Enclosure

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No.:	98-07
Docket No.:	50-293
Licensee:	BEC Energy 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Inspection Period:	July 26, 1998 - September 6, 1998
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EXECUTIVE SUMMARY

Pilgrim Nuclear Power Station
NRC Inspection Report 50-293/98-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers resident inspection for the period of July 26, 1998, through September 6, 1998; in addition, it includes the results of announced inspections by DRS inspectors. In addition, it includes an in-office review of licensee emergency procedure changes.

Operations

- Routine plant operations were performed well, especially during several downpowers to backwash the condenser water boxes. An increased use of industry operating experience was evident at the Plant Manager's morning meeting. A cross functional self assessment utilized industry peers and was self critical in identifying areas for improvement. (Section O1.1)

- The inspectors noted three occasions when degraded equipment conditions were either not expeditiously identified or communicated. Plant and operations management acknowledged the observation and stated there was an ongoing effort to strengthen management expectations in this area. (Section O2.1)

- Operations personnel performed fuel movements well in the spent fuel pool, with appropriate concerns for safety, and foreign materials exclusion. Communications between operator, spotter, and reactor engineering personnel were clear and concise. (Section O4.1)

Maintenance

- Use of mock-up training and extra electrical work precautions contributed to successful replacement of the electrical brushes on the exciter for the "B" motor generator set motor. (Section M1.1)

- The maintenance staff exhibited conservative decision making by placing the work to replace the "B" control rod drive (CRD) pump casing gasket on hold in lieu of attempting to perform the work with boundary valve leakage. (Section M1.1)

- Surveillance testing was done in a well controlled manner consistent with safety requirements. Testing was sufficient in scope to demonstrate that the subject equipment would perform their safety functions. Engineering and radiation protection personnel provided the necessary support of testing activities. (Section M1.1)

- The inspector determined that the performance and scheduling of selected infrequently performed technical specifications required surveillances were accurately tracked in the Master Surveillance Tracking Program for the core and containment cooling, 125VDC, 250VDC, core spray, standby gas treatment, and drywell/torus header systems. (Section M1.2)

- The inspector determined that the temporary repair of (salt service water) SSW spool piece JF29-8-4 was adequately performed and that the operability determination was appropriate in scope and detail. A thorough review of the work package by quality assurance personnel identified that the incorrect weld metal was used in the repair of the SSW pipe. (Section M2.1)

- The engineering staff properly utilized NRC GL 90-05 to develop and execute the non-ASME (American Society of Mechanical Engineers) repair plan for the degraded SSW (salt service water) spool piece. The inspection of nine similar pipe locations for erosion/corrosion was very rigorous. (Section M2.1)

- Improved material condition was noted in the reactor building control rod drive quadrant room due to increased management attention. Also, new permanent lighting was added to the torus room. (Section M2.2)

<u>Engineering</u>

- The NRC and the licensee's engineering staff identified deficiencies in the design of the control room high efficiency air filtration system, i.e., seismic qualification of duct supports, single failure design analyses, and inconsistencies between analysis assumptions and operational procedures. The deficiencies have been reported in two licensee event reports, including acceptable short term corrective actions. The failure to accurately transfer the system design into instructions, procedures or drawings was a violation of 10 CFR 50 Appendix B Criterion III "Design Control." (Section E2.1)

- An NRR safety evaluation agreed with the licensee resolution of non-conforming jet pump swing gate and restraint issues.

<u>Plant Support</u>

- The circumstances leading to the misplacement of the intermediate range detectors represented a lack of accountability and control of special nuclear material. The licensee was successful in finding the missing intermediate range monitors, however the search resulted in an accumulated dose of approximately one rem. The corrective actions to resolve the issue were determined to be appropriate. (Section S1.1)

TABLE OF CONTENTS

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the period at 50 percent power for a planned thermal backwash of the main condenser to reduce biofouling. Full power was attained on July 26, 1998. Several times throughout the period, power was again lowered to perform a backwash due to the buildup of biofouling in the condenser water box. The plant was at 100 percent power at the end of the report period.

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Plant operations were conducted with a proper focus on nuclear safety. The inspector verified during daily control room observations that operators remained attentive and responsive to plant conditions. Daily plant operations were conducted safely and in accordance with operating practices. The inspector conducted reviews of ongoing plant operations and observed that overall plant staff conduct was professional and safety-conscious. Operators were aware of the status of plant equipment, properly used three-way communications, and effectively implemented plant procedures.

Operators were required to monitor biofouling in the main condenser by closely trending the condenser hotwell temperature to ensure the temperature remained below the 120 degree administrative limit and above 26 inches of vacuum in the condenser. On several occasions operators lowered reactor power to stay below the limit to prevent condensate pump cavitation. Also, operators conducted several backwash operations of the condenser waterboxes which required lowering reactor power to approximately 50%. The inspector determined that operators performed very well during the power changes to cope with the effects of condenser biofouling.

A review of the standby line-up of the standby gas treatment (SBGT) was performed including electrical breakers, dampers and indications. The system was found to be properly lined up to support operability.

The Plant Manager's morning meeting included a new initiative to review recent operating experience information. This was implemented by having one manager review the current operating experience reports and select one event to be briefed at the morning meeting. The inspector attended several briefings and noted the events were covered in detail and specifically included a review for applicability at

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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.

Pilgrim. In addition, A one-week licensee self assessment was conducted which utilized industry peer members to conduct a cross functional assessment.

The fire brigade performed well during an unannounced fire drill in the reactor building. The operators assigned to the brigade arrived at the scene donned in fire protection equipment within approximately 10 minutes. The drill instructors provided close oversight of the operator response.

In summary, routine plant operations were performed well especially during several downpowers to backwash the condenser water boxes. An increased use of industry operating experience was evident at the Plant Manager's morning meeting. A cross functional self assessment utilized industry peers and was self critical in identifying areas for improvement.

O2 Operational Status of Facilities and Equipment

O2.1 Operator Awareness and Communication of Equipment Status Information

a. Inspection Scope

The inspector verified that actual equipment conditions were accurately discussed during shift briefings and documented in the daily operating report.

b. Observations and Findings

Out of service equipment was properly tracked in the technical specification logbook maintained in the control room. Degraded equipment was typically discussed at shift briefings and captured in the daily operating report. The inspector noted three inconsistencies with the identification and/or communication of equipment status information.

An operations daily morning report listed mechanical seal leakage from a reactor water clean-up system pump as 45 drops per minute when the leakage had degraded to approximately one gallon per minute. The inspector determined that the degraded seal leakage was not properly documented in the daily report or discussed at the daily management morning meeting. Also, a shift briefing did not include relevant information regarding the status of a control rod drive pump maintenance activity.

The third event occurred on August 15 when the turbine building tour operator observed that the auxiliary oil pump for the "C" reactor feed pump was unexpectedly operating. Normally, the auxiliary lubricating oil pump remains secured in the standby condition. Operators secured the auxiliary oil pump; however, the pump automatically restarted. After further investigation, the licensee identified a broken sensing line that caused the pump to start. The licensee implemented a temporary modification until the broken sensing line could be repaired. The inspector noted that the field operator identification of this condition demonstrated good attention to detail. The inspector noted that this condition had not been detected by control room operators when discovered by the field operators. The control room panel has an operating status light for each of the feed pump auxiliary lube oil pumps.

c. Conclusions

The inspectors noted three occasions when degraded equipment conditions were either not expeditiously identified or communicated. Plant and operations management acknowledged the observation and stated there was an ongoing effort to strengthen management expectations in this area.

O4 Operator Knowledge and Performance

O4.1 Observation of Spent Fuel Pool Operations

a. Inspection Scope (71707)

The inspector observed and assessed spent fuel movements in the spent fuel pool (SFP) during the search for missing intermediate range meters (IRMs).

b. Observations and Findings

The inspector observed the movement of three fuel bundles that were moved during the search for the four missing IRMs (refer to Section S1.1). The crane operator was observed to be experienced and knowledgeable in the operation of the SFP bridge crane. Foreign material exclusion practices were used by all personnel entering the controlled area around the SFP. Repeat back communication was observed between the operator, spotter, and the reactor fuels engineer responsible for inventory controls of the spent fuel. The inspector verified that the licensee followed the applicable plant procedures and that technical specification requirements were met.

c. Conclusions

Operations personnel performed fuel movement well, with appropriate concerns for safety, and foreign materials exclusion. Communications between operator, spotter, and reactor engineering personnel were clear and concise.

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) LER 50-293/97-04-01: Loss of Preferred Power and Oil Spill Due to Main Transformer Fault While Shut Down.

This issue was previously reviewed and documented in Section O8.5 of NRC Inspection Report No. 50-293/98-01 with no violations of NRC requirements identified. This LER is **closed**.

O8.2 (Closed) LER 50-293/98-06-01: Failure to Verify Compliance with Technical Specification (TS) Limit for Control Rod Worth.

This LER documented that the control rod withdrawal sequence provided by General Electric (GE) did not explicitly verify compliance with the TS limit on control rod worth. Beginning with cycle 7, GE evaluated the dropped rod accident on a generic basis (i.e., based on 280 cal/g) and no longer verified compliance with the plant specific control rod worth limits. Although compliance with the GE rod withdrawal sequence appropriately limits enthalpy and satisfied the purpose of that specification, it did not verify compliance with the maximum control rod worth specified in TS 3.3.B.3. A plant specific analysis was subsequently performed by GE which verified that the TS requirements for rod worth were met for the current operating cycle (cycle 12).

The inspector conduced an on-site review of the LER and verified that the core design procedure was in the process of being revised to include TS 3.3.B.3 to ensure that the cycle 13 core design will meet licensing requirements. The inspector determined that the failure to verify the maximum rod worth was a violation of technical specifications. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. **(NCV 50-293/98-07-01)**. This LER is **closed**.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Maintenance and Surveillance

a. Inspection Scope (61726)

The inspector observed portions of selected surveillance and maintenance activities to verify proper use of approved procedures, conformance to limiting conditions of operation, and correct system restoration following testing. Emphasis was placed on maintenance rule trending and status changes. The following activities were observed:

8A.15	"HPCI System Integrity Test"
8.5.4.1	"HPCI System Pump and Valve Quarterly Operability"
8.M.2-2.1.10	"4160 Volt Emergency Bus A5 and A6 Loss of Voltage and Degraded Voltage"
3.M.3-7.1	"Replace "B" Motor/Generator (MG) Set Motor Exciter Brushes"
8.9.1	"Emergency Diesel Generator (EDG) Surveillance"
MR E9800082	Installation of RHR Loop Decontamination Connections
MR 19800473	CRD Pump "B" Casing Gasket Replacement

b. Observations and Findings

Mock-up training was conducted at the training facility to facilitate replacement of the motor exciter brushes on the "B" reactor recirculation system MG set. The inspector observed that a maintenance supervisor attended the briefing along with operations personnel. Two electricians replaced the brushes using extra precautions since the MG set was operating. The electricians completed the work well with no problems noted by the inspector. The electricians noted some carbon build-up on the collector ring and noted the condition in the surveillance procedure. Visual examination of the brushes by the inspector determined that the degradation was due to normal wear-and-tear. The inspector determined that the use of the training mock-up and extra electrical work precautions contributed to successful replacement of the electrical brushes.

The inspector observed the maintenance work area drain process for the installation of a flanged connection in the RHR flushing line. The flanged connection will be used as a connection point during the upcoming RHR system piping decontamination. The maintenance area piping was isolated but could not be drained fully. When the piping was initially breeched, maintenance personnel used tygon tubing to drain water down to the floor in the "B" RHR quadrant room. The inspector observed that the tubing was not well secured in that it was off the floor which caused water to splash in the area and cross a contaminated area boundary. Licensee maintenance supervisory staff independently observed the condition and corrected the problem and verified that no spread of contamination occurred.

In preparation to replace the "B" CRD pump casing gasket, the licensee experienced problems with boundary valve seat leakage into the work area. The inspector estimated the leakage at approximately one gallon per minute. Maintenance management directed that the work be placed on hold pending repair of the boundary valves. The maintenance staff determined that although pump disassembly was possible, problems would be experienced during the pump re-assembly phase. The inspector determined that the maintenance staff exhibited conservative decision making by placing the work on hold in lieu of attempting to perform the work with boundary valve leakage.

Observed surveillance activities were performed safely and in accordance with approved procedures. Test details were discussed at a pretest briefing with all involved testing participants. During observation of the surveillance activities, proper use of self-checking and procedure adherence was noted. Test personnel were also noted to demonstrate good radiological protection practices. The surveillance activities were well supported by radiation protection and instrumentation and control personnel. The system engineer observed the test and performed preliminary evaluations of the test results. The inspector verified proper system restoration and that the test acceptance limits were met. No deficiencies were noted.

c. Conclusion

Use of mock-up training and extra electrical work precautions contributed to successful replacement of the electrical brushes on the exciter for the "B" MG set motor.

The maintenance staff exhibited conservative decision making by placing the work to replace the "B" CRD pump casing gasket on hold in lieu of attempting to perform the work with boundary valve leakage.

Surveillance testing was done in a well controlled manner consistent with safety requirements. Testing was sufficient in scope to demonstrate that the subject equipment would perform their safety functions. Engineering and radiation protection personnel provided effective testing and radiation protection oversight.

M1.2 Surveillance Frequency

a. Inspection Scope (61726)

The inspector reviewed technical specifications (TS) and a selected sample of infrequently performed TS surveillances in order to verify that the surveillances were completed within the allowable time intervals and properly documented in the master surveillance tracking program (MSTP) database.

b. Observations and Findings

The inspector reviewed TS surveillance requirements for the core and containment cooling, 125VDC, 250VDC, core spray, standby gas treatment, and drywell/torus header systems and verified in the MSTP that their past performance had been accurately tracked and future performances scheduled as required. Additionally, the inspector reviewed copies of surveillances 8.9.8.2 - 125VDC battery acceptance test, 8.9.2.3 - 250VDC battery acceptance test, and 8.7.2.2 - standby gas treatment inlet heater capability test and verified that the documented performance completion date was accurately reflected in the MSTP.

c. Conclusions

The inspector determined that the performance and scheduling of selected infrequently performed TS required surveillances were accurately tracked in the MSTP for the core and containment cooling, 125VDC, 250VDC, core spray, standby gas treatment, and drywell/torus header systems.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Non-ASME Code Repair on Salt Service Water (SSW) Piping

a. Inspection Scope (62707)

The licensee identified leakage coming from SSW system piping spool piece JF29-8-4. The spool piece is located in the SSW system outlet piping from the "A" reactor building closed cooling water system heat exchanger. The leakage consisted of two discrete through-wall pits. The inspector reviewed the engineering evaluation and observed the maintenance repair activity, MR 1981846.

b. Observation and Findings

The inspector verified that the licensee obtained verbal NRR approval to perform the temporary non-code repair of the SSW spool in accordance with NRC GL 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping." The repair consisted of welding a stainless steel metal patch to add strength to the degraded area. A review of the maintenance activity and maintenance records revealed that welder's qualifications were up to date. Inspection of the repaired area revealed no leakage with the system in service. The inspector noted that the quality assurance review of the work activity revealed that the wrong filler weld metal was used for the repair. A problem report and non-conformance report (NCR) were issued to document this concern. The disposition of the NCR determined that the metal used was acceptable for the application.

The licensee submitted letter 2.98.109 to the NRC for review and approval of the non-ASME code repair. The 10 CFR 50.59 safety evaluation determined that an unreviewed safety question was not introduced by the temporary pipe repair. Based on operating history, the licensee determined that the cause of the leaks were due to localized delamination of the SSW piping rubber lining. The licensee committed to visually monitor the repaired area once a shift until a permanent ASME code repair is completed, which will be performed during the next scheduled outage exceeding thirty days and no later than startup from the next refueling outage (i.e. RFO12). In addition, the licensee will perform biweekly monitoring (ultrasonic testing) of the adjacent area and ultrasonic testing of the cover plate material every three months until test results show that the test frequencies can be changed. The licensee inspected nine, vice the required five, similar piping locations in accordance with GL 90-05. These inspections found all locations greater than the manufacturer's tolerances.

c. Conclusions

The inspector determined that the temporary repair of SSW spool piece JF29-8-4 was adequately performed and that the operability determination was appropriate in scope and detail. A thorough review of the work package by quality assurance personnel identified that the incorrect weld metal was used in the repair of the SSW pipe; however, this was determined to be acceptable. The licensee properly utilized

NRC GL 90-05 to develop and execute the non-ASME repair plan for the degraded SSW spool piece. The inspection of nine similar pipe locations for erosion/corrosion was more rigorous than the GL requirements.

M2.2 (Closed) VIO 98-02-03: Degraded Equipment Problem Identification

The violation related to the NRC identification of several degraded plant equipment conditions which were not known by the plant staff. The licensee determined the reason for the violation was the failure to communicate management's expectations for prompt identification and correction of low threshold material deficiencies. The inspector determined that corrective actions were taken or planned for each individual problem identified in the violation.

Several broader corrective actions were also implemented, including re-emphasis of management expectations and the 12 week rolling maintenance schedule was changed to 13 weeks to dedicate a week for repair of lower level degraded equipment conditions. The inspector determined that the corrective actions taken and planned were reasonable and resulted in a large increase in the number of work request tags.

Licensee cross functional teams also performed detailed inspections of designated plant areas to identify and correct adverse equipment conditions. For example, the general material condition of the control rod drive (CRD) quadrant room was significantly improved. Chronic casing leakage was corrected on one CRD pump with repairs planned for the other. The inspector also noted the progress in preparations to conduct on-line chemical decontamination of the residual heat removal (RHR) loops. This was planned to improve the material and radiological conditions in the two RHR quadrant rooms which were presently designated as contaminated and high radiation areas. The chemical decontamination is planned in the fall 1998. Lastly, the inspector noted the installation of permanent lighting in the torus room. This violation is **closed.**

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 (Closed) LERs 50-293/98-08 and 98-16; (Closed) Item III.D.3.4 of NUREG 0737 (Open) VIO 50-293/98-02; Control Room High Efficiency Air Filtration System

a. Inspection Scope

The inspector performed a walkdown of the control room high efficiency air filtration system (CRHEAFS) and examined system operating, surveillance and alarm response procedures. Design calculations for the CRHEAFS were also reviewed along with the system description contained in the plant Final Safety Analysis Report (FSAR).

The CRHEAFS consisted of two redundant subsystems; a non-safety related portion used during routine plant operations, and a safety related portion that would be manually activated during a Loss of Coolant Accident (LOCA). The non-safety related subsystem was designed to maintain the temperature of the control room environment within desirable values, by passing a mixture of fresh and recirculated air through a series of heating and cooling coils. When activated during a LOCA, the safety-related subsystem was designed to prevent the infiltration of radionuclides by pressurizing the control room envelope above atmospheric pressure by supplying outside air through charcoal filters. Each subsystem had redundant components, i.e., there were two safety related charcoal filters and fans along with two non-safety related heating/cooling coils and supporting fan units. Both systems share some components; for example, both subsystems used the same air intake and exhaust plenums.

b. Observations and Findings

Design Inadequacies

In April, the NRC identified that some portions of the safety related, seismic category 1 control room ductwork subsystem were suspended from the non-safety related, seismic category 2 ductwork by metal straps. The licensee documented this condition in Problem Report 98-9202. Initially the licensee determined that this configuration was acceptable. A follow up detailed review completed on May 8, 1998, determined the arrangement was a condition that was outside of the plant design basis and additional modifications may be necessary to achieve full seismic qualification. This conclusion was reported to the NRC operations center as required by 10 CFR 50.72(b)(l)(ii)(B). The corrective action as outlined in Licensee Event Report (LER) 50-293/98-08, dated May 27, 1998, included a commitment to perform additional analysis and modifications as necessary to assure full seismic qualification.

The inspector conducted an on-site review of LER 98-08 and determined it contained an accurate description of the issue and corrective actions were adequate. Therefore, this LER was **closed**.

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that "measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, and procedures." The failure to have the as-built configuration of the CRHEAFS system in accordance with the design basis indicated inadequate measures were established to translate the design basis of the CRHEAFS into procedures, instructions and drawings. This is the first of 3 examples of a violation **(VIO 50-293/98-07-02)** of 10 CFR 50, Appendix B, Criterion III, "Design Control."

Subsequent licensee reviews of the accident mode testing of CRHEAFS identified additional seismic deficiencies and a single failure vulnerability. Specifically, the licensee noted if damper AO-N-1 and its supporting ductwork in the non-safety related subsystem failed to open during a LOCA or seismic event, outside air could enter the control room environment without going first through the charcoal filters located in the safety related subsystem of CRHEAFS. As a result, the radiation dose operators would receive during a LOCA may increase, or the control room environment may become uninhabitable following a seismic event.

When this deficiency was detected, the licensee declared both trains of the CRHEAFS inoperable and entered the 36-hour shutdown action statement contained in Technical Specification (T.S.) 3.7 "Control Room Air Filtration System." Corrective action as outlined in LER 50-293/98-16 consisted of installing seismic restraints on the non-safety related portion of the seismic ductwork and gagging closed the damper. Once the work was completed, the TS action statement was exited.

The inspector conducted an in-office review of LER 98-16 and determined it contained an accurate description of the issue and corrective actions were adequate. Therefore, this LER was **closed**.

Chapter 10.17 of the Pilgrim Final Safety Analysis Report (FSAR) states that the CRHEAFS was designed with sufficient redundancy so that no single active failure could prevent the system from achieving its safety objective. However, the as-built system configuration was not in accordance with this design requirement because one active single failure, the failure of damper AO-N-1 to close, could render the CRHEAFS inoperable during a LOCA. This represents the second example of a violation **(VIO 50-293/98-08-02)** of 10 CFR 50, Appendix B, Criterion III, "Design Control."

<u>Analysis Weaknesses</u>

In a letter dated February 11, 1981, the licensee provided the NRC part of its response to item III.D.3.4, Control Room Habitability of NUREG 0737, "Clarification of TMI Action Items," including the results of an analysis, which indicated the CRHEAFS met the exposure limits contained in GDC 19. The analysis results were reviewed and accepted by the NRC in a letter dated June 24, 1982.

The inspector reviewed portions of the February 11, 1981, submittal and determined some assumptions in the analysis did not have an adequate basis and others may no longer be valid. Specifically, the analysis did not appear to account for the time delay that would occur between receipt of a high radiation signal in the control room ventilation intake and the manual operator action to shut down the normal ventilation. The time delay could be considerable, because stopping the non-safety related subsystem requires manipulation of switches outside of the control room envelope at elevation 51 of the turbine building.

The isolation delay was further affected by the alarm response procedure for a high radiation signal in the control room ventilation intake, which instructed operators to first confirm the signal was valid before isolating the intake. This delay could be considerable because signal confirmation involved having a health physics technician confirm the signal was valid by taking radiation surveys with a meter at the control room intake ductwork.

Because of these analysis and procedure deficiencies, during a LOCA operators could be exposed to radiation levels that were greater than contained in the plant design basis. The failure to transfer the control room habitability analysis into the CRHEAFS operational procedures represented the third example of a violation **(VIO 50-293/98-08-02)** of 10 CFR 50, Appendix B, Criterion III, "Design Control."

At the close of the inspection report period, the licensee corrected the alarm response procedure to specify that operators initiate CRHEAFS immediately when high radiation was detected in the control room intake. In addition, an engineering evaluation was performed that concluded that the dose to control room personnel would remain below the GDC 19 limits assuming a time delay of 30 minutes post receipt of the radiation alarm for CRHEAFS initiation. During a September 11, 1998, telephone call, licensee engineering management stated that to identify and correct other possible system design errors, analysis weaknesses, and procedure inconsistencies, a design basis information review effort of the CRHEAFS would be performed.

Based on the review of the engineering evaluations, corrective actions taken, and a walkdown of the CRHEAFS, the CRHEAFS remains operable and capable of performing its design function. Therefore, item III.D.3.4 of NUREG 0737 is considered **closed**.

c. Conclusions

The NRC and the licensee's engineering staff identified deficiencies in the design of the control room high efficiency air filtration system, i.e., seismic qualification of duct supports, single failure design analyses, and inconsistencies between analysis assumptions and operational procedures. The deficiencies have been reported in two LERs, including acceptable short term corrective actions. The failure to accurately transfer the system design into instructions, procedures or drawings was a violation of 10 CFR 50 Appendix B Criterion III "Design Control."

E2.2 (Closed) VIO 98-02-07: Inadequate Design Control

The violation related to self disclosing events in late 1997 when the "A" SSW pump motor shaft failed twice. The cause was human error when engineers performing a substitution equivalency evaluation did not include the actual configuration of the pump pedestal which had been previously modified. As a result, motor misalignment was created that resulted in shaft failure. The maintenance staff returned the pump pedestal to its original configuration and replaced the shaft. No further alignment problems were experienced. Engineering management discussed this event with all engineering personnel to review the lessons learned. This violation is **closed**.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) IFI 97-02-03: Jet Pump Nonconforming Conditions

During inspections of the reactor core internals during the last refueling outage (i.e., RFO11), the licensee identified that several jet pump swing gate latch pins were not fully engaged and some restrainer bracket set screws were not in contact with the jet pump mixer. The licensee hired a consultant who completed a detailed structural analysis of the jet pumps. This evaluation concluded that the structural integrity was assured during design basis conditions. An NRR safety evaluation is attached to this report as Attachment 1. As stated in the NRR evaluation, the licensee plans to conduct additional jet pump inspections during RFO12 to confirm that the aforementioned conditions have not changed. The inspector confirmed through discussions with licensee engineering managers that they are still planning to conduct this inspection. The inspector determined that the licensee met the intent of NRC Generic Letter 91-18 during resolution of this issue. This item is **closed**.

IV. PLANT SUPPORT

P3 EP Procedures and Documentation

A number of specific revisions to EP procedures, listed in Attachment 2 to this inspection report, were examined during an in-office review. Based on the licensee's determination that the changes do not decrease the overall effectiveness of the emergency plan and after limited review of the changes, we determined that no NRC approval is required. Implementation of these changes will be subject to inspection in the future to confirm that the changes have not decreased the overall effectiveness of your emergency plan.

S1 Conduct of Security and Safeguards Activities

S1.1 Control of Special Nuclear Material

a. Inspection Scope (71710)

The inspector reviewed the circumstances surrounding the July 29, 1998, report of missing special nuclear material (SNM). Four intermediate range monitors (IRMs) were not able to be located during a semi-annual audit of SNM.

b. Observations and Findings

The IRMs had been removed from the reactor core and according to the material transfer forms were placed in the spent fuel pool (SFP) in a container in the November-December 1997. Each IRM contains approximately 0.008 grams of total uranium, which has approximately 0.05 microcurie of radioactivity.

The licensee executed a search plan of the spent fuel pool and placed a hold on the removal of the radioactive waste storage facility. The licensee searched the immediate area of the SFP and moved several fuel bundles in an attempt to locate the container. The search resulted in the discovery of two of the four IRM detectors and several pieces of IRM cabling. A search of the radioactive waste storage facility resulted in the location of the remaining two IRMs.

During discussions with the reactor engineering supervisor he stated that the apparent cause for the misplacement of the SNM was personnel error and insufficient care in the movement/storage of material. During a forced shutdown in December 1997, a number of IRM detectors were replaced. When transferring the IRMs to the SFP, technicians noticed that several pieces of IRM cable were mixed in with the detectors. This material was removed and the detectors were placed in the SFP and the cable pieces were transferred to the radwaste storage facility. An error occurred at this point and 2 of the 4 IRM detectors were sent to the radwaste storage facility. The engineer in charge of the transport of the SNM did not verify removal of the IRMs from the bucket to the SFP. The engineer was under the assumption (based on the pre-job brief) that the transfer bucket was to contain only four IRMs. The engineer verified, from the SFP bridge, that four items had been transferred from the transfer bucket to the SFP storage bucket. The instrumentation and control technician (I&C) performing the transfer failed to properly verify the transfer of the IRMs. The inspector determined that the failure of the licensee to properly control the IRM material transfer is a violation of 10 CFR 70.51(b)(1).

Corrective action planned for this issue include: changing the method of removal of IRMs from the core to minimize the need to identify detectors from cabling, train I&C personnel on the event, and change the method of storage of IRMs to prevent loss of material should the storage container be mishandled. The inspector reviewed root cause evaluation and corrective actions and determined that they were acceptable. This non-repetitive licensee identified and corrected violation is being treated as a Non-Cited Violation (**NCV 50-293/98-07-03**), consistent with Section VII.B.1 of the NRC Enforcement Policy. The inspector verified that the licensee had considered the reportability criteria delineated in 10 CFR 70.52 and 10 CFR 20.2201. No NRC notifications were required.

c. Conclusions

The circumstances leading to the misplacement of the intermediate range detectors represented a weakness in the accountability and control of special nuclear material. The licensee was successful in locating the missing intermediate range monitors. The corrective actions to resolve the issue were determined to be appropriate.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspector presented the inspection findings to members of the licensee management after conclusion of the inspection on September 29, 1998. The licensee acknowledged the findings presented.

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 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

Opened

VIO 50-293/98-07-02	Control Room High Efficiency Air Filtration System

Closed

LER 50-293/97-04-01	Loss of Preferred Power and Oil Spill Due to Main Transformer Fault While Shut Down.
LER 50-293/98-06-01	Failure to Verify compliance with Technical Specification (TS) Limit
LER 50-293/98-08	A Section of CRHEAFS Seismic Class 1 Ductwork found with Class II
LER 50-293/98-16	Control room Ventilation Exhaust Damper and Ductwork not Seismically Qualified
NCV 50-293/98-07-01	Failure to Verify compliance with Technical Specification (TS) Limit
NCV 50-293/98-07-03	Loss of Special Nuclear Material
IFI 50-293/97-02-03	Jet Pump Nonconforming Conditions
VIO 50-293/98-02-03	Degraded Equipment Problem Identification
VIO 50-293/98-02-07	Inadequate Design Control

LIST OF ACRONYMS USED

AOT	Allowed Outage Time
ASME	American Society of Mechanical Engineers
BECo	Boston Edison Company
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRHEAFS	Control Room High Efficiency Air Filtration System
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GE	General Electric
HPCI	High Pressure Coolant Injection
IFI	Inspection Follow-Up Item
IR	Inspection Report
IRM	Intermediate Range Meters
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MG	Motor Generator
MR	Maintenance Request
MSTP	Master Surveillance Tracking Program
NCR	Non-conformance Report
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PNPS	Pilgrim Nuclear Power Station
PR	Problem Report
RHR	Residual Heat Removal
SBGT	Standby Gas Treatment
SFP	Spent Fuel Pool
SNM	Special Nuclear Material
SSW	Salt Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VIO	Violation
WWM	Work Week Manager

ATTACHMENT 1

TIA RESPONSE
REGARDING DIFFERENCES IN THE "AS DESIGNED" AND "AS FOUND" CONDITIONS AT THE PILGRIM NUCLEAR POWER STATION

BACKGROUND

During the April 1995 Refueling Outage (RFO) 10, the jet pumps at the Pilgrim Nuclear Power Station (PNPS) were inspected for evidence of tack weld cracks as recommended by General Electric (GE) Services Information Letter (SIL) No. 574. While no weld cracks were identified, gaps were observed between the set screws and inlet-mixers on several pumps. The original jet pump design required that the set screws be in contact with the inlet mixers. An operability evaluation performed at the time of the inspection concluded that no corrective action was required prior to startup.

In June 1996, GE apprised the owners of BWRs that jet pump restrainer bracket set screw gaps had been observed in a number of GE BWRs, and could be present in any BWR with jet pumps. The GE Rapid Information Communication Service Information Letter (RICSIL) No. 078 advised that, "These gaps reduce the pump lateral support near the bottom of the jet pump inlet-mixer, and increase the potential for flow induced vibration of the inlet-mixer." The GE communication noted that, "If significant set screw gaps are present, a plant specific analysis may be required, and has been performed for some pumps, to determine the acceptability of continued operation." The RICSIL further advised that, at one BWR, gaps had been observed to form on one pump, increase in size on another pump, and decrease in size on another pump over one operating cycle; and that the root cause of how the gaps are formed is not known.

While preparing for a vessel inservice inspection during the February 1997 RFO-11, a review of RFO-10 video tapes revealed incomplete engagement of the swing gates on jet pumps numbers 5 and 11. This discovery resulted in the inclusion of visual inspections of all jet pump swing gates and set screw gaps in the RFO-11 inspection scope.

The inspections conducted during RFO-11 revealed that some of the swing gate to restrainer bracket latching pins were not fully engaged in the latch pin holes. Only two new swing gates, for pumps 5 and 11, had been procured for the outage, and the discovery of the remaining incomplete latched swing gates left no time to obtain additional replacement swing gates. However, the gates were secured to the restrainer bracket by two captured and torqued gate keeper bolts. Also, various jet pumps did not have their restrainer bracket set screws in full contact with the inlet-mixer. Boston Edison Company (BECo) completed a Safety Evaluation (SE) that concluded that it was acceptable to leave the conditions "as-found." BECo's SE included qualification analyses, performed by Structural Integrity Associates, Inc. (SI), as recommended in GE's RICSIL No. 078.

REVIEW REQUEST

By letter dated April 22, 1997, Region I, Division of Reactor Projects, forwarded the subject Task Interface Agreement (TIA) and requested NRR's assistance in determining the acceptability of BECo's conclusion that no unreviewed safety question (USQ) existed as a result of differences in the "as designed" versus "as found" conditions in the jet pumps at PNPS. BECo prepared SE No. 3084 dated April 2, 1997, regarding the differences between the "as designed" and "as found" features of the jet pumps. The Region requested that NRR's review of the SE include an assessment of the schedule and extent of appropriate future inspections of the jet pump affected areas.

REVIEW

Jet pumps are part of the reactor coolant recirculation system and are located in the annulus region between the core shroud and reactor vessel wall. PNPS has 20 jet pump assemblies arranged in pairs. Each jet pump assembly pair consists of a common riser pipe, a riser brace, a transition piece, two inlet mixer assemblies, two inlet nozzles, and two tailpipes connected by an integral restrainer bracket. The inlet mixers exit into separate, non-integral diffuser assemblies. The riser pipe is connected to the recirculation inlet nozzle thermal sleeve by way of a 90 degree short radius elbow. The riser pipe is supported near the inlet mixer by the riser brace which is welded to the riser pipe and to pads attached to the reactor vessel wall. The entrance end of each inlet mixer assembly is clamped mechanically to the riser transition piece by a bolted beam assembly. The restrainer bracket assembly provides lateral stiffness to a pair of jet pump inlet mixer assemblies. The restrainer bracket assembly provides a nonintegral three point connection for each pair of inlet mixer tailpipes. Each restrainer assembly consists of a stationary bracket welded to the riser pipe and a clamp arrangement made up of a swing gate and latch mechanism which encloses an individual inlet mixer tailpiece. Two set screws are attached to a bracket and, through contact with the inlet mixer, adjust the position of the inlet mixer tailpiece with respect to the diffuser. The swing gate is latched to the restrainer bracket by a pin and is secured to the bracket by two gate keeper bolts pretensioned to 40-45-foot pounds. The swing gate is provided with an adjustable wedge which assures the three point contact between the inlet mixer tailpiece and restrainer. Instrument lines are attached to the jet pumps to sense jet pump flow and provide indication of recirculation flow mismatch and water level monitoring. The jet pump assembly, although a non-ASME Code component, is safety-related because the structural integrity of the assembly assists in maintaining core refloodability following a postulated Loss of Coolant Accident (LOCA) and thus, is essential for safe shutdown.

BECo contracted SI to perform analyses to assess the jet pump frequency response and structural impact changes resulting from differences between the "as designed" and "as found" conditions. The purpose of the analyses was to determine whether the set screw to inlet mixer gaps and incomplete engagement of the restrainer bracket swing gate latches would have a detrimental effect on the operation of the jet pump with regard to increased vibration or structural degradation.

The original design and analyses of the PNPS jet pump assembly could not be located in BECo's or GE's archives. GE provided design basis loading conditions and a new set of seismic response spectra based on the recent re-analysis of the reactor vessel shroud. The loading conditions were reconstructed and updated to reflect current criteria based on a review of the FSAR, drawings, and the reactor vessel analysis of record.

The SI analyses were based on a two-dimensional, jet pump finite element model representing the piping, elbows, inlet mixers and diffusers. The model was fixed at the recirculation inlet nozzle and laterally and vertically restrained at the riser brace by a flexible member attached to the pressure vessel wall. The restrainer bracket assembly was modeled as a series of massless beams that linked the riser pipe to the inlet mixers to restrict horizontal motion but allow rotation and vertical movement. The model represented the "as designed" condition. The "as found" condition model removed the beams that linked the riser pipe to the inlet mixer. Analyses were performed using the above described model to determine the effects on jet pump operation under "as found" conditions as compared to "as designed" conditions.

The results of the frequency analysis show that the "as found" condition with set screw gaps reduces the fundamental frequency of the pump assembly to 26.4 Hz as compared to the original design fundamental frequency of 34.3 Hz. Both frequencies are not within the region of resonance with the pump blade passing frequency of 139 Hz or a primary harmonic or sub-harmonic of that frequency.

The structural analyses performed by SI compared the "as designed" and "as found" conditions with regard to differences between load distribution, stress, and deformation. The structural evaluation determined that the structural integrity of the jet pumps with set screw gaps and a dislodged restrainer bracket swing gate remains within acceptable loading and the specified stress limits. All reported stress limits are within the ASME Code allowable limits. The analyses show that the stress levels for the "as designed" and "as found" conditions are very similar. The stress levels and fatigue usage factors for the jet pumps remain effectively unchanged. SI performed analyses on the capability of the two gate keeper bolts, torqued to 40-45 foot pounds, to hold the swing gates in place. SI concluded that an applied torque of 40 foot pounds is adequate for design conditions. BECo's SE concluded that, as a design consideration, the gate bolt torque will provide adequate restraint for normal and transient operating conditions.

BECo's SE summary concluded that, based on the SI analyses and the RFO-11 inspections, the structural integrity is assured under design basis conditions, and the impact of vibratory loads are minimal as confirmed by 10 years of inservice inspections. No indications of adverse jet pump motions were identified during the RFO-11 inspections, which is consistent with the analysis demonstration of minimal inlet mixer deflections. Therefore, inservice loadings have been ruled out, by BECo, as the cause of the set screw gaps.

EVALUATION

The staff reviewed the information provided in BECo's SE, including the attached SI analyses and report of the RFO-11 inspection results. Based on the information provided, and in consideration of BECo's report that periodic in-vessels inspections over the past five fuel cycles have not detected any fatigue cracking, excessive wear indications, or evidence of damage to the jet pump components, the staff believes that BECo's conclusion that no USQs exist is acceptable.

With regard to future inspections, we recommend that the following considerations be addressed by the licensee within the scope of inspection of jet pump's components affected by the as-found conditions:

1) Based on findings from the RFO-11 inspection of the jet pumps, BECo developed an explanation for the cause of the incomplete swing gate latching that apparently resulted from an incorrect installation performed on the gate replacement for jet pumps 5 and 11. BECo concluded that the root cause of the incomplete latching was the result of either improper installation, or improper fabrication, and that the condition had existed for 10 years. It was further postulated that this provided a reasonable explanation for the lack of metal-to-metal contact between the restrainer bracket set screws and inlet mixers. Although the explanation appears reasonable, except for the experience gained during RFO-11, no evidence from any previous inspections has been provided to verify that the proposed root cause has existed for over 10 years. Further, RICSIL No. 078 reports that at one BWR, gaps have been observed to form on one pump, increase in size on another pump, and decrease in size on yet a third pump over one operating cycle.

2) The lateral restraining force which prevents unrestrained inlet mixer motion is attributed to the swing gate torqued keeper bolts. The torqued keeper bolts provide a compressive clamping force that resists motion. This clamping force exerts a lateral friction force proportional to the compressive force and the selected coefficient of friction. The friction force is calculated under static load conditions. In a vibratory environment the contact area of the compressive force could be subject to wear, which over time would reduce the compressive force and change the coefficient of friction.

The licensee's SE indicated that, over the past 10 years, the data obtained during the RFO-11 inspection represented the only reliable and accurate information on the set screw gap measurements, assessment of the swing gate latching conditions, and the torqued keeper bolts integrity. On this basis, we recommend that the licensee reinspect a sufficient portion of these areas and compare the results with the RFO-11 data as a baseline to verify and support its contention that the conditions have not changed and have existed for over 10 years.

CONCLUSIONS

We conclude that the licensee's determination that no USQ exists is acceptable. Further, our view with regard to the scope and interval of appropriate future inspections, is that the licensee should reinspect sufficient portions of the "as found" conditions of the jet pumps at an interval to be developed and justified, to verify and support their contention that conditions have not changed and have existed over the past 10 years. Results of future inspections should be compared to data from the RFO-11 as a baseline.

ATTACHMENT 2

Emergency Plan Implementing Procedures Reviewed

Procedure Number	Title	Revision No.
#-210	Control Room Augmentation	5A
#-229	TSC/OSC Equipment Operation	4
#-230	OSC Activation and Response	1A
#-231	Onsite Radiation Protection	4
#-240	Emergency Security Organization Activation and Response	7
#-250	EOF Activation and Response	6A
#-251	Offsite Radiation Protection	3
#-252	Facilities Support	5
#-254	Communications Support	1
#-310	Radiation Monitoring Team Activation and Response	2
#-315	Offsite Personnel Monitoring Team Activation and Response	3
#-330	Core Damage	2B
#-410	Evacuation/Assembly	3
#-420	Search and Rescue	1A
#-440	Emergency Exposure Controls	5A