March 29, 2000

EA 00-020

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. 05000293/2000-001

Dear Mr. Bellamy:

This refers to the inspection conducted on January 10, 2000, through February 20, 2000, at the Pilgrim Nuclear Power facility. The enclosed report presents the results of this inspection.

During the six weeks covered by this inspection period, the conduct of activities at the Pilgrim facility was characterized by safe plant operation. This was especially evident during the planned maintenance activity to replace the master reactor vessel water level controller at power. Good planning, oversight, and mock-up testing were observed.

Your occupational radiation safety program, including your ALARA program, was inspected by a Region I health physicist during this period. The programs were effectively established, implemented, and maintained in accordance with regulatory requirements.

Based on the results of this inspection, the NRC has determined that three severity level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with the NRC Enforcement Policy, dated November 9, 1999. The first NCV involved the failure to test the high pressure coolant injection system as described in technical specifications. The second NCV involved the location of two containment isolation valves that were installed in 1999 in the safety relief valve accumulators alternate nitrogen supply line. The written 10 CFR 50.59 safety evaluation did not provide adequate basis to determine that the design changes did not involve an unreviewed safety question. The third NCV involved the failure to update the technical specifications following the implementation of a plant modification in the standby gas treatment system in 1987. A detailed description for each of these violations is included in the attached report. If you contest the violations or severity level of the NCVs, 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 1, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Clifford J. Anderson, Chief
Projects Branch 5
Division of Reactor Projects

Docket No.: 05000293
License No.: DPR-35

Enclosure: Inspection Report 05000293/2000-001

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

License No.:	DPR-35
Report No.:	05000293/2000-001
Docket No.:	05000293
Licensee:	Entergy Nuclear Generation Company Pilgrim Nuclear Power Station 600 Rocky Hill Road Plymouth, MA 02360
Facility:	Pilgrim Nuclear Power Station
Inspection Period:	January 10, 2000, through February 20, 2000
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector L. Cheung, Senior Reactor Inspector G. Cranston, Reactor Inspector K. Kolaczyk, Reactor Inspector J. Noggle, Sr. Health Physicist
Approved by:	C. Anderson, Chief Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Pilgrim Nuclear Power Station
NRC Inspection Report 05000293/2000-001

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers resident inspection for the period of January 10, 2000, through February 20, 2000; and includes the results of announced inspections in the engineering area during January 10-14 ,2000, and a radiological controls inspection during January 31 through February 4, 2000.

Operations

- As a result of increased management focus on improving operator shift briefings, the licensee recently requested that the various departments attend the 7:30 briefs. Personnel from chemistry, health physics, operations support and work control have recently begun to attend and participate in the shift briefings. (Section O1.1)

- Initial event follow-up was good for two operational events during this period, an inadvertent trip of the spent fuel pool cooling pump and the removal of the wrong fuses for a tagout. Problem reports and event reviews were initiated for both events. (Section O1.1)

- An operator on reactor building rounds alertly identified a steam leak on the outside containment isolation valve, AO-220-45. This valve is in the reactor water clean-up heat exchanger room, which is a locked high radiation area. The leak was promptly isolated by closing both isolation valves in that line. (Section O1.1)

- Operators responded effectively to a malfunction in the master reactor vessel water level controller which caused a water level transient. Operators quickly took manual control to restore water level back to the normal operating band. (Section O2.1)

Maintenance

- Good use of mock-up training for the replacement of the master reactor vessel water level controller potentially prevented an inadvertent reactor scram. (Section O2.1)

- Good procedure adherence was displayed during observed maintenance and surveillance activities. The activities observed and reviewed were performed safely and in accordance with approved procedures. The maintenance supervisor frequently visited the job site and was cognizant of equipment status for work on the secondary containment actuators. (Section M1.1)

- Good preplanning was displayed by operations for surveillance testing as evidenced by their review of electrical schematics for the “B” core spray system automatic pump start logic test. (Section M1.1)

Executive Summary (cont'd)

- The training of electrical maintenance department personnel on troubleshooting techniques met the department's needs and was improved with the recent incorporation of a training simulator. (Section M5.1)

- The inspector identified that the licensee used an alternate test acceptance criteria to meet the intent of a technical specification (TS) surveillance for testing the high pressure coolant injection (HPCI) system. Although the surveillance may be technically equivalent, changes to the TS require prior NRC review and approval. The failure to test the HPCI system as described in TS is considered a violation of TS 4.5.C.1.d. This severity level IV violation of NRC requirements is being treated as a Non-Cited Violation (NCV) in accordance with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 98.9636. (NCV 05000293/2000-01-01) (Section M8.1)

<u>Engineering</u>

- With one exception, safety evaluations (SE) prepared by the licensee conformed to the requirements of 10 CFR 50.59. The licensee resolved all compliance and safety issues pertinent to the associated design changes. The SEs properly described and evaluated the change and justified the bases for the licensee's conclusions. Licensee personnel preparing SEs were trained and qualified for the task. However, one violation involving design basis noncompliance was identified. This severity level IV violation is being treated as a Non-cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 00.0130. (NCV 05000293/2000-01-02) (Section E1.1)

- With one exception, the plant design changes and field revision notices (FRNs) were adequately designed and properly implemented and the issues associated with design changes were adequately resolved. The affected sections of the Updated Final Safety Analysis Report and the associated procedures and drawings were appropriately updated to maintain proper configuration control. The FRNs were properly screened to determine that no safety evacuations were required. (Section E1.2)

- Temporary modifications were installed in accordance with station procedures. The number of bypasses installed was consistent with similar plants in the industry. No long standing temporary modifications were observed. (Section E1.3)

- With one exception, the reviewed engineering service requests and the resultant engineering response memorandums (ERMs) were adequately dispositioned. The exception involved an ERM that revised the output of the heaters in the standby gas treatment system. Although the requisite surveillance procedure was revised to reflect the new value, the licensee failed to update the appropriate section of the technical specifications (TS), which contained the non-conservative value since 1987. This condition constituted a violation of 10 CFR 50.36(b). This severity level IV violation is being treated as a Non-cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 00.0141. (NCV 05000293/2000-01-03) (Section 1.4)

- The licensee provided a dedicated team to work on the design basis upgrade program. Reasonable progress had been made to meet the committed completion date of December 2001. The reviewed system report and topical report were well prepared and adequately validated. The contents were logically arranged, making design basis information retrieval relatively easy. The reviewed instrument set point calculations were technically sound. (Section E2.1)

- The licensee is trending the engineering backlogs and has substantially reduced the backlog of some engineering activities, such as problem reports and operating experience evaluations. Engineering backlogs at Pilgrim are being adequately managed. (Section E2.2)

- The licensee's corrective actions for two open items associated with the motor-operated valve program were found acceptable, and the items were closed (Sections E8.1 and E8.2)

Plant Support

- Radiological postings were adequate and very high radiation area controls were effectively implemented during a Low Pressure Coolant Injection (LPCI) logic test conducted in the Transverse In-core Probe (TIP) room. (Section R1.1)

- During 1999, internal exposures were minimal with the maximum individual recorded at 1.2% of regulatory limits. Internal dose assessments reviewed were accurately calculated. (Section R1.2)

- Pilgrim Station recent exposure history has been improving. The largest ALARA challenge is the high radiological source-term at the station. Two major piping systems (recirculation and RHR) have been chemically decontaminated and the same piping systems have been partially shielded with permanent shielding. Additional plans for RFO 13 include fully shielding the recirculation piping system and chemical decontamination of the reactor water cleanup and fuel pool cooling piping systems. Based on the source-term reduction activities conducted so far, and those being planned, the ALARA program has been found to be effective. (Section R1.3)

- Few radiation protection problems occurred during 1999. The radiation protection department produced many self-assessments for 1999 resulting in a few recommendations. The QA surveillance program provided a thorough review of the radiation protection program and resulted in identifying several deficiencies. Oversight of the radiation protection program was effective. (Section R7)

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ATTACHMENTS

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the period operating at 100% reactor power. On January 11, operators lowered reactor power to approximately 78% core thermal power to perform a control rod pattern change. The unit was returned to 100% power, where it operated during the remainder of the inspection period.

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Using procedure 71707, the inspector conducted frequent reviews of ongoing plant operations. The inspector observed proper control room staffing, effective briefings, and expected plant response for the plant configuration and plant activities in progress. The control room shift morning briefings conducted by the control room supervisor (CRS) were informative and covered the planned activities and work assignments. The inspector noted the briefings were more structured and formal as a result of increased management focus. Personnel from chemistry, health physics, operations support and work control regularly attended and actively participated in the shift briefings.

Operations personnel initiated problem reports to document, evaluate and correct identified equipment and human performance problems. Two events occurred this period during system restorations from planned maintenance activities. In the first instance, incorrect electrical fuses were removed and second checked as part of a tagout. This was self-identified by the licensee during the tagout restoration process and PR 00.0136 was initiated. The licensee initiated an apparent cause review to examine the circumstances and human performance aspects of this event. The licensee found that poor labeling in the instrument rack was a contributing factor in this event.

The second operational event involved a low level in the skimmer surge tank that caused the operating fuel pool cooling pump to automatically trip. This occurred while returning the fuel cooling system into operation after replacing the resin in the demineralizer. Subsequently, operators found a boundary valve partially open which diverted flow to the spent resin tank. A critique convened by operations management found that the maintenance staff had inadvertently dropped a wrench on the boundary valve operating stem causing it to partially open. The inspector had no questions or additional concerns on these two operational events.

During the reactor building rounds, an operator identified a packing leak on the outside containment isolation valve AO-220-45. This valve is located in a 1 inch sample line that connects to the “B” recirculation loop piping. This was alertly found during the normal

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

surveillance of the reactor water clean-up heat exchanger room. This room is a locked high radiation area. Operators closed AO-220-45 and AO-220-44, inside containment isolation valve, which stopped the leakage.

O2 Operational Status of Facilities and Equipment

O2.1 Reactor Vessel Water Level Transient

a. Inspection Scope (71707,93702)

The inspector reviewed operator performance during a reactor vessel water level transient that was induced by a malfunction of the feed water system master level controller.

b. Observations and Findings

A high reactor vessel water level alarm alerted control room operators to a problem with reactor vessel water level. Operators followed the alarm response procedure and placed the master controller into manual and restored water level back to the normal band. This occurred late at night and the operations shift superintendent (OSS) contacted an engineer for technical assistance. The licensee decided to return the master controller to automatic and monitor its performance.

A similar problem was again experienced with the master reactor vessel water level controller on the following day shift. A decision was made to replace the master controller at power after performing a work mock-up on the plant simulator. The mock-up was performed due to the potential effect this maintenance activity could have on the plant. The master controller was replaced and operators placed the controller into service with no problems; a simulator scram was experienced during the mock-up. No problems or concerns were identified by the inspector during this review.

c. Conclusions

Operators responded effectively to a malfunction in the master reactor vessel water level controller which caused a water level transient. Operators quickly took manual control to restore water level back to the normal operating band. Good use of mock-up training for the replacement of the master reactor vessel water level controller potentially prevented an inadvertent reactor scram.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Maintenance

a. Inspection Scope (61726,62707)

The inspector observed portions of selected maintenance and surveillance activities to verify that the applicable procedures and technical specifications were satisfied.

b. Observations and Findings

The inspector observed all or portions of the following activities:

- 8.M.2-2.10.1-7, “Automatic Start Core Spray Pump P-215B Logic System Functional Test”
- 8.M.2-2.10.3-2, “RHR LPCI Containment Spray Subsystem B Logic Test”
- P9501069, Remove/Rebuild Secondary Containment Damper Actuators AO-N-90 and -91

c. Conclusions

Good procedure adherence was displayed during observed maintenance and surveillance activities. The activities observed and reviewed were performed safely and in accordance with approved procedures. Good preplanning was displayed by operations for surveillance testing as evidenced by their review of electrical schematics for the “B” core spray system automatic pump start logic test. The maintenance supervisor frequently visited the job site and was cognizant of equipment status for work on the secondary containment actuators.

M5 Maintenance Staff Training and Qualification

M5.1 Maintenance Training

a. Inspection Scope (61726, 62707)

The inspector attended a portion of the electrical maintenance training for basic troubleshooting techniques to determine training and qualification effectiveness. This included training in both the classroom and simulator.

b. Observations and Findings

The training was provided to electrical maintenance department personnel at the industrial park training center. The training consisted of a five-day course and included both classroom and practical experience on the maintenance simulator system. The licensee had recently built a closed loop flow system (simulator) to assist in the training of craft personnel. The system consists of an electrical system, pump, air and motor operated valves, instrumentation, and associated piping. The simulator is used for various training scenarios and can be configured to provide training for electrical, mechanical, and instrumentation and control disciplines.

The simulator allowed the training instructor to install various system faults that the electrical maintenance mechanics had to diagnose. In addition to identifying the faults, the students demonstrated proper electrical safety precautions and use of the licensee's troubleshooting procedures. Discussion with craft personnel indicated that the training was appropriate and meeting their needs, and was enhanced with the implementation of the training simulator.

A review of the Pilgrim Nuclear Power Station Accrediting Board Report did not disclose any safety significant issues with regards to the maintenance training program.

c. Conclusions

The training of electrical maintenance department personnel on troubleshooting techniques met the department's needs and was improved with the recent incorporation of a training simulator.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) URI 50-293/98-11-01; High Pressure Coolant Injection (HPCI) Testing

NRC Inspection Report No. 50-293/98-11, date February 19, 1999, Section O2.1, documented that the acceptance criteria for the HPCI pump flow test at 150 psig was not consistent with the value specified in technical specification 4.5.C.1.d. Technical specifications (TS) state that the licensee verify that the HPCI pump deliver at least 4250 gpm for a system head corresponding to a reactor pressure of 150 psig, whereas the licensee's surveillance procedure specified a flow of 2000 gpm with a pump discharge pressure of 280 psig. The use of an alternate test acceptance criteria that meets the intent of TS surveillance, although technically equivalent, requires prior NRC review and approval. The licensee generated problem report 98.9636 to address this concern. The licensee reviewed other pump tests and did not identify other surveillance procedures that used an alternate test criteria. A TS amendment was submitted to the NRC on May 11, 1999, to reflect the alternate test criteria. The failure to test the HPCI system as described in TS is considered a violation of TS 4.5.C.1.d. This severity level IV violation of NRC requirements is being treated as a non-cited violation (NCV) in accordance with Section VII.B.1.a of the NRC Enforcement Policy (**NCV 05000293/2000-01-01**).

III. ENGINEERING

E1 Conduct of Engineering

E1.1 10 CFR 50.59 Safety Evaluation Review

a. Inspection Scope (37001)

The inspector reviewed a sample of eleven 10 CFR 50.59 safety evaluations (SE) associated with plant design changes at Pilgrim to determine whether the SEs were completed in accordance with 10 CFR 50.59. The inspector also reviewed the licensee's training and qualification program for SE preparation to determine its adequacy.

b. Observations and Findings

The inspector reviewed eleven SEs and observed that, with one exception, the SEs conformed to the requirements of 10 CFR 50.59, "Changes, Test and Experiments." The licensee resolved all compliance and safety issues pertinent to the associated design changes. The SEs properly described and evaluated the design changes and justified the bases for the licensee's conclusions.

Additionally, the inspector interviewed five engineers who prepared or approved the sampled SEs and found them to be knowledgeable, experienced and qualified to prepare SEs. Based on discussions with the technical training personnel, the inspector found that an engineer was required to complete a training and qualification program before the individual was allowed to prepare SEs independently. The inspector reviewed the SE training and qualification program, which included periodic retraining, and found the program acceptable.

In reviewing SE 3185 on Plant Design Change (PDC) 96-19, the inspector found two examples where the SE did not adequately address the design changes that deviated from the criteria specified in the Pilgrim Updated Final Safety Analysis Report (UFSAR). PDC 96-19 was installed in 1999 to provide seismically qualified nitrogen (N2) makeup piping from the exterior of the drywell to the safety relief valve accumulators inside the drywell, using part of existing piping and tubing associated with existing drywell penetration X-46E. The design change was necessary to provide N2 makeup to the accumulators following a postulated seismic event. The two deviations were:

1. The newly installed, normally closed, manual primary containment isolation valves were evaluated and specified in the SE, and listed in Table 5.2-4 of the UFSAR as Class B valves. Per the UFSAR, Class B valves must have remote-operation capability from the control room. However, these valves did not have this capability. The SE did not address this issue and no justification was provided. The new valves are normally closed, and would only be open for a short time to recharge the safety relief valve accumulators following a seismic event if shutdown cooling is unavailable. This backup nitrogen supply is not intended to mitigate the consequences of a design basis accident. The licensee

explained that these containment isolation valves were mis-classified and should have been classified as Class B-X, which was for instrument tubing applications and does not require remote-operation capability. The licensee issued Problem Report (PR) 00.0131 to initiate the UFSAR change for this discrepancy. The inspector determined that this design control violation of minor significance was not subject to formal enforcement action.

2. The Pilgrim UFSAR states that, when both containment isolation valves are outside the drywell, the first isolation valve must be as close to the containment as practical. The previously existing first containment isolation valve on this line was outside the drywell and was removed in conjunction with this modification. The existing tubing was lengthened and the new first containment isolation valve was installed at a greater distance from the drywell. No basis was given in the SE that the previous location was not practical for this modification or that the new valve needed to be installed farther from the drywell. The inspector observed that, for a loss of coolant accident (LOCA), the nitrogen piping and tubing inside the drywell, up to and including the safety relief valve accumulators, is considered a closed system per the design and licensing bases. No credit was taken for that portion of the system as a post LOCA containment isolation barrier in the design and licensing basis of the plant. The inspector considers the safety significance of this issue to be very low.

The licensee wrote PR 00.0130 to document the discrepancies and provided, for the inspector's review, a preliminary revisions to Section 5.2. of the UFSAR to provide clarification regarding the new location of the new primary containment isolation valve in UFSAR Section 5.2.3.5.1.

The inspector determined that the above design basis noncompliance constitutes a violation of 10 CFR 50.59(b)(1), in that the written SE did not provide an adequate basis for the determination that the design change does not involve an unreviewed safety question. The violation of 10 CFR50.59 is considered a violation of NRC requirements. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy **(NCV 05000293/2000-01-02), EA 00-020**.

c. Conclusions

With one exception, safety evaluations prepared by the licensee conformed to the requirements of 10 CFR 50.59. The licensee resolved all compliance and safety issues pertinent to the associated design changes. The SEs properly described and evaluated the change and justified the bases for the licensee's conclusions. Licensee personnel preparing SEs were trained and qualified for the task. However, one violation involving design basis noncompliance was identified. This severity level IV violation is being treated as a Non-cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

E1.2 Plant Design Changes Review

a. Inspection Scope (37550)

The inspector reviewed five plant design changes (PDC) and three Field Revision Notices (FRN) to determine whether these PDCs and FRNs (minor modifications to the plant that did not require a 10 CFR 50.59 safety evaluation) were adequately designed and properly implemented and that the issues associated with design changes were adequately resolved. Plant risk insights information was used to select PDCs associated with the higher risk systems.

b. Observations and Findings

The following PDCs and FRNs were reviewed:

1) PDC 96-16, which was issued to provide seismically qualified N2 makeup piping from the exterior of primary containment drywell to safety relief valve accumulators inside the drywell for long term pressure control. Two issues associated with the 10 CFR 50.59 safety evaluation of the PDC were identified resulting in a non-cited violation, as discussed in Section E1.1 of this report.

2) PDC 97-26, Load Shedding System Changes for Regulatory Compliance, was used to provide safety related control cables and circuitry to replace the existing non-safety related cables for the load-shed circuits, and to rewire the load-shed contacts as the last element before the starter holding coils in selected control circuits.

3) PDC 99-12, Emergency Diesel Generator (EDG) Ventilation and Radiator Fan Modifications, was prepared to correct problems associated with high diesel engine process temperatures. Previous plant design changes (PDCR 77-79 and PDCR 87-55) did not correct the problem. The licensee identified the deficiencies in the previous design changes and other design deficiencies that required correction to ensure EDG operation in accordance with the plant's design and licensing basis.

4) PDC 99-18, Installation of Isolation Dampers in Main Control Room Environmental Control System (MCRECS) Supply and Exhaust Ducts, was prepared to install redundant, safety related isolation dampers to correct the deficiencies associated with the MCRECS in meeting its design and licensing basis due to the failure of non-safety related ventilation dampers.

5) PDC 99-22, Weko Seals on the Seawater Service Water System Discharge Piping, was prepared in conjunction with the repair of deteriorated rubber lining at locations of pipe spool fit-up of the service water piping.

6) FRN 98-01-38, Residual Heat Removal (RHR) Area Coolers Modifications for Unresolved Safety Issue (USI) A-46, was prepared to install limit-stop devices to provide assurance that seismic excitation will not result in unacceptable horizontal or vertical displacements at the support of the RHR area coolers.

7) FRN 99-01-58, Repair and/or Replacement of Degraded Seawater Service Water System Piping, was prepared to replace existing corroded carbon steel pipe spool pieces where excessive wall thinning had occurred and to repair the rubber lining.

8) FRN 99-01-67, Main Steam Isolation Valve Speed Control Valve Replacement, was prepared to replace the existing speed control valves with valves that were designed and qualified for the high temperature environment in which they are required to operate.

The inspector found that, with the exception noted in Section E1.1, the above PDCs and FRNs were adequately designed and properly implemented and the issues associated with design changes were adequately resolved. The inspector also verified that the affected UFSAR sections and the associated procedures and drawings were appropriately updated to maintain proper configuration control. The FRNs were properly screened to determine that a SE was not required.

c. Conclusions

With one exception, the PDCs and FRNs were adequately designed and properly implemented and the issues associated with design changes were adequately resolved. The affected UFSAR sections and the associated procedures and drawings were appropriately updated to maintain proper configuration control. The FRNs were properly screened to determine that no SEs were required.

E1.3 Temporary Modification Program

a. Inspection Scope (37550)

The inspector reviewed the process used to implement the temporary modification (bypass) program. The scope included the review of Procedure 1.5.9, "Temporary Modifications," which established controls for installation and removal of bypasses; the review of the control room bypass log; and the physical inspection of temporary modifications installed in the plant.

b. Observations and Findings

There were seventeen temporary modifications installed. This number was consistent with similar plants in the industry that had completed one third of their fuel operating cycle. No long standing temporary modifications were observed. The oldest modification, which involved providing a manual method to provide makeup water to the fuel pool skimmer surge tank, had been installed since February 1998.

None of the bypasses individually or collectively appeared to place an undue burden on the operations staff. Most of the bypasses were installed on equipment that was not risk-significant or on non safety-related systems. Plant drawings had been updated to reflect the installation of the bypasses, and plans were in place to resolve the conditions that necessitated installation of the bypasses.

The inspector examined several bypasses in the field, and verified that they had been installed in accordance with Procedure 1.5.9. The licensee was actively trying to reduce the number of outstanding bypasses, and had several modifications in the final design phase scheduled for installation during the second quarter of 2000 that would eliminate four bypasses.

c. Conclusions

Temporary modifications were installed in accordance with station procedures. The number of bypasses installed was consistent with similar plants in the industry. No long standing temporary modifications were observed.

E1.4 Engineering Service Request and Engineering Response Memorandum Review

a. Inspection Scope (37550)

The inspectors reviewed a sample of Engineering Service Requests (ESRs), and Engineering Response Memorandums (ERMs) to determine if the dispositions contained in the ERMs adequately resolved the ESR issues, and the corrective actions were implemented appropriately.

b. Observations and Findings

Background

Nuclear Operations Procedure (NOP) 84E1, “Engineering Service Request Process,” described the process used to prepare ESRs and ERMs. NOP 84E1 indicated that an ESR should be processed when engineering support was needed to address plant issues such as the following:

- Procuring spare or replacement parts
- Clarifying or creating repair/refurbishment procedures
- Clarifying or Interpreting design basis
- Improving plant performance
- Preparing temporary modifications
- Preparing design changes to retire temporary modifications
- Providing technical support for regulatory responses.

ERMs are the responses to ESRs. The priority for dispositioning ESRs and preparing the requisite ERMs is based upon a matrix contained in NOP 84E1. ESRs that concerned technical specifications (TS) equipment are assigned the highest priority. Likewise, ESRs that do not effect equipment operability or safety-related components are assigned a low priority for action.

ESR/ERM Review

The inspectors reviewed approximately 50 ESRs and the resultant ERMs that were written since 1988. Although the majority were dispositioned appropriately, the inspectors did find one example where the licensee failed to update the TS to reflect the results of the ESR disposition.

The issue involved ERM 92-0103, which indicates that the heaters in the standby gas treatment (SBGT) system should generate at least 16.1 kW of heat to dehumidify the air to 70% relative humidity (RH) before entering the charcoal filters. Although Pilgrim had updated the requisite surveillance procedure 8.7.2.2, “Demonstration of Standby Gas Treatment Inlet Heater Capability,” to reflect the new value, the corresponding plant TS was not updated. Specifically TS Section 4.7.B, “Standby Gas Treatment System and Control Room High Efficiency Air Filtration System,” which outlines the surveillance requirements for the SBGT system, still indicates a minimum heat output of at least 14 kW as acceptable, which is non-conservative. The inspector’s finding was entered into the Pilgrim corrective action program as Problem Report (PR) 00.0141.

The heater output was changed as a result of a 1987 modification to the SBGT system. The modification involved changes to Section 5.3.3.4, Standby Gas Treatment System, and Section 7.18, Reactor Building Isolation and Control System, of Pilgrim UFSAR. Apparently when the modification was installed, TS Section 4.7.B was not revised.

The above condition constitutes a violation 10 CFR 50.36(b), which requires the technical specifications be derived from the analyses and evaluation included in the safety evaluation report. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the Enforcement Policy **(NCV 050000293/2000-01-03)**.

c. Conclusions

With one exception, the reviewed ESRs and the resultant ERMs were adequately dispositioned. The exception involved an ERM that revised the output of the heaters in the SBGT system. Although the requisite surveillance procedure was revised to reflect the new value, the licensee failed to update the appropriate section of the technical specifications, which contained the non-conservative value since 1987. This condition constitutes a violation of 10 CFR 50.36(b). This severity level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy.

E2 Engineering Support of Facilities and Equipment

E2.1 Licensing/Design-Basis Upgrade Program

a. Inspection Scope (37550)

In October 1996, the NRC issued a letter to each licensee, in accordance with 10 CFR 50.54 (f), requesting the licensee to provide information for ensuring the facility operation was maintained within the design bases. The licensee responded in their February 10, June 24, and December 8, 1997, letters describing the additional design basis documentation they committed to develop, and the completion date of the program. The inspector reviewed the activities associated with this program to determine the licensee's progress in this area. The inspector also reviewed completed design basis documents to assess their quality.

b. Observations and Findings

The licensee organized a dedicated team, headed by the Project Manager of the Design Basis Information (DBI) Program, to complete the design basis upgrade effort. The team has about 30 part-time and full-time personnel, consisting of consulting specialists, General Electric Company personnel, and Entergy engineers. In the December 8, 1997, response letter to the NRC, the licensee committed to complete the DBI program before the fourth quarter of 2001. There are two major types of design basis documents generated by the DBI team, the system reports and the topical reports. The licensee estimated that there would be about 24 system reports and 30 topical reports. These numbers could change as the program progresses. At the time of the inspection, eight system reports and 12 topical reports were completed. The inspector selected one system report, Residual Heat Removal (RHR) System, dated July 6, 1999, and one topical report, Instrument Uncertainty, dated August 6, 1999, for review. The inspector noted that the reports were well prepared, and adequately validated. The contents were logically arranged, making design basis information retrieval relatively easy. In addition, the inspector reviewed two calculations, No. IN1-214, Uncertainty Calculation for RHR Flow Indicators FI-1040-1A & B, dated November 30, 1999, and No. I-N1-199, Set Point Calculation for LPCI Minimum Flow Bypass Line Valve Control, DPIS 1001-79A & B. The inspector found these calculations technically sound.

c. Conclusion

The licensee provided a dedicated team to work on the design basis upgrade program. Reasonable progress had been made to meet the committed completion date. The reviewed system report and topical report were well prepared, and adequately validated. The contents were logically arranged, making design basis information retrieval relatively easy. The reviewed instrument set point calculations were technically sound.

E2.2 Engineering Backlogs

a. Inspection Scope (37550)

The inspector review the status and trend of the licensee's backlog of engineering activities, including open modifications, operation experience evaluation, vendor manual evaluations, root cause evaluations, and overdue problem reports, to determine whether the licensee has maintained adequate control of the engineering backlogs.

b. Observations and Findings

The licensee did not have trending data for the backlog of permanent modifications and resolutions of issues associated with plant maintenance requests. The licensee had trending data for open problem reports (PR), temporary modifications (TM), open vendor manual (VM) evaluations, open operating experience (OE) evaluations and open regulatory commitments (RC). The licensee has shown progress in reducing the backlogs in each of the activities that they have trending data, although the number of TM was reduced only slightly from 20 at the beginning of 1999 to 17 at the beginning of 2000. The total number of open engineering PRs was reduced from 570 in January 1999 to about 330 in December 1999. The numbers of past due PRs (> 6 months) and the past deadline PRs were also reduced from 250 and 65 in January 1999 to 150 and 0 in December 1999. The numbers of open OE evaluations (42 to 17), open VM evaluations (73 to 35) and open RCs (62 to 25) were also substantially reduced during 1999.

c. Conclusion

The licensee is trending the engineering backlogs and has substantially reduced the backlog of some engineering activities, such as problem reports and operating experience evaluations. Engineering backlogs at Pilgrim are being adequately managed.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) IFI 05000293/97-13-03: Valve Factor Assumptions.

This item was opened to document that certain assumptions such as motor-operated valve (MOV) valve factor, which the licensee had used to calculate the performance of (MOV)s at Pilgrim, did not have an adequate basis. To resolve this issue, the licensee reanalyzed the performance of the MOVs using the Electric Power Research Institute (EPRI) Performance Prediction Program (PPM). This program has been reviewed and approved by the NRC for analyzing MOV performance and does not rely upon the unjustified assumptions previously used. During the reanalysis, no valve operability concerns were identified. However, as a result of the reanalysis, the licensee decided to modify several valves to increase their design margins. The inspectors reviewed the reanalyses for three MOVs, and verified that the licensee had used system operating pressures that were equivalent to the values specified in the plant Final Safety Analysis Report (FSAR). This item is **closed**.

E8.2 (Closed) IFI 50-293/97-13-04: Recirculation Pump Discharge Isolation Valves

This item was opened to document that the design basis analysis for the recirculation pump discharge loop isolation valves had not been completed. Specifically, Pilgrim had three design basis calculations for the valves that used different valve factors and differential pressures. For example, one analysis was based upon industry data, and assumed a valve factor of 0.5 and a differential pressure of 200 psid. A second analysis relied upon plant specific data, and used a valve factor of 0.75 and a differential pressure of 32 psid. A third analysis used still other input parameters. At the time of the MOV inspection, the licensee had not yet decided which calculation should be used as the analysis of record to establish the valve switch settings and thrust requirements. To resolve the issue, the licensee contracted with the supplier of the isolation valves to research the history of the these valves, and to recommend which calculation approach should be used. Based on their review, the supplier recommended that the MOV design basis analysis should be based upon plant specific data. Subsequently, the licensee reanalyzed the MOV design basis using plant specific data. The analysis concluded that both valves were operable. The inspector reviewed the design calculation for the MOVs and verified that Pilgrim plant specific data were used in the calculation. This item is closed.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 External Exposure Controls

a. Inspection Scope (83750)

Tours were conducted of accessible areas of the plant, and independent surveys were conducted to verify radiological postings in these areas. The inspector accompanied a work crew into a posted very high radiation area to observe work controls during a low pressure coolant injection (LPCI) containment spray logic test.

b. Observations and Findings

Appropriate job briefing and overall control of the LPCI logic test was conducted by operations control room personnel. The RP technician briefing included an existing survey that reflected the as-found radiation fields in the transverse in-core probe (TIP) room. The RP technician was aware of the latest TIP core insertion and neutron exposure period, with the TIPs having decayed significantly and stored in their shield containers. Key control and guarding of the TIP room was appropriately implemented and controlled, with double verification of locked door accomplished after exiting the room. No deficiencies were observed.

Most of the reactor building spaces were maintained as clean areas. The refueling floor and the control rod drive (CRD) rebuild room, and the CRD hydraulic control unit (HCU) banks were areas that had been converted to clean areas that had historically been contaminated. Both of the residual heat removal (RHR) quad rooms continue to be contaminated areas. All accessible areas were free of debris and no unnecessary radiological hazards were observed.

c. Conclusions

Radiological postings were adequate and very high radiation area controls were effectively implemented during a LPCI logic test conducted in the TIP room.

R1.2 Internal Exposure Controls

a. Inspection Scope (83750)

The 1999 air sample records and whole body count records were reviewed as well as selected internal dose assessments completed by the licensee.

b. Observations and Findings

During 1999, the highest air sample recorded was 11 derived air concentration (DAC). Seven individuals were assigned DAC-hrs from air samples with the highest recorded at 5.09 DAC-hrs (approximately 13 mrem) for a combined total for 1999 of 36.6 DAC-hrs (91.5 mrem). From positive whole body counts taken during the year, there were 28 intake assessments and internal dose assignments being performed. The highest was 60 mrem with a combined total for 1999 of 566 mrem. Selected intake and dose assessments were reviewed. The assessments conservatively included contributions from non-gamma emitting radionuclides that are known to exist in the solid radioactive waste streams. No discrepancies were identified.

c. Conclusions

During 1999, internal exposures were minimal with the maximum individual recorded at 1.2% of regulatory limits. Internal dose assessments reviewed were accurately calculated.

R1.3 Implementation of the As Low As is Reasonably Achievable (ALARA) Program

a. Inspection Scope (83728)

Inspection of this area consisted of a review of industry and Pilgrim collective exposure histories and source-term comparisons, previous refueling outage ALARA report, Pilgrim dose reduction strategy and current plans for reducing collective exposures at Pilgrim Station.

b. Observations and Findings

Pilgrim has a 2 year refueling cycle. Therefore, two-year collective exposure averages appropriately characterize Pilgrim Station's occupational exposure performance (averaging the high exposure refueling outage year with a low exposure non-outage year). Recent annual exposure results are provided below.

	Pilgrim	Two-year Average
1996	116 rem	299 rem
1997	588 rem	352 rem
1998	71 rem	330 rem
1999	346 rem	208 rem

Pilgrim Station continues to be challenged by a high source-term. Average recirculation piping contact dose rates during the May 1999 refueling outage averaged 385 mrem/hr as compared to the average BWR at 220 mrem/hr.

In order to achieve and maintain lower occupational exposure results, several source-term reduction/mitigation initiatives have been undertaken. During the 1997 refueling outage, a recirculation piping system chemical decontamination was performed. During the Fall of 1998 both RHR loop piping systems were chemically decontaminated. In addition, during the Spring 1999 refueling outage, approximately one-half of the recirculation piping system was permanently shielded with one-inch equivalent lead thickness.

Revision 3 of the Pilgrim Station Dose Reduction Strategy, dated January 2000, lists several planned initiatives designed to address the high source-term. During the next refueling outage (Spring 2001), there are plans to complete the permanent shielding of the recirculation piping system, and chemically decontaminate both the reactor water cleanup piping and fuel pool cooling piping systems. Control of source-term buildup appears to have been successful with the use of depleted zinc injection into the reactor water during plant operations that was begun in late 1996.

Remaining areas of source-term challenge include: permanent shielding of the RHR piping (both inside the drywell and in the quad rooms), and shielding of the control rod drive scram volume discharge headers.

An additional challenge is controlling outage length and sufficient lead time in outage planning. During the Spring 1999 outage approximately one-third of all scheduled jobs were planned after the 12-month scope freeze resulting in a large number of finished work packages reaching the ALARA planning group only weeks before the beginning of the outage.

c. Conclusions

Pilgrim Station recent exposure history has been improving. The largest ALARA challenge is the high radiological source-term at the station. Two major piping systems (recirculation and RHR) have been chemically decontaminated and the same piping systems have been partially shielded with permanent shielding. Additional plans for RFO 13 include fully shielding the recirculation piping system and chemical decontamination of the reactor water cleanup and fuel pool cooling piping systems. Based on the source-term reduction activities conducted so far, and those being planned, the ALARA program has been found to be effective.

R7 Quality Assurance (QA) in RP&C Activities

a. Inspection Scope (84750-01)

The inspection consisted of a review of selected 1999 problem reports and all RP self-assessments and QA surveillances performed during 1999.

b. Observations and Findings

A review of radiation protection problem reports documented during 1999 indicating very few problems had occurred. Examples included: repeat reactor water cleanup pump seal failures, core spray nozzle weld exam exposures, reliability study of electronic dosimeters, and investigating torus floor tritium activity. The problem reports were adequately characterized and the extent of review was commensurate with the risk.

Radiation protection department self-assessments for 1999 covered selected technical areas that were studied for program enhancements. These included: ALARA performance in RFO 12, ALARA outage planning, shielding effectiveness, contamination monitor sensitivity, and radiation worker practices. These reviews were of variable quality resulting in some recommendations. There were 27 RP QA surveillances conducted during 1999 that represented the principle subject areas that were historically covered by an audit. Areas reviewed included: locked high radiation areas, calibration of high range radiation monitors, shipping surveys, radiation worker practices, contractor training, control rod drive replacements, exposure reports, and contamination control. The surveillances were of good quality and resulted in generating 17 problem reports.

c. Conclusions

Few radiation protection problems occurred during 1999. The radiation protection department produced many self-assessments for 1999 resulting in a few recommendations. The QA surveillance program provided a thorough review of the radiation protection program and resulted in identifying several deficiencies. Oversight of the radiation protection program was effective.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspector met with licensee representatives at the conclusion of the inspection on March 9, 2000. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

X3 Management Meeting Summary

On February 4, 2000, Mr. Clifford Anderson, Branch Chief, visited the site.

ATTACHMENT 1

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 82301:	Evaluation of Exercises for Power Reactors
IP 83728	Maintaining Occupational Exposures ALARA
IP 83750	Occupational Radiation Exposure
IP 84750-01	Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND UPDATED

Open

None

Closed

IFI 05000293/1997013-03	Valve Factor Assumptions
IFI 05000293/1997013-04	Recirculation Pump Discharge Isolation Valves
NCV 05000293/2000001-01	High Pressure Coolant Injection (HPCI) Testing
NCV 05000293/2000001-02	10 CFR 50.59 Safety Evaluation Review
NCV 05000293/2000001-03	Engineering Service Request and Engineering Response Memorandum Review
URI 05000293/1998011-01	High Pressure Coolant Injection (HPCI) Testing

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRHEAF	Control Room High Efficiency Air Filtration
DAC	Derived Air Concentration
DBI	Design Basis Information
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ERM	Engineering Response Memorandums
ESR	Engineering Service Request
FRN	Field Revision Notices
FSAR	Final Safety Analysis Report
HCU	Hydraulic Control Unit
IR	Inspection Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MR	Maintenance Request
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NOP	Nuclear Operations Procedure
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDC	Plant Design Change
PDR	Public Document Room
PNPS	Pilgrim Nuclear Power Station
PPM	Performance Prediction Program
PR	Problem Report
PWT	Post Work Test
QA	Quality Assurance
RFO	Refueling Outage
RHR	Residual Heat Removal
RP	Radiation Protection
RP&C	Radiological Protection and Chemistry
SBGT	Standby Gas Treatment
TIP	Transverse In-core Probe
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VIO	Violation
VM	Vendor Manual