Deember 21, 2001

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - INSPECTION REPORT 50-293/01-07

Dear Mr. Bellamy:

On November 17, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report documents the inspection findings that were discussed on November 29, 2001, with Mr. W. Riggs and other members of your staff.

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

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). The issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRV is treating this issue as non-cited violation, in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report 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. 20555-0001; and the NRC Resident Inspector at the Pilgrim Station.

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

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

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

Sincerely,

/RA/

Clifford Anderson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 50-293/01-07

Attachment: Supplemental Information

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

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	50-293/01-07
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates :	September 30, 2001, through November 17, 2001
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector J. Jang, Senior Health Physicist D. Silk, Senior Emergency Preparedness Inspector N. McNamara, Emergency Preparedness Inspector S. Dennis, Resident Inspector, Oyster Creek A. Della Greca, Senior Reactor Inspector
Approved by:	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-07; on 9/30 - 11/17/2001; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station; Event Follow-up.

The inspection was conducted by resident inspectors, a senior health physicist, two emergency preparedness inspectors, and a reactor inspector. The inspection identified one significant finding. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for failure to assure that the 1993 design changes made to the reference leg of the reactor level instrumentation were adequate and subject to the same control measures applied to the original design.

  This finding, originally identified in Inspection Report 50-293/01-05, was evaluated through the Phase 3 Significance Determination Process and found to be of very low safety significance because although the finding contributed to a loss of automatic emergency core cooling system initiation affecting a mitigation system, (1) the duration of the event was short; (2) diverse instruments were available for the automatic start of reactor inventory makeup systems; (3) redundant vessel level indication was available to the operators throughout the event for manual initiation of protective functions, if needed; and (4) any significant decrease of reactor pressure (usually associated with a decrease in reactor inventory) would have restored the level instruments to operable. Because the finding is of very low safety significance and was captured in the licensee's corrective action program, this finding is being treated as a Non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (Section 4OA3)

B. Licensee Identified Violations

- None.

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Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period at 100 percent core thermal power. On October 5, 2001, power was reduced to 70 percent for a rod pattern exchange. The unit operated at 100 percent power for the remainder of the period.

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

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed the Safety Evaluations (SE) associated with twelve plant changes that were implemented during the previous 18 months. The reviews were performed to verify that changes made to the plant or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The SEs reviewed covered activities associated with three cornerstones: initiating events, mitigating systems, and barrier integrity. The inspectors also reviewed eleven plant changes that the licensee considered to be exempt from the safety evaluation process. This review was performed to verify that the screening process was appropriately implemented.

In addition, the inspectors reviewed administrative procedure NOP83E5, Revision 14, “Safety Reviews,” and Revision 15, “10CFR50.59 Review Process,” to ensure that the guidance contained therein for preparing, screening-out and issuing SEs adequately covered the requirements of 10 CFR 50.59 during the period of review.

The inspectors interviewed, as appropriate, the engineering personnel engaged in the preparation and the review of the selected 10 CFR 50.59 safety evaluations.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspector conducted a complete walkdown on the accessible portions of the reactor core isolation cooling (RCIC) system. The walkdown included reviews of the system operating procedure 2.2.22, “Reactor Core Isolation Cooling System (RCIC),” piping and instrument drawing M245, Updated Final Safety Analysis Report Section 4.7, and plant technical specifications. In addition, the inspector performed a system line-up review including verifying that valves and electrical breakers were in the proper standby line-up condition. The inspector also reviewed open work orders, problem reports, temporary modifications and operability evaluations to assess any outstanding RCIC equipment and/or component deficiencies.

The inspector conducted a partial system walkdown of the standby liquid control (SLC) system. This included reviewing SLC valve static mimic to ensure proper system alignment and a walk down of accessible portions of the SLC system. The inspector confirmed that the system was properly aligned to support normal and emergency plant operations.

A complete walk down was performed on the emergency diesel generators (EDGs) which are considered risk significant. Related temporary modifications, open maintenance requests, operability evaluations and operator work arounds were reviewed. All components related to operability of the EDGs were verified for proper alignment. Electrical control power was available as required. The licensee had planned adequate corrective actions to resolve degraded equipment conditions related to the EDGs.

b. Findings

No findings of significance were identified.

1R05 Fire Protection Quarterly

a. Inspection Scope

The inspector toured plant areas important to safety to observe conditions related to: (1) transient combustibles and ignition sources; (2) the material condition and readiness of fire protection systems and equipment; and (3) the condition and status of fire barriers used to prevent fire damage or fire propagation. The areas toured include:

- Reactor core isolation cooling quadrant,
- Emergency diesel generator building,
- Residual heat removal quadrants,
- Station Blackout diesel generator,
- Reactor building component cooling water rooms, and
- General reactor building areas

The inspector monitored the performance of the fire brigade training drill conducted on November 5, 2001. The drill involved a simulated fire in the generator portion of the “B” emergency diesel generator. The inspector observed fire brigade personnel performance, and verified that the licensee’s pre-planned drill scenario was followed and that the drill objectives were met. The inspector verified that proper protective clothing and breathing apparatus was donned, that sufficient fire fighting equipment was brought to the scene, and fire protection personnel entered the fire area in a controlled manner. The inspector also ensured that fire hoses were capable of reaching the fire location, and that communication between brigade members was efficient and effective.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspector reviewed documentation of operating history since the last requalification program inspection. This period covered September 1999 through September 2001. Documents reviewed included NRC inspection reports, licensee event reports, licensee Deviation/Event Reports, and the NRC plant issues matrix. The inspector did not detect operational events that were indicative of possible training deficiencies.

The following inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” and Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” Appendix A, “Checklist for Evaluating Facility Testing Material,” Appendix B, “Suggested Interview Topics,” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP),” as acceptance criteria.

The operating tests for the weeks of September 17, 2001 through October 29, 2001, and the biennial written exams given in May 2000 through June 2000 were reviewed for quality. PRA risk insights were used in examination development.

Training remediation and re-evaluation records were reviewed for the past two years and training attendance documentation was verified to be complete and up-to-date.

The inspector observed the dynamic simulator exams and job performance measures (JPMs) being administered. These observations included facility evaluations of crew and individual performance on the dynamic simulator exam.

Operators were interviewed for feedback on the license operator requalification training (LORT) program and quality of the training they had received. Training department personnel were interviewed to determine their knowledge and understanding of the training program.

Simulator performance and fidelity were reviewed for conformance with the reference plant control room. The inspector also reviewed simulator deficiency reports covering the period from September 1999 through September 2001.

A sample of records for requalification training attendance, license reactivations, and medical examinations were reviewed for compliance with license conditions and NRC regulations.

The results of all operating tests for all licensed operators, 25 senior reactor operators (SROs) and 19 reactor operators (ROs), for year 2001 were reviewed for performance and grading. Final results indicated all operators and all crews passed the operating tests.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed problems involving selected in-scope systems, structures, and components (SSCs) to assess the effectiveness of the maintenance rule (10 CFR 50.65) program. The inspector verified that the unavailability associated with the maintenance activity was properly captured. The review focused on proper characterization of failed SSCs as related to the following:

- Proper classification of an equipment failure for the station blackout (SBO) diesel generator as documented in problem reports 01.9573 and 01.9979 (chipped teeth on SBO ring gear).

- Proper classification of an equipment problem with control rod drive system hydraulic control unit no. 50-31.

- Proper classification of work associated with the diesel fire pump.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspector reviewed four on-line maintenance activities (listed below) to evaluate the licensee's risk assessment process. The inspector reviewed the work plans and packages against the criteria contained in procedures 1.5.21, "Integrated Scheduling Activities and Guidelines," and 1.5.22, "Risk Assessment process." The inspection included a review of the risk assessments and contingencies that were established, and verification that effect on plant risk and protected equipment was discussed during briefings and shift turnovers.

- Nitrogen tank replacement per MR E0000045

- Reactor vessel water level reference leg backfill per procedure 2.2.80

- Replace RCIC system turbine flow controller

- Replace "A" standby gas treatment system heater

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed the following operability evaluations to verify that continued operability was justified adequately. The Pilgrim Updated Final Safety Analysis Report, technical specifications, and procedure 1.3.34.5, “Operability Evaluations,” were used as references to assess the adequacy of the evaluations. The inspector also reviewed associated engineering evaluations, problem reports, design calculations, and test data that supported the licensee’s determinations of equipment operability. Lastly, planned corrective actions and schedules were reviewed.

- OE 01-49 “B” Emergency Diesel Generator speed oscillations at low or no load conditions.
- OE 01-52 “A” Emergency Diesel Generator jacket water heater supply breaker tripped and wouldn’t reset. Corrective maintenance planned for the next EDG overhaul outage scheduled for March 2002.
- OE 01-61 “B” Emergency Diesel Generator did not return to the proper frequency band when changed to the isochronous mode of operation.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed ten risk-significant plant modification packages to verify that: (1) the design bases, licensing bases, and performance capability of risk significant Structures Systems or Components (SSCs) had not been degraded through modifications; and, (2) the modifications performed during increased risk configurations did not place the plant in an unsafe condition.

For the selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations, such as instrument set-point and uncertainty calculations, to determine the design adequacy of the changes. The inspectors additionally reviewed the field change notices that were issued during installation, the post-modification testing and the instrument calibration records. These reviews were performed to ensure that the design packages accurately described the installed configuration of the components and that the installed modifications adequately met the design performance requirements. For components that required environmental and/or seismic qualifications, the inspectors reviewed the qualification records to ensure that the selected components were capable of performing their safety function at their installed location. Finally, the inspectors reviewed the affected drawings and UFSAR sections to verify that the affected documents were appropriately updated.

For selected plant modifications, the inspectors also conducted field surveys of the installations to ensure that the physical configuration of the installed components and design corresponded to the description included in the applicable design documents.

The modification packages reviewed are identified in the list of documents reviewed. The selected modifications took into consideration the significance of the changes, the cornerstone they protected, and the contribution of the system to core damage frequency.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed the post maintenance test for maintenance request (MR) 01119155, replacement of the feed water master level controller. The review included ensuring that the effect of the test on the plant had been evaluated adequately, verifying the test data meet the required acceptance criteria, and ensuring that the test activity was adequate to verify system operability and functional capability following maintenance.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed the results of the following surveillance tests:

- 8.M.2-2.10.8.2, “Diesel Generator “B” Initiation by RHR Logic,”
- 8.M.2.2.6.7, “RCIC Simulated Automatic Actuation”

The inspector verified that the test acceptance criteria was consistent with technical specifications requirements, that the test was performed in accordance with the written procedure, the test data was completed and met procedural requirements, and that the system was returned to service properly following testing.

b. Findings

No findings of significance were identified.

1EP2 Alert and Notification System Testing (ANS)

a. Inspection Scope

During an EP program inspection conducted in May 2001, the NRC raised questions with the accuracy and consistency of the licensee's siren test records that were used to develop the data for the ANS performance indicator (PI). In addition, questions were raised regarding the adequacy of the licensee's ANS testing criteria for determining siren operability. The inspector characterized these issues as unresolved items, (URI) 50-293/01-03-01 and URI 50-293/01-03-02. The inspector reviewed the Pilgrim Nuclear Power Plant Public Alert and Notification System Design Basis Document, dated June 1985 and the associated corrective actions for the test record errors. Also, the inspector reviewed the corrected PI data to ensure the licensee had not exceeded the green response band.

(Closed) URI 50-293/01-03-01: ANS Data in Need of Revision

The inspector reviewed problem reports 01.1952, 01.1953, 01.1954 and 01.1963 and found the licensee's corrective actions were adequate. The licensee reviewed all past siren data to ensure errors were corrected and recalculated their PI data. There was no color change to the ANS PI as a result of the corrections. Based on this information, this URI is closed.

(Closed) URI 50-293/01-03-02: Adequacy of Siren Testing Criteria Not Described in the Emergency Plan

It was noted during the program inspection in May 2001, that the licensee does not conduct quarterly growl tests as described in NUREG 0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The licensee spoke with the siren vendor and confirmed that their siren system is incapable of initiating a growl test (short, low volume pulse) but can manually initiate a wail test in which the siren wails at full volume for a few seconds. The licensee only performs a wail test on a yearly basis because they feel the full volume will unnecessarily alert the public; however, they test the electronic system that activates the wail function during the monthly maintenance program. Also, the inspector reviewed the licensee's siren design basis document that was submitted for approval to the Federal Emergency Management Agency in June 1985 and found that the document commits to a testing program but doesn't specifically describe the testing criterion. The licensee stated that they are planning to replace the current siren system in 2002-2003 and is considering a system that can perform growl tests. The NRC determined that the licensee's current siren testing program complies with the design basis document. Based on this information, this URI is closed.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector conducted an in-office review of licensee submitted changes for the emergency plan-related documents listed below to determine if the changes decreased the effectiveness of the plan. A thorough review was conducted of documents related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS documents. The inspector reviewed the following submitted documents:

EP-IP 100, “Emergency Classification and Notification”, Rev. 15
EP-IP 231, “Onsite Radiation Protection”, Rev. 6
EP-IP 240, “Emergency Security Organization Activation and Response”, Rev. 9
EP-IP 300, “Offsite Radiological Dose Assessment”, Rev. 3 & 4
EP-IP 330, “Core Damage”, Rev. 3
EP-IP 400, “Protective Action Recommendations”, Rev. 9

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2PS2 Radioactive Material Control Program

a. Inspection Scope

The inspector reviewed the selected documents and conducted the following activities to ensure that the licensee met the requirements specified in its program for the unrestricted release of material from the Radiologically Controlled Area (RCA):

- the most recent calibration results for the radiation monitoring instrumentation (Small Article Monitor, SAM-9), including the (a) alarm setting, (b) response to the alarm, and (c) the sensitivity;
- the licensee’s criteria for the survey and release of potentially contaminated material using gamma spectroscopy (calibration efficiency for bulk sample analyses);
- QC control charts for the gamma spectroscopy;
- the methods used for control, survey, and release from the RCA; and
- associated procedures.

The review was against criteria contained in 10CFR20, NRC Circular 81-07, NRC Information Notice 85-92, NUREG/CR-5569, Health Position Data Base (Positions 221 and 250), and the licensee’s procedures.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP)

a. Inspection Scope

The inspector reviewed the following documents to evaluate the effectiveness of the licensee's Radiological Environmental Monitoring Program (REMP) at the Pilgrim site. The requirements of the REMP are specified in the Technical Specifications/Offsite Dose Calculation Manual (TS/ODCM).

- the 1999 and 2000 Annual REMP Reports;
- selected analytical results for 2001 REMP samples;
- the most recent ODCM (Revision 8, August 2, 1998) and technical justifications for ODCM changes, including sampling media and locations for Revision 9;
- the 1999/2000 QA Audit Reports (Audit Report Nos. 99-02 and 00-02) for the REMP/ODCM implementations;
- review of 2000/2001 QA Surveillance Reports (Surveillance Report Nos. 00-121, 00-138, 01-002, 01-003, 01-004, 01-048, and 01-056);
- the most recent calibration results (from March-September 2001) for all TS/ODCM air samplers;
- the most recent calibration results (March 2001) of the meteorological monitoring instruments for wind direction, wind speed, and temperatures;
- weekly span check results (2000 and the 3rd Quarter 2001) of the meteorological monitoring instruments and the 2000/2001 Operation Logs;
- review of the 2000 meteorological monitoring data recovery statistics;
- review of contractor laboratory (Duke Engineering & Services, Environmental Laboratory) in the areas of:
    a. QA/QC Manual for the contractor laboratory;
    b. 2000 Semi-Annual Quality Assurance Status Reports, Analytical Services; and
    c. 2000 Semi-Annual Quality Assurance Status Reports, Dosimetry Services.
- implementation of the environmental thermoluminescent dosimeters (TLDs) program;
- the Land Use Census procedure and the 2000 results; and
- associated procedures to implement the REMP.

The inspector toured and observed the following activities to evaluate the effectiveness of the licensee's REMP:

- observation for the operability of meteorological monitoring instruments at the tower and the control room;
- observation for the weekly span checks performance (wind speed, wind direction, temperature, and delta temperature) at the primary meteorological monitoring tower;
- observation for air iodine/particulate and water sampling techniques; and
- walkdown for determining whether all air samplers, milk farms, and 25%TLDs were located as described in the ODCM (including control and indicator stations) and for determining the equipment material condition.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed the following documents to ensure the licensee met all requirements of the performance indicator from the third quarter 2000 to the second quarter 2001:

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- associated procedures.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed problem reports (PRs) associated with 10 CFR 50.59 issues and plant modification issues to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the corrective actions for the issues were appropriate. The inspectors also reviewed the licensee's resolution of PR 01.9774 involving inaccurate reactor vessel level indication experienced during reactor cool down from the August 13, 2001, automatic reactor scram.

On May 28, 1993, the NRC issued IE Bulletin 93-03 requesting licensees to "implement hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability." The NRC Bulletin evolved from the licensees' identification of level indication errors that emerged during reactor depressurization evolutions. Specifically, licensees determined that, during reactor operation, noncondensible gases could become dissolved in the reference leg of the reactor vessel water level instruments and lead to a false high level indication as the gases expanded and expelled water from the reference leg during reactor depressurization. In their response to the Bulletin, the licensee informed the NRC that they would install a modification that caused a small amount of control rod drive water to flow continuously through the reference leg tubing, from the transmitter to the condensing pot. This continuous upward flow would prevent the noncondensible gases from entering the reference leg. Vulnerabilities of this modification led to the indicated post-scram water level problems on August 13, 2001.

The design vulnerabilities were discovered during the post-event investigation. Specifically, the licensee found that, under certain circumstances, the current design could cause the redundant level instruments to provide erroneous indication (see Section 4OA3). As a result of this finding, the licensee issued PDC 01-15. This procedure change would allow them to operate the instrument reference leg backfill system for only three hours every 90 days, rather than continuously, as operated in the past and originally intended. The NRC evaluated the regulatory compliance and adequacy of the licensee's resolution of the issue.

b. Findings

The inspector found that the licensee's resolution of the reactor level instrument performance issue described in problem report PR 01.9974 was reasonable. The bases for the licensee's resolution are contained in the root cause analysis of the August 13, 2001, event (engineering evaluation EE 01-43), calculation M-1185, and the design analysis of PDC 01-15. The licensee had also evaluated the Pilgrim level instrument performance issues in a November 1992 analysis, BEC-002-R-01, "Investigation of Level Indication Anomalies at Pilgrim Nuclear Power Station." In these evaluations, the licensee theorized that non-condensible gases dissolved in the upper layer of the water in the level instrument condensing pot, would essentially remain in the condensing pot, unless transported downward into the reference leg by some other mechanisms, such as convection and/or piping leakage. Because the water temperature is highest at the top of the reference leg, near the condensing pot, the licensee excluded convection as a transport means.

Regarding travel of non-condensible gases in the instrument reference leg due to leakage, the licensee determined that the small amount of gases being dissolved in the water in the upper portion of the piping, between the condensing pot and the containment penetration, would produce only minor level indication errors during reactor depressurization and was, therefore acceptable. Their conclusion was based on the penetration being located less than 18 inches below the condensing pot and the gentle slope of the piping which prevents accumulation of non-condensible gases at a high point in the piping. Based on the above determination, the licensee calculated that, with no control rod drive (CRD) water flowing into the level instrument reference leg, the non-condensible gases would not travel beyond the containment penetration as long as the leakage was maintained below 15 milliliters (mL) per day for 90 days. The licensee also calculated that, with the reference leg backfill system in service for a three-hour period, the CRD system is capable of injecting into the reference leg more than three times the amount of water contained in the condensing pot and the in-containment piping. This amount of water would be sufficient to flush out any gas laden water in the volume of concern.

To ensure that the instrument line leakage remained below the 15 mL per day postulated in their analysis, the licensee tightened all connections and fittings and sprayed the instrument lines with a photo sensitive powder that would allow them to detect very small leaks. In addition, they initiated regular, bi-weekly inspections of the instrument lines and racks.

In response to the inspectors' question regarding leakage from the reference to the active leg through the transmitter manifold equalizing valve, the licensee stated that, during the refueling outage, they normally conduct two tests, either of which would reveal a leaking valve. Regarding failure of the valve in service, the licensee believed that the differential pressure across the bypass valve was sufficiently low to preclude any leakage across it. The licensee also stated that any identified leakage would be measured and that the flushing period would be adjusted downward from the 90 days if it exceeded 15 mL per day. Based on the above review, the inspectors concluded that the licensee's evaluation and decision to flush the level instrument reference leg for three hours every 90 days was reasonable and acceptably resolved the system vulnerabilities identified following the August 13, 2001, event. The licensee indicated they had engaged a consultant to further evaluate the issue and identify alternative solutions.

4OA3 Event Follow-up

1. (Closed) URI 50-293/01-05-05: Loss of Emergency Core Cooling System (ECCS) Initiation on Low Reactor Vessel Water Level.

Green. The inspectors identified a Non-Cited Violation of the Design Control Program for failure to assure that the 1993 design changes made to the reference leg of the reactor vessel level instrumentation were adequate and subject to the same control measures applied to the original design.

Following the automatic reactor scram, on August 13, 2001, the wide range reactor vessel level indicated higher than the actual water level for approximately 30 minutes. This erroneous indication resulted from the inadvertent draining of the instruments reference legs into the hydraulic control unit (HCU) accumulators, when the control rod drive (CRD) charging header (the source of the reference leg backfill) was isolated and the scram was reset. During the 30-minute period, the emergency core cooling systems (ECCS) would have not initiated on low water level as designed. The adequacy of the design and the significance of the event were undetermined pending the licensee's completion of the detailed root cause analysis.

A connection between the CRD system charging header and the reactor vessel level instruments was installed in 1993, in response to NRC Bulletin 93-03, to address concerns regarding erroneous level indication during reactor depressurization. The intent of the modification was to provide a continuous upward flow of CRD water in the level instrumentation reference legs to prevent accumulation of non-condensible gases therein. Accumulation of non-condensible gases in the level instrument reference leg was determined to be the cause of the level indication errors.

The August 13 event was unrelated to the Bulletin concerns, but design vulnerabilities introduced with the 1993 modification allowed the event to occur. During the root cause analysis of the event, the licensee determined that a loss of CRD water flow into the reference legs due to a CRD pump trip or manual closure of the CRD charging water header supply valve 301-25 with scram reset could result in the equalization of the pressure across the check valves that isolate the CRD system from the level instrumentation. This pressure equalization would prevent the check valves from sealing and result in a loss of inventory in the reference leg of redundant level instruments, as in the August 13 event. The resulting level indication errors could prevent or delay the initiation of the safety functions associated with the level instruments.

As stated in inspection report 50-293/01-05, this event was more than minor because it affected multiple trains of safety-related reactor vessel instrumentation. The Phase 1 Significance Determination Process (SDP) determined that a Phase 2 evaluation was required because there was an actual loss of a safety function (automatic initiation of the emergency core cooling system). Since the Phase 2 workbooks do not explicitly include instrumentation, the Senior Reactor Analysts determined that a Phase 3 risk evaluation should be performed. The Phase 3 risk assessment concluded that the safety significance of this event was very low (GREEN). The bases for the SDP conclusion are as follows:

- The duration that both trains of reactor vessel level indication were inoperable was approximately 30 minutes. The chance of having a low probability event, like a loss of reactor vessel inventory accident, during a 30-minute period is extremely unlikely. The calculated delta-CDF for a loss of reactor inventory event during a 30-minute period is orders of magnitude below the green-white threshold.

- Diverse containment high pressure instruments were available to automatically start reactor inventory makeup systems if a loss of coolant accident had occurred during the 30-minute period of vulnerability.

- Redundant reactor vessel level indication was available throughout the event which provided operators accurate vessel level indication. Had operators concluded that the available level indications were unreliable, plant emergency procedures would have directed the operators to take actions to ensure that adequate reactor inventory levels were maintained.

- Any significant decrease in reactor pressure would result in the reference legs refilling from the control rod drive charging water header. Refilling the reference legs would restore the level instruments to operable. Most decreases in reactor vessel inventory also result in a decrease in reactor pressure.

10 CFR 50, Appendix B, Criterion III, “Design Control”, requires, in part, that design changes be subject to design control measures commensurate with those applied to the original design. The design control measures shall provide for verifying the adequacy of the design by the performance of design reviews or suitable testing program. Contrary to this requirement, during the 1993 design changes, the licensee did not verify the adequacy of the revised design and did not assure that the design control measures applied to the changes were commensurate with those of the original design. Because the finding is of very low safety significance and was captured in the licensee's corrective action program as PR 01.9974, it is being treated as a Non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 50-293/01-07-01)**

2. <u>(Closed) LER 2001-002: Control Room High Efficiency Air Filtration System (CRHEAFS) Unable to Maintain Control Room Positive Pressure at One Location.</u>

   This LER identified that the CRHEAFS failed to adequately pressurize one door of the control room during a surveillance test. The inspector reviewed the corrective actions taken to determine if they were appropriate; this included a review of procedure 2.2.46, “Control Room, Cable Room, and Computer Room Heating, Ventilation, and Air Conditioning System.” Problem report 01.9082 was written to address this condition. No violation of NRC regulations were identified. The final root cause evaluation has not been completed.

3. <u>(Closed) LER 2001-005: Manual Scram While Subcritical Due to Personnel Error.</u>

   This issue was previously discussed in NRC Inspection Report 50-293/2001-003, Section 40A7. This event was characterized as a Non-Cited Violation. The licensee’s immediate corrective actions were reviewed and determined to be appropriate. The final root cause evaluation has not been completed.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. W. Riggs, Director of Nuclear Assessment, and other members of licensee management at the conclusion of the inspection on November 29, 2001. The licensee acknowledged the findings presented.

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

ATTACHMENT

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

E. Almeida	Manager, Design Engineering
M. Bellamy	Site Vice President
J. Bonner	Supervisor, Electrical Design Engineering
C. Brenneni	Engineering Superintendent Regulatory
P. Dietrich	General Manager - Plant Operations
E. Graham	Senior Operations Instructions
C. Hickey	Quality Assurance Senior Engineering
M. Jacobs	Supervisor, S&SA
W. Lobo	Regulatory Affairs
T. McElhinney	Manager, Technical Support Engineering
P. Pace	M/C Design Engineering
W. Riggs	Director, Nuclear Assessment
M. Santiago	Operations Training Superintendent
T. Trepanier	Training Director
S. Willoughby	Licensed Operator Requal Supervisor
J. Veglia	Engineering Programs and Components

b. List of Items Opened, Closed and Discussed

Closed

50-293/01-02	LER	Control Room High Efficiency Air Filtration System (CRHEAFS) Unable to Maintain Control Room Positive Pressure at One Location
50-293/01-05	LER	Manual Scram While Subcritical Due to Personnel Error
50-293/01-07-01	NCV	Inadequate Design Control pertaining to Reactor Level Instrument Design Change
50-293/01-03-01	URI	ANS Data in Need of Revision
50-293/01-03-02	URI	Adequacy of Siren Testing Criteria Not Described in the Emergency Plan
50-293/01-05-05	URI	Loss of Emergency Core Cooling System (ECCS) Initiation on Low Reactor Vessel Water Level

c. List of Documents Reviewed

Plant Design Changes

PDC 99-09	Decrease of the EDG Building Low Temperature Design Limit, Rev. 0
PDC 99-18	Installation of Isolation dampers on MCRECS Supply and Exhaust Ducts. (E-900041)
PDC 00-11	Modify M01001-7 A/B/C/D Body Drains (E-0000122)
PDC 00-18	Replacement of Panel D6 Breakers
PDC 00-19	Replacement of Neutron Monitoring Recorders on Panel C905, Rev. 0

PDC 00-20	Reactor Building Roof Replacement (E-000043)
PDC 00-25	New Accumulator Air-Compressor and SBGT capacity Expansion (E-00012921)
PDC 00-32	MOV Design Changes-MO220-2 (P-9900204)
PDC 01-05	Restoration of Control Rod Drive Stabilizing Valves, Rev. 0
PDC 01-15	Reactor Water Level Instrumentation Reference Leg Backfill System

10 CFR 50.59 Safety Evaluations

SE- 3298	EDG Air Start System, Rev. 0
SE- 3326	Reactor Building Closed Cooling Water System Procedure 2.2.30, Rev. 0
SE- 3329	Change EDG Fuel Consumption Rate, Rev. 0
SE- 3294	Test RCIC and HPCI Pumps at ½ rated Pressure
SE- 3338	Standing PDC for Mechanical and Civil/Structural Department Modifications.
SE- 3354	Restoration of CRD Flow Stabilization Loop and Replace Carbon Steel Pipe with SS Pipe.
SE- 3351	480V MCC Molded Case Acceptance Criteria
SE- 3365	New HPCI Test Valve HO-2301-320 to Replace RO2301-59
SE- 3339	Standing PDC for Instrument and Control Modification 2001
SE- 3319	Effect on LOCA/ECCS Analysis Results of PS230, Rev. 1
SE- 3317	Re-introduce HPCI Design Information to FSAR, Sec. 6
SE- 3309	Perform 150# RCIC Pump Test at rated Flow w/orifice in

Screened-Out 10CFR50.59 Safety Evaluations

FRN 91-08-23	PSV-1401-28B, Replacement of IST Relief Valves-Gasket Change, Rev 0
FRN 99-05-06	Emergency Bus Restoration
FRN 98-23-03	Installation of Timer Unit on Relays 27-B2X2 & 27-B2Z2
FRN 00-01-68	MSIV Test Line Drain
FRN 00-01-93	Reactor Head Vent Flange Replacement
FRN 00-01-95	Seismic Restraints for Standby Gas Treatment System Fans 210A & B
FRN 00-02-03	Replacement Motor for Fuel Pool Cooling Pump P210A, Rev. 0
FRN 01-01-43	HPCI Orifice Installation, Rev. 0
FRN 01-01-01	Installation of Tubing Support to Eliminate excessive Vibration
FRN 01-01-56	CRD Hatch Bolt Replacement
FRN 01-01-80	Seismic Restraint for Replacement Nitrogen Tank T212

Problem Reports

PR 99.9421, 00.1847, 00.3192, 00.3292, 01.0063, 01.3059, 01.4292, 01.4293, 01.9004, 01-9774, 01-9838

Procedures

2.2.8	"Standby AC Power System (Diesel Generators)", Rev. 69
2.1.12.1	"EDG Daily Surveillance", Rev. 37
2.2.19.5	"Residual Heat Removal Modes of Operation for Transients", Rev. 6
2.2.30	"Reactor Building Closed Cooling Water System", Rev. 47
2.2.82	"Reactor Vessel Water Level Control System", Rev. 31
3.M.2-10	"Feedwater Control Valve Isolation and Maintenance", Rev. 16
NE3.20	"Preparation, Review, Approval, Revision, and Closeout of Modifications", Rev. 8
NOP83E1	"Control of Modifications at Pilgrim Station", Rev. 27

NOP83E5	"Safety Reviews", Rev. 14
NOP83E5	"10 CFR 50.59 Review Process", Rev. 15
TP99-038	"Feedwater Control Valves - Postwork Testing (PDC 98-29)", Rev. 2

Other Documents

BEC-002-R-01(Q)	Investigation of Level Indication Anomalies , Rev. 0
EE 01-43	C2208 Reactor Water Level Backfill System
M-1185	Reactor Water Level Reference Leg Backfill System Design Evaluation
M1D12-4	Process Diagram Control Rod Drive Hydraulic System, Rev. E7
MR 01118045	Generic Control Rod Movement Anomaly - Perform Continuity Check of Amphenol Connectors
MR 01111710	Stroke Time Control Rods Due to Excessive Insert Signal
MR 01114167	Perform On-line Timing of Control Rods Slow to Insert 48 to 46 During Control Rod Exercise
MR01114168	Perform Power Reduction Timing of Control Rods Slow to Insert 48 to 46 During Control Rod Exercise
M-893	EDG Building Low Temperature Design Evaluation, Rev. 0
PS-229	Setpoint Calculation Degraded Voltage Time Delay
PS-230	Timing Calculation to Power Emergency Buses During LOCA
S&SA157	Estimate of PCT Increase due to Swing Bus Time Delay

Vendor Manual V-2036, Feedwater Regulating Valves, Rev. 2
Maintenance Rule System Summary for the Month of August 2001

d. List of Acronyms

CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CRD	Control Rod Drive
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EE	Engineering Evaluation
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
LER	License Evaluation Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
mL	Milliliter
MSIV	Main Steam Isolation Valve
MR	Maintenance Request
ODCM	Offsite Dose Calculation Manual
PDC	Plant Design Change
PMT	Post Maintenance Test
PR	Problem Report
QA	Quality Assurance
QC	Quality Control
RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling

REMP	Radiological Environmental Monitoring Program
RSPS	Risk Significant Planning Standard
SDP	Significance Determination Process
SE	Safety Evaluation
SRO	Senior Reactor Operator
SSC	Structures Systems or Components
TLD	Thermoluminescent dosimeter
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item