June 22, 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 INTEGRATED INSPECTION REPORT NO. 05000293/2001-003

Dear Mr. Bellamy:

On May 17, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report presents the results of that inspection. The results were discussed on June 4, 2001, with Mr. R. Bellamy and other members of your staff.

This inspection was an examination of 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified one issue, the safety significance of which is to be determined (TBD). The issue involved an apparent violation of NRC requirements. However, because the safety significance of the issue is under review by the NRC, it is being treated as an unresolved item.

The inspectors also identified one issue of very low safety significance (Green) that was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest these non-cited violations, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at the Pilgrim facility.

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 Cowgill, Chief
Projects Branch 6
Division of Reactor Projects

Docket No. 05000293
License No. DPR-35

Enclosure: Inspection Report 05000293/2001-003
Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	05000293
License No:	DPR-35
Report No:	05000293/2001-003
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	April 1, 2001, through May 17, 2001
Inspectors:	D. Dempsey, Acting Senior Resident Inspector R. Arrighi, Resident Inspector J. Furia, Senior Health Physicist T. Burns, Reactor Inspector C. Sisco, Operations Engineer N. McNamara, Emergency Preparedness Inspector
Approved By:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-03; on 04/01 - 05/17/2001; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station. Correction of Emergency Preparedness Weakness and Deficiencies, Problem Identification and Resolution, and Licensee Identified Violation.

The inspection was conducted by resident inspectors, a radiation safety inspector, an operations engineer, an emergency preparedness inspector and a reactor inspector. The inspection identified one Green finding, which was a non-cited violation and one unresolved item involving an apparent violation of NRC requirements. 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: Emergency Preparedness

- Green. The inspector identified a Non-cited Violation for failure to maintain respirator qualifications current, as required by the licensee's Emergency Plan, Section O and 10 CFR 50.54(q).

  The finding was of very low safety significance because there were sufficient responders with respiratory qualifications to fill the positions. Twenty-three percent of the responders were not qualified. (Section 1EP5).

Cornerstone: Mitigating Systems

- TBD. The inspector identified an apparent violation of the corrective action requirements of 10 CFR 50, Appendix B, Criterion XVI involving failure to implement effective actions to preclude recurrence of spiking of reactor vessel water level instruments when the reactor coolant system is depressurized.

  The safety significance of the finding is under review, and the apparent violation is being treated as an unresolved item. (Section 4OA2)

B. Licensee Identified Violations

- A violation of very low significance, which was identified by the licensee, has been reviewed by the inspector. Interim corrective actions taken by the licensee appear to be reasonable. The violation is listed in Section 4OA7 of this report.

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Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period at 100 percent core thermal power. On April 20, 2001, operators commenced a plant shutdown and cooldown of the reactor for the commencement of refueling outage 13 (RFO13). On May 16, 2001, after completion of RFO13 activities, the mode switch was placed in startup and operators were pulling rods to achieve criticality, when they manually scrammed the reactor from a subcritical condition. Reactor water level was rising with no reject path due to an automatic isolation of the reactor water cleanup system (reference Section 40A7). On May 17, 2001, operators brought the reactor critical and on May 19, 2001, placed the unit on line.

1. REACTOR SAFETY

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector reviewed the licensee's preparations for a potential coastal storm on April 18, 2001. The licensee entered procedure 2.1.37, "Coastal Storm - Preparation and Actions," based on predicted north-to-east winds of 15 to 25 miles per hour (Northeasterner). In the control room, the inspector monitored intake structure seawall level, traveling screen differential pressure, and seawater pump current, and verified that readings were within acceptable limits. Through review of Attachment 1 of procedure 2.1.37, "Screenhouse Parts and Tools Inventory List," the inspector verified that required equipment was staged appropriately and that traveling screen tagouts were prepared. Finally, the inspector toured the intake structure and verified normal operation of the traveling screens, the screenwash system, and the safety-related salt service water pumps.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection scope

The inspector toured several plant areas important to reactor safety to observe conditions related to: (1) transient combustibles and ignition sources; (2) the material condition and readiness of fire detection and suppression equipment; and (3) the conditions and status of fire barriers that are used to prevent fire damage or fire propagation. Applicable portions of the Pilgrim Final Safety Analysis Report and Specification 89XM-1-ER-Q, "Updated Fire Hazards Analysis," were reviewed to facilitate the tours. The areas that were toured included both salt service water pump rooms, the "A" and "B" train emergency diesel generator rooms (including the compressor and day tank rooms), the intake structure and fire pump areas, and the reactor core isolation cooling, high pressure coolant injection, control rod drive, and "A" train residual heat removal system corner room.

The inspector also reviewed the following procedures and surveillance test results:

- 2.2.28 Dry Chemical Systems
- 8.B.21 Emergency Lighting Units (Fixed)
- 8.B.4.12 Fire Panel C93, Emergency Diesel Generator Building Functional Test
- 8.B.6.1 EDG "A" Pre-Action Sprinkler System Functional Test
- 8.B.7 Fixed Dry Chemical Fire Protection Systems
- 8.B.17.2 Inspection of Fire Damper Assemblies
- 8.B.1 Fire Pump Test

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

The inspector observed selected samples of nondestructive examination (NDE) activities. Also, the inspector performed a documentation review of selected NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems in which degradation would result in a significant increase in risk of core damage. The observations and documentation review were performed to verify that the activities were performed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspector reviewed a sample of inspection reports and deviation reports initiated as a result of problems identified during ISI examinations. Also, the inspector evaluated effectiveness in the resolving and correcting problems identified during ISI activities.

The inspector observed ultrasonic testing (UT) and magnetic particle (MT) testing activities to verify effectiveness of the examiner and process in identifying degradation of risk significant systems, structures and components and to evaluate the activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code. The inspector observed the ultrasonic tests performed on high pressure coolant injection (HPCI) system welds 23-I-16 and 23-I-17 and the MT of integral attachment welds 10R-O-25HL1(4) to the residual heat removal (RHR) piping system. In addition, the inspector reviewed the radiographic examination and test results of butt welds 10R2-IB-7 and 10R-IB-8 made during replacement of RHR check valve 1001-068B.

The inspector reviewed a sample of video recordings of the remote in-vessel visual inspection of the in vessel core spray piping base material, butt welds and tee boxes. The inspector also reviewed the visual examination of the core shroud stabilizer assemblies including the rods, pins, springs and cap screws. In addition, the inspector reviewed the remote visual inspection of selected jet pumps with specific attention to the set screws, mixer wedge, restrainer brackets and swing gate assemblies. This review was conducted to confirm that the test conditions permitted the performance of the visual (VT-3) examination of the selected vessel internals. Also, the inspector confirmed that for the recordings evaluated, the visual examination was in compliance with the requirements of ASME Section XI. The inspector reviewed the plan, procedures, and results of the visual examination of the containment liner for compliance with the requirements of ASME Section XI, IWE (requirements for class MC and metallic liners of class CC components). Inspection reports for selected portions of the liner were reviewed by the inspector to assess the extent of coating failure, corrosion and damage to moisture barriers and the adequacy of the corrective action specified for the identified nonconforming conditions.

The inspector reviewed welding activities associated with the repair and replacement of selected components to verify that the activities were performed in accordance with the requirements of ASME Sections IX and XI. The inspector reviewed maintenance request MR E9800005 (replacement of check valve 1001-68B) in the RHR system. The inspector reviewed the joint process control instructions, welding instructions and activities, NDE requirements, and the test results of welds 10R-IB-7R and 10R-IB-8R in the RHR system.

The inspector interviewed the licensee's radiographic personnel responsible for the review and approval of test results. Radiographs of welding activities were reviewed to ensure proper identification, characterization, and size of rejectable indications for welds 10R2-IB-7 and 10R2-IB-8 in the RHR system.

The inspector examined the licensee's evaluation and disposition of non conforming conditions identified during ISI activities (problem reports 01.1841, 99.1145.00 and nonconformance reports 99-065, 074, 091 and 99-098) and verified that the analyses justified continued operation without repair or rework of the nonconforming conditions.

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) as related to the following:

- The Automatic Depressurization System (ADS) logic and the performance of the safety relief valves. The inspector reviewed a listing of problem reports (PR) that were generated during the previous 18 months and found that there were none concerning the performance of the ADS.

- The 24/125/250 VDC system. The inspector reviewed PR 01.0908 concerning the omission of the Y-10 transfer switch SE relay from the scope of the maintenance rule and PR 01.1904 concerning failure of a 250 VDC battery cell during testing.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspector observed operating crew performance during two non-routine events that occurred during the inspection period:

- On April 21, 2001, with the plant in the hot shutdown condition, automatic Group 1 isolation (main steam isolation valve closure), reactor scram, and other system isolation signals occurred due to reactor vessel level instrumentation spiking.

- On May 16, 2001, operators manually scrammed the reactor prior to attaining criticality due to loss of the reactor water cleanup system letdown path and reactor vessel water level control.

b. Findings

No findings of significance were identified. Other aspects of the April 21, and the May 16, events are discussed in Section 4OA2 and Section 4OA7, respectively.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- MR 01110271 Repair HPCI exhaust leak downstream of tap-off for 2301-106
- MR 01110273 Repair RCIC exhaust line vacuum breaker VRV-9067
- MR P9900516 Disassemble, inspect, repair core spray injection valve MO-1400-25B

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspector observed and reviewed selected refueling outage activities to verify that technical specification requirements were met and that risk, industry experience, and previous site specific problems were considered.

- Outage Plan: The inspector reviewed detailed outage schedules and temporary procedures to verify that technical specification required safety system availability was maintained (or exceeded), that risk was considered, and that contingency plans existed for restoring key safety functions such as electrical power and primary cooling system makeup. The documents reviewed included procedure TP01-035, "RFO13 Compensatory Measures," and temporary modification TM01-11, "Installation of Temporary Panel, Temporary Battery, and Associated Cables," and "Water Management For RFO-13."

- Plant shutdown and cooldown: The inspector observed portions of the plant shutdown and cooldown from April 20 - 22, and verified that the technical specification cooldown rate limits were satisfied. The procedures observed included: (1) 2.1.5, "Controlled Shutdown From Power," 2.1.6, "Reactor Scram," (3) 2.1.7, "Vessel Heatup and Cooldown," (4) 2.2.22.5, "RCIC Injection and Pressure Control."

- During the course of the refueling outage, the inspector observed selected reactor vessel disassembly activities and walked down clearances to verify that tagouts were properly hung and that equipment was configured properly. The inspector walked down clearances 54-0001-D, "Reactor Cavity Flood Up Instructions," 46-0004, "4KV Bus A6 Maintenance and Testing," 01-0045-A, "Main Stream Isolation For Flood Up," and 01-01, "Repair Control Rod HCU 34-35." The inspector also reviewed 10 CFR 50.59 safety evaluation 3366, "Removal of Last (3rd) Layer of Reactor Cavity Shield Plugs 1 Hour Afer Shutdown with Reactor Temperature > 212 Degrees," and verified (1) that all control rods were fully inserted prior to removal of the third shield plug, and (2) that the temporary shielding specified in the safety evaluation was installed.

- The inspector verified through review of the outage schedule and plant tours that the licensee maintained and adequately protected electrical power supplies to safety-related equipment, and that technical specification requirements were met.

- The inspector periodically verified proper alignment and operation of the shutdown cooling and spent fuel pool cooling systems. The verification also included maintenance of protected reactor cavity and fuel pool makeup paths and water sources, and administrative control of potential drain down paths.

- The inspector reviewed procedures 4.3, “Fuel Handling,” and the results of procedure 8.10.1, “Refueling Platform Interlocks Functional Test,” to ensure that technical specification requirements for fuel movement were met. From the refuel bridge, the inspector witnessed movement of 15 fuel bundles and verified their proper location in the core. The inspector also verified through review of procedure 8.7.3, “Secondary Containment Leak Rate Test,” that containment requirements for fuel movement were met.

- The inspector observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and surveillance test reviews that the safety-related equipment required for mode changes was operable, that containment integrity was set, and that reactor coolant boundary leakage was within technical specification limits. Procedures reviewed included: (1) 2.1.1, “Startup From Shutdown,” (2) 8.7.1.7, “Local Leak Rate Testing of the Containment Personnel Air Lock,” and (3) 3.M.4-9, “Inspection of the Drywell and Suppression Chamber.”

b. Findings

During the reactor startup on May 16, 2001, prior to attaining criticality, the licensee lost the ability to maintain reactor vessel level through reactor water cleanup system letdown and manually scrammed the reactor. This event is discussed in Sections 1R22 and 4OA7.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed the following surveillance test procedures and associated testing activities to assess whether: (1) the test preconditioned the components tested, (2) the effect on the plant was adequately addressed by engineering personnel, (3) the system requirements were correctly incorporated into the test procedures and the test acceptance criteria were consistent with technical specifications, the licensee’s In-service Testing Program, and the Updated Final Safety Analysis Report, (4) the test was performed in the proper sequence and in accordance with the written procedure, and (5) the test equipment was removed following testing and system configuration was restored to a normal state of readiness.

The inspector reviewed and observed portions of the following surveillance tests. The review also included an evaluation of the completed surveillance test data to verify that it met the procedure requirements.

- 8.7.1.5 Local Leak Rate Testing of Primary Containment Penetrations, Isolation Valves, and Inspection of Containment Structure, Attachment 29
- 8.A.2 Drywell to Suppression Chamber Vacuum Breaker Leakage Rate Test
- 8.5.3.2.1 Salt Service Water Pump and Valve Operability Tests With Full Flow Conditions (“E” salt service water pump)

- 8.5.4.3 High Pressure Coolant Injection Operability Demonstration and Flow Rate Test At 150 Psig
- 8.5.5.3 RCIC Flow Rate Test At Less Than or Equal To 150 Psig
- 8.M.3-1 Special Test For Automatic ECCS Load Sequencing of Diesels and Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test
- 3.M.3-47 Load Shed Relay Operational/Functional Test
- 8.9.1.1 Diesel Fuel Oil Transfer System Pump and Valve Quarterly Operability

As part of the inspection, the following problem reports related to surveillance testing were reviewed to verify that the licensee has appropriately identified and corrected problems:

- PR 01.9379 Inability to record reactor vessel head temperatures
- PR 01.9273 Standby gas treatment system “B” placed in service incorrectly
- PR 01.1279 Emergency diesel generator voltage regulator exercise
- PR 01.9341 High vibration “E” salt service water pump
- PR 01.2265 Failure of RCIC turbine exhaust line vacuum breaker

During the performance of procedure 8.7.1.5, the test director failed to establish an appropriate test boundary and to ensure that an adequate vent path existed during testing of valve AO-220-45, causing an automatic (Group 6) isolation and manual scram. This issue is discussed further in Section 4OA7.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector reviewed three temporary plant modifications using 10 CFR 50, Appendix B, Criterion III, “Design Control,” procedure 1.5.9, “Temporary Modifications,” and 10 CFR 50.59, “Changes, Tests, and Experiments,” as acceptance criteria. The inspection included review of the modifications and associated safety evaluations to ensure that they did not adversely affect permanent system availability or operability. The inspector walked down the modifications to verify that they were correctly installed, and discussed their installation with plant technicians and operators. The following modifications were reviewed:

- TM01-03 Reactor shutdown level during RFO13 vessel disassembly
- TM01-05 Monitoring fuel pool cooling flow, skimmer surge tank level, spent fuel pool temperature, and vessel water temperature
- TM01-11 Installation of temporary panel, temporary battery, and associated cables.

With regard to modification TM01-11, the inspector walked down the temporary battery system and verified that the installation conformed with the requirements of specification E-347A, “Design, Procurement, Installation, and Termination and Miscellaneous Electrical Items.”

The inspector verified that problems associated with temporary modifications were being addressed and properly resolved in the licensee’s corrective action program. The following problem reports associated with modification TM01-03 and TM01-05 were reviewed:

- PR01.1801 Coordination of swapover from reactor shutdown level indication to temporary modified C905 level indicator
- PR01.1982 TM01-03 paperwork no properly completed
- PR01.1984 Incomplete documentation of MR10000857

b. Findings

No findings of significance were identified.

1EP2 Alert and Notification System Testing (ANS)

a. Inspection Scope

In accordance with NRC Inspection Procedure 71114, Attachment 02, an onsite review of the licensee’s Prompt Alert and Notification System (PANS) was conducted to ensure prompt notification of the public to take protective actions. The inspector reviewed: (1) the design of the siren system; (2) siren testing data; and (3) maintenance records for correcting siren failures. In addition, the inspector reviewed the following procedures: EP-AD-417, “Annual Siren Test Program,” EP-AD-418, “Monthly Testing of the PANS Two-Way System,” and EP- AD-419, “Annual Maintenance of the PANS Two-Way System.” The inspector interviewed the personnel responsible for maintaining and testing the system and observed a monthly test of one of the offsite sirens.

b. Findings

The siren test records and the siren tracking system were not reviewed for accuracy and consistency, which resulted in incorrect reporting of the performance indicator (PI) data to the NRC. The inspector reviewed nine months of siren testing data and found the following:

a. The inspector found two problems that contributed to the PI data being incorrectly reported: (1) Seven out of nine months of data were entered incorrectly into the database used for calculating the PIs because the system users were not familiar with the system and were copying erroneous data from month to month. This resulted in counting successes as failures which the inspector confirmed by the original test records; and (2) The licensee did not include siren rotation failures as failures towards the PI. Preliminarily, as reported by licensee representatives, the counting errors should not result in a PI color change for past quarters.

b. The siren procedures were not specific in describing the PASS/FAIL criteria or actions to be taken following a siren failure (e.g retest), and a few test records indicated that a PASS/FAIL criteria had been applied inconsistently.

c. Siren failures were not entered into a system for tracking, assessing, or trending to ensure that corrective actions were timely.

The inspector noted that inspection findings (b) and (c) may have contributed to the incorrect PI data. Overall, the inspector was able to determine that the licensee was testing and performing routine maintenance on their offsite sirens and that the sirens were operable as a part of the ANS. The licensee indicated that they would recalculate the ANS PI and make the appropriate changes during the next reporting period. The licensee initiated problem reports 01.1952, 01.1953 and 01.1954 to document and correct these issues. Based on outstanding licensee corrective action to recalculate the PI, make it accurate and verify that a change in color band did not result, this issue is being treated as an Unresolved Item. **(URI 50-293/01-03-01)**

The licensee's Emergency Plan does not describe the types of siren testing used for verifying operability (i.e., silent test every two weeks, growl test every quarter, complete cycle annually as described in NUREG 0654 and FEMA-REP 10, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"). NUREG 0654 was used as guidance for Emergency Plan reviews by the NRC and the design review of the ANS by the Federal Emergency Management Agency (FEMA). For example, the licensee does not do a quarterly growl test, but does sound the system semi-annually. The siren vendor states that a growl test is inappropriate for the licensee's system. As a result of this inspection, it was not clear what testing details and/or deviations from the standard were accepted by the NRC. This issue is considered unresolved pending further review of licensee commitments regarding siren testing requirements that have been accepted by the NRC staff and FEMA. **(URI 50-293/01-03-02)**

1EP3 Emergency Response Organization (ERO) Augmentation Testing

a. Inspection Scope

An onsite review of the licensee's ERO staffing commitments and the process for notifying the ERO was conducted to verify the readiness of key ERO staff to respond to an event and for timely facility activation. The inspector reviewed the qualification records of key ERO positions, procedures for initiating ERO call-in, and surveillance test records of the computerized automated notification system (CANS). Also, two augmentation call-in drill reports and a self assessment report on the adequacy of ERO response during off hours was reviewed to determine if the licensee identified ERO augmentation deficiencies. Interviews with newly assigned emergency responders were performed to ensure their understanding of the call-in procedures. The review was conducted in accordance with Attachment 03 of NRC Inspection Procedure 71114.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) Revision Review

a. Inspection Scope

A regional in-office review of revisions to the Nuclear Emergency Plan, Implementing Procedures, and EAL changes was performed to determine that the changes did not decrease the effectiveness of the Emergency Plan. The reviewed revisions covered the period from January 1, 2001, through April 27, 2001, and was conducted in accordance with Attachment 04 of NRC Inspection Procedure 71114.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness (EP) Weaknesses and Deficiencies

a. Inspection Scope

The inspector reviewed corrective actions implemented by the licensee pertaining to findings from the licensee's annual quality assurance audit, drill reports, regular self-assessments, and from self-revealing problems arising from surveillance tests and actual events. Problem reports assigned to the EP department also were reviewed to determine the significance of the issues and to determine if repeat problems were occurring. In addition, the inspector reviewed the quality assurance reports for 1999 and 2000 to assess that the reviews met 10 CFR 50.54(t) requirements and whether any repeat issues were identified. The review was conducted in accordance with Attachment 05 of NRC Inspection Procedure 71114.

b. Findings

The inspector reviewed a selection of ERO qualification records and found that 23% of the ERO responders required to maintain respirator qualifications had let their qualifications lapse. This issue was identified previously by the licensee in a QA audit conducted in 1998, and again in 2000, and was characterized in the licensee's PR system as a significant condition adverse to quality. This issue also was identified by the NRC in1997. The inspector determined that the licensee had not taken adequate corrective actions to prevent recurrence of this issue. In accordance with the licensee's Emergency Plan, Section O, "Emergency Response Training," and as described further in Section B of the Nuclear Training Manual, ERO staff that are required to wear respirators must have annual respirator training that includes a medical physical and a mask fit test. This issue was entered into the licensee's corrective action system as PR 01.1981.

Inspection Manual Chapter 0609, Appendix B, was used to assess the risk significance of this finding. While this finding was a failure to implement a regulatory requirement, the inspector determined this issue to be of very low safety significance (Green) because there were sufficient responders with respiratory qualifications to fill the positions. The inspector determined this issue was an implementation problem and not a failure to "meet" a planning standard (Sheet 1, Appendix B, MC 0609). It is more than minor because the issue involves personnel radiation protection. However, 10 CFR 50.54(q) states that licensees will follow and maintain in effect an E-Plan which meets the planning standards of 10 CFR 50.47(b) and the requirements of 10 CFR Part 50, Appendix E. This is a violation of 10 CFR 50.54(q), E-Plan, Section O, and the Nuclear Training Manual, Section B, which describes the qualifications necessary to maintain proficiency as an emergency responder. This violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NUREG 1600). **(NCV 50-293/01-03-03)**

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control (7112101)

a. Inspection Scope

The inspector examined exposure significant work areas, high radiation areas, and airborne radioactivity areas in the plant and reviewed associated controls and surveys of these areas to determine if controls (i.e., surveys, postings, barricades) are acceptable. Areas examined were determined by the work being performed in support of the refueling outage, and included: local power range monitor removal; main steam isolation valve inspection, testing and repair; turbine and generator inspection and repair; feedwater heater inspection and repair; and, safety relief valve removal/installation. Observation of work activities occurred during both day and night shifts. For these areas, the inspector reviewed all radiological job requirements and attended job briefings; determined if radiological conditions in the work area were adequately communicated to workers through briefings and postings; verified radiological controls, radiological job coverage and contamination controls; and verified the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place, whether licensee surveys and postings were complete and accurate, and that air samplers were properly located. Reviews of RWPs used to access these and other high radiation areas, and to identify what work control instructions or control barriers have been specified, was conducted. Plant technical specification (TS) 5.7 and the requirements contained in 10 CFR 20, Subpart G were utilized as the standard for necessary barriers. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspector also examined the licensee's programmatic controls for highly activated/contaminated materials (non-fuel) stored within the spent fuel pool.

The inspector also examined the circumstances regarding a release of primary steam from the reactor head vent which took place on April 22, 2001. The inspector interviewed licensee personnel and reviewed records of surveys, dosimetry results, determinations of internal uptakes from whole body counts, air sample results and the licensee's event reports.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (7112102)

a. Inspection Scope

The inspector reviewed work performance during the refueling outage (RFO13). The inspector evaluated the licensee's use of engineering controls to achieve dose reductions; determined if workers were utilizing the low dose waiting areas and are effective in maintaining their doses ALARA; determined if workers receive appropriate on-the-job supervision to ensure the ALARA requirements were met; and reviewed individual exposures of selected work groups.

The inspector observed radiation worker and RP technician performance during high dose rate or high exposure jobs and determined whether workers demonstrated the ALARA philosophy in practice. The inspector observed radiation worker performance to determine whether the training/skill level was sufficient with respect to the radiological hazards and the work involved.

The inspector reviewed ALARA job evaluations, exposure estimates and exposure mitigation requirements and ALARA plans were compared with the results achieved. A review of the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and generated shielding requests and their effectiveness to dose rate reduction was also conducted

A review of actual exposure results versus initial exposure estimates was conducted, including comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness and exposure report distribution to support control of collective exposures to determine compliance with the requirements contained in 10 CFR 20.1101(b).

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity, including portable field survey instruments, friskers, portal monitors and small article monitors. The inspector conducted a review of instruments observed during the refueling outage, specifically verification of proper function and certification of appropriate source checks for these instruments which are utilized to ensure that occupational exposures are maintained in accordance with 10 CFR 20.1201.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 Section 1R22 describes the circumstances and licensee actions regarding a loss of reactor water level control and manual reactor scram prior to attaining criticality that occurred May 16, 2001.

.2 Section 4OA2 describes the circumstances concerning the actuation of several reactor protection and engineered safety feature system functions during a normal plant cooldown on April 21, 2001.

.3 April 21, 2001, Reactor Vessel Water Level Instrumentation Spiking and Group I Isolation Event

a. Inspection Scope

The inspector reviewed Pilgrims' assessment and root cause determination of reactor vessel water level instrumentation spiking and a Group I (main steam isolation valve) isolation event that occurred during a plant cooldown on April 21, 2001. The event was documented in PR 01.9385. The inspection included review of: (1) NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," (2) plant design change (PDC) 93-24, "Reactor Water Level Reference Leg Back Fill System," and (3) operating and surveillance procedures, and interviews with licensee operators and engineers.

Pilgrim's response to NRC Bulletin 93-03 was inspected. The inspectors evaluated PDC 93-24, under which Pilgrim installed a reactor vessel water level instrument reference leg back fill system in 1993. Some of the provisions of the PDC were not implemented, and the system was not adequately maintained and operated over time.

Background

Reactor vessel water level instrument spiking occurs when noncondensable gases in the level transmitter reference legs come out of solution as the reactor is depressurized. The gas bubbles momentarily displace water in the reference legs and cause indicated level to be higher than actual level. The amount of gas that is present in the reference legs and the configuration of the piping determine the shape of the indicated level; i.e. spikes or square wave. The condition was identified at Pilgrim in 1992. NRC Bulletin 93-03 required licensees to install hardware modifications to ensure the long-term reliability of reactor vessel water level instruments.

During a normal plant cooldown and depressurization on April 21, 2001, reactor vessel water level instrument spiking occurred which caused automatic closure of the main steam isolation valves (separating the primary system from its normal heat sink), and about one hour later, a reactor scram (control rods already fully inserted) and reactor water cleanup and reactor building ventilation system isolations. Reactor vessel water level indication is an important accident mitigation system in that it provides emergency core cooling system (ECCS) initiation signals and information to operators regarding the adequacy of core cooling. The inspectors evaluated a Pilgrim determination that the level instrument spiking on April 21, 2001 was caused by inadequate venting of the CRD charging header following maintenance on the "B" CRD pump in February-March 2001. This allowed air to be charged through the back fill system into the reference legs. The CRD pump suction piping arrangement prevents adequate static venting of portions of the system.

Small leaks in mechanical joints and valve packing allow gas-saturated water in the reference leg condensing chambers to migrate down the reference legs. The back fill system maintains a sufficient influx of water (nominally 0.008 gallons per minute) up the reference legs to prevent the downward flow of the gas-saturated water. The back fill panel receives a common water supply from the CRD charging header. Backfill system flow does not purge gas-saturated water from the reference legs - they must be free of noncondensable gasses prior to plant startup. This is accomplished by manual backfill of the reference legs using a booster pump and the demineralized water system.

Plant Design Change, PDC 93-24, supporting Pilgrim procedures and records, and Pilgrim PR root cause documentation were reviewed by the inspector. Included in the inspector's review were the following Pilgrim root cause evaluation "related findings."

- Each refueling outage, operation of the back fill system shall be terminated a minimum of two days, and reactor vessel level indication shall be surveyed for mismatches on redundant instrument channels. This tests sensing instrument equalizing valves for leakage. (PDC 93-24) This provision was not scheduled in a preventive maintenance program and was not performed.

- Flowmeters must be calibrated each refueling outage. (PDC 93-24) The periodicity was recommended by the instrument vendor. This preventive maintenance activity was deferred by Pilgrim and calibrations were not performed 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 of the reference legs was not performed since June 1995.

- Procedure guidance regarding control rod drive (CRD) charging header venting following maintenance was inadequate in that no action steps exist in procedure 2.2.87, "Control Rod Drive System." Problem report 95.9316 identified a similar problem in 1995 when a false low level scram signal was received after placing certain level instruments back onto the back fill system. The PR erroneously concluded that existing procedure guidance was adequate.

b. Findings

The inspector independently and in parallel with Pilgrim, identified those items characterized by the licensee as "related findings." The inspector concluded that Pilgrim did not effectively implement corrective actions to preclude recurrence of reactor vessel water level instrument spiking. Between June 1993 and April 21, 2001, Pilgrim did not implement effective corrective actions to preclude recurrence of a significant condition adverse to quality in that certain maintenance and operating provisions of the reference leg back fill system modification were not performed. As a result, the reliability of safety-related reactor vessel water level instruments was degraded and several automatic ECCS and reactor protection system actuation signals occurred that complicated a normal plant cooldown.

The reactor vessel water level instrument spiking condition has a credible impact on safety in that false level indication during severe transients or a design basis accident (large break loss of coolant accident) can cause isolation of the preferred heat sink or potentially lead operators to shut off ECCS equipment erroneously. The condition therefore could adversely affect the availability or reliability of core decay heat removal systems. The inspector and an NRC Region I senior reactor analyst evaluated this condition using Phase 1 of the NRC's Significance Determination Process (SDP). Under mitigation systems, the condition was not a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18. Although automatic trip functions were unaffected, the system's indication function was impaired. Since Phase 2 of the SDP does not readily apply to the condition, the safety significance of the condition is to be determined (TBD) by a Phase 3 SDP evaluation.

This failure to implement corrective actions effectively is an apparent violation of Criterion XVI of 10 CFR, Appendix B. Since the NRC's SDP evaluation is ongoing, the safety significance of the apparent violation is to be determined (TBD), and this issue is unresolved. This issue is documented in Pilgrim's corrective action program as PR 01.9385. (**URI 50-293/01-03-04**)

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. R. Bellamy, VP Operations, and other members of licensee management at the conclusion of the inspection on June 4, 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.

4OA7 Licensee Identified Violation

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-293/01-03-05	Pilgrim Technical Specifications 5.4.1 ” requires written procedures be implemented that meet the requirements of Appendix “A” of Regulatory Guide 1.33, which include maintenance procedures or instructions. Entergy procedure 8.7.1.5, “Local Leak Rate Testing of Primary Containment Penetrations, Isolation Valves, and Inspection of Containment Structure,” Attachment 60, step 2.2 requires that a pretest lineup be performed and step 2.4 requires that vent valves outside the pressurization boundary be open prior to the performance of the local leak rate test. On May 16, 2001, the test director failed to establish an appropriate pretest lineup and ensure that an adequate leak off path existed as required during testing of reactor recirculation sample valve AO-220-45. Test boundary valve 2-HO-134 was left open and vent valve GSV-8029 was not closed. This issue is documented in the licensee’s corrective action program as PRs 01.9485 and 01.9486. This is being treated as a Non-Cited Violation.

ATTACHMENT 1
SUPPLEMENTAL INFORMATION

a. Key Points of Contact

W. Lobo, Licensing Engineer
E. Solomon, Senior Emergency Planner
T. Sowdon, EP Superintendent
K. Sullivan, EP Planning Coordinator
G. Vazquez, Emergency Readiness Coordinator
C. Wend, Radiation Protection Manager

b. List of Items Opened, Closed, and Discussed

URI 50-293/01-03-01	ANS Data in Need of Revision
URI 50-293/01-03-02	Adequacy of Siren Testing Criteria Not Described in the Emergency Plan
NCV 50-293/01-03-03	ERO Respirator Qualification Lapse
URI 50-293/01-03-04	Ineffective Corrective Action for Reactor Vessel Level Spiking
NCV 50-293/01-03-05	Failure to Establish Line Up

c. List of Baseline Inspections Performed

71111-01	Adverse Weather Protection
11111-05	Fire Protection
71111-08	Inservice Inspection (ISI) Activities
71111-12	Maintenance Rule Implementation
71111-14	Personnel Performance During Non-routine Plant Evolutions
71111-19	Post-Maintenance Testing
71111-20	Refueling and Outage Activities
71111-20	Surveillance Testing
71111-23	Temporary Plant Modifications
71114-02	Alert and Notification System Testing
71114-03	Emergency Response Organization Augmentation Testing
71114-04	Emergency Action Level and Emergency Plan Changes
71114-05	Corrections of Emergency Preparedness Weaknesses and Deficiencies
7112101	Access Control
7112102	ALARA Planning and Controls
7112103	Radiation Monitoring Instrumentation

d. List of Acronyms Used

ADS	Automatic Depressurization System
ALARA	As Low As is Reasonably Achievable
ANS	Alert and Notification System
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CANS	Computerized Automated Notification System
CFR	Code of Federal Regulations
CRD	Control Rod Drive
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ERO	Emergency Response Organization
FEMA	Federal Emergency Management Agency
HPCI	High Pressure Coolant Injection
ISI	Inservice Inspection
IVVI	Invessel Visual Inspection
MR	Maintenance Request
MT	Magnetic Particle
NCV	Non-cited Violation
NDE	Nondestructive Examination
PANS	Prompt Alert and Notification System
PDC	Plant Design Change
PI	Performance Indicator
PR	Problem Report
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permits
SDP	Significance Determination Process
TBD	To Be Determined
TS	Technical Specification
UT	Ultrasonic Testing