September 20, 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 - NRC INSPECTION REPORT 50-293/01-05

Dear Mr. Bellamy:

On August 18, 2001, the NRC completed an inspection at your Pilgrim reactor facility. The enclosed report documents the inspection findings which were discussed on September 10, 2001, with you 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, the safety significance of which is to be determined (TBD). The issue involves the temporary loss of the initiation of ECCS on reactor vessel low low water level signal. This issue became self revealing shortly after the automatic reactor scram that occurred on August 13, 2001. This issue is being treated as an unresolved item.

The inspectors identified one issue of very low safety significance (Green). The NRC inspectors found that safety-related cables located inside manholes were submerged in water for an extended period of time. Although not a specific violation of NRC requirements, this was a notable weakness since your staff did not have a routine monitoring and inspection program for these underground cables.

The inspectors also identified two issues which were determined to be violations of NRC requirements. However, these violations were not cited due to their very low safety significance and because the finding was entered into your corrective action program. If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555-0001; with copies to the Regional Administrator, Region I; the director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at the Pilgrim facility.

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 50-293/01-05
Attachment: Supplemental Information

cc w/encl: M. Krupa, Director, Nuclear Safety & Licensing
J. Alexander, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
S. Brennion, Regulatory Affairs Department Manager
J. Fulton, Assistant General Counsel
R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	50-293/01-05
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates :	July 1, 2001, through August 18, 2001
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector P. Frechette, Physical Security Inspector J. Furia, Senior Health Physicist D. Dempsey, Reactor Engineer
Approved By:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000293-01-05, on 07/01-08/18/2001; Entergy Nuclear Generation Company; Pilgrim Nuclear Power Station, Resident Inspection. Mitigating Systems and Licensee Identified Violation.

The inspection was conducted by resident inspectors, a physical security inspector, a reactor inspector and a radiation safety inspector. The inspection identified one finding that required further NRC review to determine safety significance. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP) . The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/overs0ight/index.html Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- GREEN. The inspector identified that portions of safety-related cables located in Appendix R ductline manholes were submerged in water. The licensee had not inspected the manholes since initial installation in 1987.

  The finding was of very low significance because no operability problems have been identified. (Section 1R06)

- TBD. A 30 minute period of inoperability of the emergency core cooling system (ECCS) initiation from low reactor vessel water level occurred shortly after the reactor scram on August 13, 2001. The safety significance of this finding is under review by the NRC risk analysts and is being treated as an unresolved item. (Section 1R14)

Cornerstone: Occupational Radiation Safety

- No findings of significance were identified.

Cornerstone: Physical Protection

- No findings of significance were identified.

B. Licensee Identified Findings

Two violations of very low significance, which was identified by the licensee, have been reviewed by the inspector. Interim corrective actions taken by the licensee appear to be reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

SUMMARY OF PLANT STATUS

Pilgrim Nuclear Power Station began the period at 100 percent core thermal power. On July 22, 2001, the licensee commenced a planned down power to 50 percent to perform a thermal backwash of the main condenser waterbox. Power was restored to 100 percent on July 23. On August 10, 2001, power was momentary reduced to 90 percent while the licensee placed the safety-related 4160 volt emergency buses on the emergency diesel generators. This was in accordance with procedure 2.4.144, "Degraded Voltage," when the voltage at the primary side of the startup transformer cannot be maintained above 342kV following a postulated plant trip of the unit. During this condition, the startup transformer is considered inoperable. On August 13, 2001, the plant automatically scrammed during logic system test on the "A" 4160 volt emergency bus. Both reactor recirculation pumps tripped causing a restoration from a flow bias scram. The unit was brought to cold shutdown to repair a body-to-bonnet leak in the reactor core isolation cooling steam admission valve MO-1301-17. On August 18, 2001, the mode switch was taken to startup and the reactor was taken critical.

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

1R04 Equipment Alignment

a. Inspection Scope

Three partial walkdown inspections were performed on the residual heat removal system, core spray system and the reactor core isolation cooling system (RCIC). Control room indications were checked to verify normal operating status. A random sampling of valve positions in the field were verified to be properly aligned in accordance with operating procedures. Work request tags were checked to determine that degraded equipment conditions awaiting corrective maintenance did not adversely affect operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

Three plant areas important to reactor safety were toured to observe conditions related to: (1) transient combustibles and ignition sources; (2) the material condition and readiness of fire protection systems and equipment; and (3) the condition and status of fire barriers used to prevent fire damage or fire propagation. The areas toured included the cable spreading room, emergency diesel generator rooms and the high pressure core spray pump room. Any degraded conditions were properly compensated for until appropriate corrective actions could be taken.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector conducted a walkdown inspection of the reactor building, auxiliary bay, intake structure and emergency diesel generator rooms to assess the effectiveness for the internal flood control measures. The Updated Final Safety Analysis Report, sections 8.9.2 and 2.4.4, Pilgrim Safety Analysis Report 50-84, "Internal Flooding Analysis," and the licensee's response to the Atomic Energy Commission letter dated August 3, 1972, were reviewed prior to the walkdown.

Items selected for review during the walkdown included watertight doors and piping penetrations, floor level alarms, and floor sump systems. Also, passive equipment such as curbing and drains on each level of the reactor building were inspected, as well as verifying that floor gratings were not blocked and found free of debris. The drain scuppers in the EDG building were verified to move freely and were not clogged by foreign debris.

The inspector also inspected three of the seven underground Appendix R ductline manholes subject to flooding, which contain safety-related cables. This cabling was installed as part of plant design change request number 84-03A. The inspector observed the condition of the cabling and adequate drainage from manholes.

b. Findings

Green. Inspection of Appendix R ductline manholes revealed portions of safety-related cables submerged in water. The licensee generated problem report 01.9816 to document and evaluate the condition.

Inspection of Appendix R manholes 25A, 25B, and 26B revealed approximately three feet of water in manholes 25A and 26B. The cabling in manhole 25A was found to be fully submerged. Through discussions with licensee personnel, and a review of plan layout drawing E344, the inspector noted that the Appendix R ductline manholes contain a sump for drainage, but no sump pumps or level alarm circuits. The cabling consists of both class 1E and non-1E circuits. There is no established frequency to inspect these safety-related manholes. The licensee had not inspected the manholes since the initial installation in 1987.

An Initial review of the licensee's operability evaluation revealed that all the class 1E, non-class 1E, and Raychem splices to be operable. All the cables in the manholes are direct burial cables and had been EQ qualified in harsh environments. The finding was more than minor because, if left uncorrected, it could become a more significant safety concern resulting in electrical grounds and the loss of safety-related equipment. Due to a lack of actual found condition duration and a lack of qualification test data, the long term aging and/or other degradation effect could not be determined. Since no operability problems have been identified, this finding is considered of very low safety significance and was therefore characterized as Green by the Significance Determination Process.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed the implementation of the maintenance rule (10 CFR 50.65) for selected systems and components. The review included a review of the applicable maintenance rule basis document and Updated Final Safety Analysis Report.

- Proper classification of equipment failures for the emergency diesel generator (EDG) issued since January 1999. The EDGs are designated as an (a)(2) system. Problem reports (PR) reviewed included PR 98.9149 (EDG cylinder indicator cock valve loose) and PR 99.9057 (EDG failure to start).

- Proper classification of equipment failures for the salt service water (SSW) system issued since January 1999. The SSW system is designated as an (a)(2) system. Problem reports (PR) reviewed included PR00.2705 ("E" SSW pump in required action range for high vibrations), PR 00.9281 ("D" SSW pump outside required action range for total dynamic head), and PR 01.9341 ("E" SSW pump in required action range for high vibrations).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

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

The inspector reviewed the risk associated with taking the reactor core isolation cooling system out of service for surveillance testing and reviewed the licensee's work plan for the week of July 23, 2001.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Evolutions

a. Inspection Scope

The inspector reviewed the integrated plant equipment and human performance following an automatic reactor scram that occurred on August 13, 2001. The resident inspector responded to the site during backshift hours, shortly after the scram, to independently verify plant conditions and plant data. The licensee initiated a post trip review per procedure 1.3.37 and classified the event as a Type 3, which required that a Multi-Disciplined Analysis Team (MDAT) conduct an investigation to identify the root cause and corrective actions. The inspector verified satisfactory resolution of several equipment issues prior to restart. Additionally, the inspector attended a special operations review committee (ORC) meeting which reviewed the post trip review report. Lastly, a conference call was held on August 16, 2001, between Entergy site management and NRC Region I management, to discuss the results of the post scram review.

The inspector reviewed the post scram information and verified the following reactor scram details:

- The automatic reactor scram occurred during the conduct of Procedure 3.M.3-1, logic system functional testing (LSFT) on the “A” emergency diesel generator (EDG). During restoration from procedure 3.M.3-1, the “A” emergency electrical bus (A5) inadvertently became de-energized due to a procedural error. The “B” emergency bus A6 remained energized throughout the event. The NRC enforcement aspects of this procedure error are further discussed in Section 40A7 of this report. The loss of A5 de-energized the “A” reactor protection system (RPS) which resulted in a ½ scram as designed.

- Reactor recirculation system flow was lost during the event. The loss of recirculation system flow resulted from the tripping of both reactor recirculation pumps, the details of which are discussed below. As a result of the loss of recirculation flow at power, a full reactor scram signal resulted when the “D” APRM tripped on hi as part of the flow biasing scram circuit. Operators responded in accordance with plant procedures to stabilize plant conditions and place the plant in the Cold Shutdown condition. No significant human performance issues were identified during this review. Equipment performance in response to the event is addressed in problem report (PR) 01.9779 and in the post scram review.

- The details of the recirculation system pump trips indicated that the pumps tripped for different reasons. The “A” pump tripped due to the loss of power to its AC oil pumps resulting from the loss of Bus A5. The

"B" pump tripped due to a degraded motor generator set voltage regulating circuit.

- During reactor cool down, operators identified that both "A" and "B" trains of wide range level instrumentation experienced inaccurate reactor vessel water level indication for approximately 30 minutes. The redundant and shutdown range reactor vessel water level instrumentation was not affected and continued to provide accurate indication. The inaccurate indication occurred after operators closed control rod drive system charging header valve CRD 301-25. CRD 301-25 was shut to insert one control rod which had settled at position 02 after initial insertion to 00.

The inspector continued to evaluate the licensee's corrective actions, which were still in progress at the end of the inspection period. Interim corrective actions included measures to keep the reference leg backfill system isolated to support plant restart. Engineering analysis in support of the interim measures, concluded plant operation could continue without the reference leg backfill system in operation for a defined period of time without concern for reactor vessel notching during a subsequent reactor vessel depressurization. The defined period of time is based on maximum instrument rack leak rates and piping configuration. The inspector verified that the level deviations noted after this scram were not indicative of previous level notching conditions caused by degassing (as discussed in NRC Bulletin 93-03).

b. Findings

Significance to be determined (TBD). An unresolved item was identified for a 30 minute loss of the emergency core cooling system (ECCS) initiation on low reactor vessel water level.

The wide range reactor vessel level indicated higher than actual water level with the "A" train reaching a maximum deviation of 26 inches and the "B" train reaching a maximum deviation of 11 inches. These erroneously high readings occurred for approximately 30 minutes. During this time period, the ECCS systems would not have initiated on low water level as designed. This was attributed to the reference legs draining back through the reference leg backfill system when the 301-25 valve is shut and the scram was reset. This can occur when the CRD system, the source of reference leg backfill, is at slightly lower pressure than the reactor, such that the differential pressure across the backfill system check valves does not fully seat the check valves.

This finding was greater than minor (0610* Group 1 questions) because it had an actual impact on safety in that the ECCS system would not have automatically initiated as designed on low level for approximately 30 minutes. This deficiency directly affected the Mitigation System cornerstone (0610* Group 2 questions) of the NRC significance determination process. The inspector completed the SDP Phase 1 screening process and determined that an SDP Phase 2 estimation was necessary because the issue represented an actual loss of safety function. More detailed risk review needs to be performed to consider the full effects of operating and shutdown risks.

The regulatory compliance and design adequacy aspects of this issue remain open pending licensee completion of the detailed root cause and NRC evaluation. This item is unresolved. **(URI 293/010005-05)**

1R15 Operability Evaluations

a. Inspection Scope

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

- OE 00-009, Relative humidity of standby gas treatment system units.
- OE 01-034, Reactor core insolation cooling steam admission valve, MO-1301-17, body-to-bonnet seal steam leak.
- OE 01-038, Quadrant cooling backdraft damper counter weights.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the list of operator work-arounds, lifted lead and jumper log, and licensee procedure 1.3.34.4, "Compensatory Measures," for determining impact for the aggregate effect of work-arounds on the operators ability to implement abnormal or emergency operating procedures. The inspector also verified that the licensee had entered the identified conditions in their corrective action program for resolution.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control (7112101)

a. Inspection Scope

The inspector toured exposure significant work areas, high radiation areas, and airborne radioactivity areas in the plant and reviewed associated controls and surveys of these areas to determine if controls (i.e., surveys, postings, barricades) were acceptable. For these areas, the inspector reviewed all radiological job requirements and attended job briefings; determined if radiological conditions in the work area were adequately communicated to workers through briefings and postings; verified radiological controls, radiological job coverage and contamination controls; and verified the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place, whether licensee surveys and postings were complete and accurate, and that air samplers were properly located. Reviews of RWPs used to access these and other high radiation areas and to identify what work control instructions or control barriers have been specified was conducted. Plant technical specification (TS) 5.7 and the requirements contained in 10 CFR 20, Subpart G, were utilized as the standard for necessary barriers. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspector also examined the licensee's programmatic controls for highly activated/contaminated materials (non-fuel) stored within the spent fuel pool.

The inspector examined licensee assessments including three Quality Assurance Surveillance Reports (QASR) [01-008, 01-013 and 01-031] and three Radiation Protection Department Self-Assessments [01-08, 01-09 and 01-11], in part, to evaluate the licensee's program for self-evaluation, problem identification and resolution.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (7112102)

a. Inspection Scope

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

The inspector reviewed ALARA job evaluations, exposure estimates and exposure mitigation requirements and ALARA plans were compared with the results achieved. A review of the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and generated shielding requests and their effectiveness to dose rate reduction were also conducted. Five exposure significant jobs were reviewed by the inspector, which included: motor operated valve inspection and repair; reactor vessel disassembly and reassembly; in-service inspection; inspection and testing of drywell snubbers; and valve betterment.

A review of actual exposure results versus initial exposure estimates was conducted, including comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness and exposure report distribution to support control of collective exposures to determine compliance with the requirements contained in 10 CFR 20.1101(b). The licensee's outage goals (less than 30 days duration and not more than 150 person-rem) were both met (28 days and 124 person-rem), with each being records for Pilgrim.

The inspector also examined nine post-work review records [01-002, 01-003, 01-004, 01-005, 01-006, 01-007, 01-009, 01-010 and 01-011] performed by the radiation protection staff to evaluate performance during RFO13.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

3. SAFEGUARDS
Cornerstone: Physical Protection

3PP1 Access Authorization Program (71130.01)

a. Inspection Scope

The following activities were conducted to determine the effectiveness of the licensee's behavior observation portion of the personnel screening and fitness-for-duty programs as measured against the requirements of 10 CFR 26.22 and the Licensees Fitness for Duty Program documents.

Five supervisors representing the Emergency Preparedness, Chemistry, Operations, Maintenance and Engineering departments were interviewed, on July 11, 2001, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. Two (2) Access Authorization/Fitness-for-Duty self-assessments, an audit, and event reports and loggable events for the four previous quarters were reviewed, during July 9-12, 2001. On July 11, 2001, five (5) individuals who perform escort duties were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were reviewed on July 10, 2001.

b. Findings

No findings of significance were identified.

3PP2 Access Control (71130.02)

a. Inspection Scope

The following activities were conducted during the period July 9-12, 2001 to verify that the licensee has effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10 CFR 73.55(d) and the Physical Security Plan and Procedures.

Site access control activities were observed, including personnel and package processing through the search equipment during peak ingress periods on July 9, 10, and 11, 2001, and vehicle searches, on July 10 and 11, 2001. On July 10, 2001, testing of all access control equipment; including metal detectors, explosive material detectors, and X-ray examination equipment, was observed. The Access Control event log, an audit, and three (3) maintenance work requests were also reviewed.

A review was conducted of one Condition Report (CR) generated and entered into the licensees corrective action program. The specific CR is identified in the list of documents contained in this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspector reviewed operator logs, licensee event reports and NRC inspection reports for the period of April 2000 to July 2001 to determine the accuracy and completeness for the reported Pilgrim performance indicators (PI). The inspector verified that the licensee had characterized past events in accordance with the NRC endorsed criteria contained in NEI 99-02, “Regulator Assessment of Performance Indicator Guideline.” The following PIs were reviewed:

- Unplanned Scrams per 7,000 Critical Hours
- Scrams with a Loss of Normal Heat Removal
- Unplanned Power Changes per 7,000 Critical Hours
- Heat Removal System Unavailability (RCIC)
- High Pressure Injection System Unavailability (HPCI)

The inspector reviewed the licensee’s programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators. The review included the licensee’s tracking and trending reports, personnel interviews and security event reports for the Performance Indicator data collected from the 2nd quarter of 2000 through the 2nd quarter of 2001.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Bellamy, Site VP, and other members of licensee management at the conclusion of the inspection on September 10, 2001. The licensee acknowledged the findings presented.

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

4OA7 Licensee Identified Violations

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

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 293/01005-02	10 CFR 50, Appendix B, Criterion III, “Design Control,” for inadequate design implementation. On July 18, 2001, the licensee identified that the shipping bolts on the drywell-to-torus vent line expansion bellows were installed; this condition has existed since original construction. The licensee generated problem report PR 01.9690 to document and address this condition. The licensee determined that the primary containment remains operable in this configuration and is developing a plan to remove the bolts.
NCV 293/01005-03	Pilgrim Technical Specifications 5.4.1 requires written procedures, appropriate for the circumstances, be implemented that meet the requirements of Appendix “A” of Regulatory Guide 1.33, which include Surveillance Tests. Entergy Procedure 3.M.3-1, Att. 8A, was not appropriate for the circumstances due to a missing step in the restoration section that resulted in the loss of electrical bus A5 and a plant scram on August 13, 2001. This issue is documented in the licensee’s corrective action program as PR 01.9779. This is being treated as a Non-Cited Violation.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

D. Burke, Security Superintendent, (acting)
M. Christopher, Radiation Protection Supervisor
C. Dugger, Vice President Operations/General Manager
B. Ford, Manager, Licensing
J. Henderson, Radiation Protection Supervisor
B. Olson, Radiation Protection Supervisor - Instruments
W. Riggs, Director, Nuclear Assessment
T. Tetzlaff, Radiation Protection Supervisor - Operations
C. Wend, Radiation Protection Manager
R. Wheat, Project Manager, PTI

b. List of Items Opened, Closed and Discussed

Opened

URI 293-010005-01	Reactor vessel water level deviation

Closed

NCV 293-010005-02	Drywell-to-torus expansion bellows bolts left installed
NCV 293-010005-03	Inadequate Surveillance Procedure causes loss of electrical bus A-5

c. List of Documents Reviewed

Plant Access Training - Fitness for Duty
Security Program Annual Audit Report 01-02
Fitness For Duty and Access Authorization Audit Report 00-02
Fitness for Duty Program Audit Report 01-01
Security Loggable event report, 01/00-03/01
Fitness for Duty Performance Data for six month period ending 12/31/2000, January 2,2001
Fitness for Duty Performance Data for six month period ending 6/30/2001, July 1, 2001
PR-01-1441, Evaluate adverse trend in perimeter equipment reliability

d. List of Acronyms

ALARA	As Low As is Reasonably Achievable
CFR	Code of Federal Regulations
CR	Condition Report
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
HPCI	High Pressure Coolant Injection
LSFT	Logic System Functional Testing
MDAT	Multi-Disciplined Analysis Team
ORC	Operations Review Committee
PARS	Publicly Available Records
PR	Problem Report
QASR	Quality Assurance Surveillance Reports
RCIC	Reactor Core Isolation Cooling
RWP	Radiation Work Permits
SSW	Salt Service Water
UFSAR	Updated Final Safety Analysis Report