensee procedure 1.3.34.5, "Operability Evaluations," were used as a reference to assess the adequacy of the operability evaluations. The inspector also checked that the identified corrective actions to correct the degraded conditions were adequate and scheduled in the licensee's work control process.

- OE 02-01 Anticipated Transient Without Scram (ATWS) system level transmitter 263-120B bypass valve leaks-by and may cause level notching on the "B" side level instrument. This equalizing valve leakage was suspected during a diagnostic test performed as part of the root cause review of the December 27, 2001 shutdown notching event. Further details are contained in PR 02.9042 which was initiated on January 23, 2002. In a verbal operability determination, the licensee concluded the "B" side reactor vessel water level instrumentation remained operable based on previous generic BWR studies contained in NRC Generic Letter 92-04 and a licensee engineering evaluation approved at ORC Meeting 92-72. The licensee planned to replace the suspected leaking bypass valve at the end of this inspection period. As of February 5, 2002, no risk significance color had been assigned to this OE since the written operability determination had not yet been completed.

After this inspection period, the licensee determined that the ATWS system level transmitter bypass valves did not contribute to the level notching identified on December 27, 2001. While this changed a possible cause for the event, it did not change the licensee's operability determination for the affected instrumentation. The licensee had implemented more frequent use of the reference leg backfill system to ensure continued operability of the instrumentation and to decrease vulnerability to non-condensable gas migration from the reference legs to the instrument racks. The licensee continues to evaluate this situation and has initiated a design change which would restore the reference leg back-fill system to a continuous mode of operation.

- OE 01-063 Station blackout diesel generator ring gear not manufactured to OEM specifications. The licensee contacted the vendor and determined that the non-conforming ring gear design is superior to the OEM design. The licensee plans on issuing a plant design change by March 31, 2002 to accept this non-conforming condition.

- OE 01-067 Packing leak on valve 1201-205, a 3/4 inch test connection valve located inside the reactor water cleanup (RWCU) heat exchanger room. This valve is located between the RWCU system containment isolation valves. The packing leak resulted in a 4 inch steam plume. The licensee completed a written operability evaluation that documented the basis for continued operability. During this inspection period, the licensee performed a temporary leak repair process which stopped the packing leak.

- OE 01-058 Voltage regulating transformer X-57 has a slight under voltage condition. The available voltage was determined to be slightly above the minimum requirement. The transformer supplies power to the 120V/240V safeguard panel Y13, which feeds the train "A" H2/O2 analyzer system, post accident sampling system (PASS) system heat tracing, and PASS sample station. The licensee has scheduled to replace internal tap control boards. Engineering personnel have initiated a review to determine whether this degraded performance was age related.

b. <u>Findings</u>

No findings of significance were identified.

1R19 <u>Post-Maintenance Testing</u>

a. <u>Inspection Scope</u>

The inspector reviewed the following post maintenance testing activities:

- MR 01122493, Replace standby diesel generator fuel pump gaskets
- MR 02101578, Replace ATWS level transmitter LT-263-120B manifold
- MR 02102886, Troubleshoot LPCI loop "A" injection throttle valve MO-1001-28A

The review included ensuring that the effect of the test on plant had been evaluated adequately, verifying the test data met the required acceptance criteria, and that the test activity was adequate to verify system operability and functional capability following maintenance.

2. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed the results of the following surveillance tests:

- 8.9.1, "Emergency Diesel Generator and Associated Emergency Bus Surveillance,"
- 8.5.3.1.2, "Salt Service Water System Pump and Vale Operability Test with Full Flow Test Conditions"

The inspector verified that the test acceptance criteria was consistent with technical specifications and Updated Final Safety Analysis Report requirements, the test was performed in accordance with the written procedure, the test data was complete and meet procedural requirements, and the system was properly returned to service following testing.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA3 Event Followup

(Closed) LER 50-293/2001-04 and LER Supplement 50-293/1999-04-01: Target Rock Relief Valves Setpoint Drift Issues. During the cycle 12 refueling outage (RFO 12), three (i.e., 3A/B/C) of four Target Rock, two-stage main steam relief valves experienced setpoint drift. The as found popping pressures would not have resulted in exceeding the code allowable pressure for the reactor vessel. These three SRVs pilot assemblies were replaced with certified pilot assemblies. The root cause was determined to be stellite oxidation between the pilot disc and seat. During RFO 13, two SRV pilot assemblies (i.e., 3B and 3C) again experienced setpoint drift, but showed improvement from the test results in RFO12. The pilot assemblies were again replaced. One corrective action after RFO12 required removing the leaking pilots prior to reactor vessel flood up, which may have accounted for the improved performance. Based on industry experience, the licensee had already replaced the existing valve discs with Stellite 21 to further improve performance. These LERs adequately documented the related issues and corrective actions and are considered **closed**.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Tom Trepanier, Acting Site Vice President and other members of licensee management at the conclusion of the inspection on March 5, 2002. 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. List of Items Opened, Closed and Discussed

Closed

LER 50-293/2001-04	Target Rock Relief Valves' Test Pressures Exceed Technical Specification Limit
LER 50-293/1999-04-01	Setpoint of Target Rock Relief Valve Found Out of Tolerance During Testing

b. List of Documents Reviewed

Pilgrim Procedure 7.8.1, Rev 22. Water Quality Limits
Pilgrim Procedure 8.5.3.14, Rev 14. SSW Flow Rate Operability Test
Pilgrim Procedure 8.5.3.14.1, Rev 3. RBCCW Heat Exchanger Thermal Performance Test
Pilgrim Procedure NOP 02E1. Rev 0. SW Inspections, Maintenance, & Testing in Response to Generic Letter 89-13.
Pilgrim Specification M561, Rev E6 for SSW & RBCCW SR Piping and Heat Exchanger Inspection, Maintenance & Test Requirements in Response to Generic Letter 89-13.
Calculation No. M-641, Rev 0. RBCCW Heat Exchanger Performance
Calculation No. M-710, Rev 0. Heat Exchanger Performance Testing
Calculation No. M-1036, Rev 0. Evaluation of RBCCW Suction Header Piping
Calculation No. M-663, Rev 1. RHR Heat Exchanger Performance
Calculation No. M-664, Rev 1. Containment Heat Removal
EPRI TR-107396, Closed Cooling Water Chemistry Guideline, dated October 1997
RBCCW Strategic Chemistry Plan, Section 2 as current on 1/17/02
Pilgrim Service Water Operational Performance Inspection (SWOPI) Closeout Report dated December 1999
Design Basis Document SDBD-29, Rev E0. SSW System
Design Basis Document SDBD-30A, Rev E0. RBCCW System
Design Basis Document SDBD-10, Rev E0. RHR System
SSW, RHR and RBCCW System Report Cards, current as of 1/17/02
Drawing M212 SH1, Rev E80. P&ID, Service Water System
Drawing M27, Rev E12. Intake Structure Plan and Sections

c. List of Acronyms

ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DBT	Design Basis Threat
EOP	Emergency Operating Procedure
HPCI	High Pressure Coolant Injection
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
MR	Maintenance Request
OE	Operability Evaluation
ORC	Operations Review Committee
PARS	Publicly Available Records
PASS	Post Accident Sampling System
PCIS	Primary Containment Isolation System
PR	Problem Report
RBCCW	Reactor Building Closed Cooling Water
RFO	Refueling Outage
RHR	Residual Heat Removal
RWCU	Reactor Water Cleanup
SSW	Salt Service Water System
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report