Enclosure 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Docket No.:	50-293
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Inspection Period:	February 25, 1998, through April 18, 1998
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector
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EXECUTIVE SUMMARY

Pilgrim Nuclear Power Station
NRC Inspection Report 50-293/98-02

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers resident inspection for the period of February 25, 1998, through April 18, 1998.

Operations

- Generally good operator performance was noted during the period. A few operator performance issues were noted indicating a minor weakness in control room formality. The inspector identified a failure of an operator to circle a plant parameter that was outside of the expected range for a non-technical specification required instrument. The NRC identified that the technical specification clarifications in the control room copy of the technical specifications were not being updated. Also, the inspector noted a problem with a chart recorder on a back control panel. The licensee identified the failure of field operators to promptly detect a decreasing trend in the standby diesel generator glycol level. (Section O1.1)

- A planned power reduction to 50% power and return to full power was completed well with no noted operator human performance issues. The pre-evolutionary briefing, good communications and the use of a dedicated reactivity manager contributed to the positive operational controls. (Section O4.1)

Maintenance

- Thorough planning and controls were used during a planned downpower for work on the "B" feed water system regulating valve. Surveillance test data was properly evaluated to ensure acceptance criteria were met. (Section M1.1)

- The process for assessing risk associated with scheduled on-line maintenance work activities was good. (Section M1.2)

- Several lower level equipment problems were identified by the inspector in the plant which collectively indicate that some conditions adverse to quality were not promptly identified and corrected. (Section M1.3)

- An inadequate maintenance work plan resulted in damaging an internal cooling coil in the "A" core spray pump motor. The resultant increased work scope cost an additional 530 millirem of radiation exposure and an additional 48 hours of safety system unavailability. (Section M3.1)

Engineering

- Engineers resolved two degraded equipment issues involving the residual heat removal quadrant room cooler supports and a thru-wall pipe leak in the outlet of the "B" turbine building closed cooling water heat exchanger. In both cases, interim corrective actions included physical modifications and demonstrated effective teamwork between engineers and the maintenance staff. (Section E2.1)

- Engineering review of the failure of a salt service water pump shaft did not detect a subtle pump pedestal alignment problem until a second shaft failure occurred. The licensee's root cause evaluation after the second shaft failure addressed the specific concern for the SSW pumps, but did not evaluate the oversight in the design and/or work control process which overlooked a previous modification. (Section E8.3)

Plant Support

- A significant dose reduction activity in the "D" thru "G" condensate demineralizer room was effectively performed. The use of a design change in the resin transfer line to the spent resin storage tank eliminated the need for manual transfer of drums filled with spent resin. The dose rates in the room dropped from 500 - 1000 mr/hr to 10 - 15 mr/hr, which was a significant reduction. In a second dose reduction activity, a portable high pressure water source was used to clean the 51 foot elevation floor drains in the reactor building. (Section R1.1)

- Proper radiological practices were demonstrated by maintenance workers when opening and venting the RCIC pressure detectors and during work on the "B" feed regulation valve. (Section M1.1)

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REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) entered the report period at 100 percent reactor power. Power was reduced to approximately 50 percent on two occasions: once on March 27, 1998, to perform a thermal backwash of the main condenser; and again, on April 18, 1998, to backwash the condenser and make repairs to the "B" feed regulating valve controller. At the end of the inspection period preparations were underway to return the unit to 100 percent power.

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspector conducted frequent reviews of ongoing plant operations. The inspector observed proper control room staffing, effective pre-evolution briefings, and plant behavior was commensurate with the plant configuration and plant activities in progress. Also, during tours of the site, the inspector verified that the licensee properly displayed the proper 10CFR50, Part 19 postings.

The inspector monitored control room activities which included reviewing operator logs and performing detailed control panel walkdowns. Three minor problems were identified. The first problem involved the chart recorder for the off-gas flow/flux tilt monitor on a back control room panel where the paper became disengaged from the gears. Operators quickly repaired the chart recorder. A second problem involved the identification of three outdated licensee technical specification (TS) clarifications in the control room TS log book. The operators discarded the outdated clarifications since a related license amendment (i.e., 196) had been issued in April 1997. Operators initiated a problem report (PR) to document and evaluate this issue. Lastly, the inspector noted unclear operations department management expectations for highlighting abnormal non-TS log readings. Some operators were circling data outside the listed range while others did not. As a result, procedure 2.1.35, Control Room Readings, was enhanced. No problems were found with TS required log data. These three minor problems identified during control room tours were promptly corrected but indicate a minor weakness in control room formality.

On March 12, 1998, an operator on tour identified that the glycol level for the station blackout (SBO) diesel was not visible in the site glass and the SBO diesel was declared inoperable. Coolant was added and the SBO diesel declared operable. This level is recorded daily by operators on tour. Inspector review of the daily logs revealed that the glycol level had been slowly trending down, and on two occasions

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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.

level was recorded at the lower level of the required band. The inspector noted that a leak of approximately 10 to 15 drops per minute had been identified approximately two weeks earlier from the leak-off line on the jacket water pump. The licensee initiated a PR to evaluate and implement corrective actions. Operators missed opportunities to add coolant prior to going low out of the normal band. A written operability evaluation determined that the SBO diesel remained operable when the coolant level was low.

O4 Operator Knowledge and Performance

O4.1 Planned Downpower (71707)

a. Inspection Scope

The inspector monitored portions of a planned power reduction to approximately 50% for a thermal backwash of the main condenser to assess operator performance.

b. Observations and Findings

The nuclear operations supervisor (NOS) reviewed all planned activities with the crew members during a pre-evolution briefing. Good crew participation in the briefing was evident by several questions raised by crew members. The role of several operator license candidates was carefully discussed to ensure proper oversight of their reactivity manipulations which were planned in 5% power increments. Abort criteria during the thermal backwash was established in anticipation of any problems. Also, the nuclear watch engineer (NWE) stressed the importance of proper command-and-control between the crew and the NOS. Lastly, a quality assurance engineer and operator training instructor attended the briefing and observed portions of the power reduction.

During the down power from 100% to 50% power, the inspector observed deliberate and controlled reactivity manipulations using both control rod insertion and reactor recirculation system pump speed changes. A designated reactivity control manager supplemented the normal shift composition and provided good oversight of all reactivity changes. Reactor engineering personnel monitored reactor core thermal limits and interfaced well with crew members. A reactor operator promptly identified fluctuations in the air ejector steam pressure regulator during the power reduction. Actions were taken to evaluate and compensate for the fluctuations in the pressure regulator. No human performance problems were identified by the inspector. Additionally, after completing the thermal backwash of the condenser and several planned maintenance activities, operators returned the unit to full power with no problems.

c. Conclusions

A planned power reduction to 50% power and return to full power was completed well with no noted operator human performance issues. The pre-evolution briefing, good communications and the use of a dedicated reactivity manager contributed to the positive operational controls.

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) IFI 50-293/97-03-01: Locked Valve List

PNPS procedure 8.C.13, "Locked Component Lineup Surveillance," includes criteria to determine which valves are required to be locked in position. Criteria "A" specifies that manual valves which, if found out of position, could defeat a safety-related function, provide an inadvertent flow path, or permit loss of critical inventory, are required to be locked in the required position. The examples listed under criteria "A" referenced process fluid valves (e.g, suction, discharge, and drain valves) as opposed to instrument and pressure switches. The inspector questioned whether the literal interpretation of the procedure could encompass the salt service water (SSW) pressure switches root and isolation valves. Pressure switches (PS-3828A/B and PS-3829A/B) sense low SSW header pressure and provide an input to start a SSW pump when recovering from a loss of AC electrical power condition.

In response to this concern, BECo revised procedure 8.C.13 by including a note which stated that unless shown on a plant and instrumentation drawing, instrument rack isolation and/or instrument manifold isolation valves are not required to be locked and do not fall under any of the criteria listed in the procedure. These valves are verified in their correct position by other means such as tagouts, independent verification, surveillance testing, and system operating procedures. Based on the change made to procedure 8.C.13 that clarified the requirements for locked components, this item is closed. No violation of regulatory requirements were identified.

O8.2 (Closed) URI 97-07-01 and NCV 98-02-01: Unexpected Residual Heat Removal (RHR) Pump Trip

NRC Inspection Report No. 50-293/97-07, dated September 15, 1997, Section M4.1, documented a self-disclosing event where the "C" RHR pump tripped inadvertently during a routine surveillance test. The "C" RHR pump was running in the torus cooling mode prior to the start of surveillance test 8.M.2-2.1.10. A prerequisite step in procedure 8.M.2-2.1.10 required that the "C" RHR pump was not running. Both the nuclear operations supervisor (NOS) and the maintenance engineer performing the test missed this procedural step. The inspector determined that the failure to follow the surveillance test procedure was a violation of technical specification 6.8, "Procedures".

Although the "C" RHR pump tripped during the surveillance, no damage was incurred to the pump. The licensee held a critique at the time of the event and initiated a problem report to evaluate and implement corrective actions. Subsequently, the test was performed satisfactorily with no further problems. The inspector reviewed the corrective actions and determined that they were timely and thorough. This non-repetitive licensee identified and corrected violation is being treated as a Non-Cited Violation (**NCV 50-293/98-02-01**), consistent with Section VII.B.I of the NRC Enforcement Policy. This unresolved item is closed.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Maintenance

a. Inspection Scope (62707)

The inspector observed all or portions of the following work activities:

- 8.7.4.5 "Main Steam Isolation Valve Twice Weekly Exercise"
- 8.M.2-2.6.4 "Reactor Coil Isolation Cooling (RCIC) Steam Line Low Pressure"
- 19702424 Replacement of 3 Way Manifold for High Pressure Coolant Inspection (HPCI) flow transmitter, FT-2358
- 19702945 Replace "B" Feed Regulating Valve I/P transmitter
- P9501376 "B" Turbine Building Closed Cooling Water (TBCCW) Channel Head Replacement
- 19701704 Replacement of "A" Core Spray Pump Flexible Hoses

b. Observations and Findings

The inspector found the work performed under these activities to be professional and thorough. All work observed was performed with the work package and/or procedure at the job site and frequently being referenced. The inspector verified that the proper isolation was established for the jobs and that the appropriate technical specification had been entered prior to the disabling of the equipment. Proper radiological practices were demonstrated by maintenance workers when opening and venting the RCIC pressure detectors and during work on the "B" feed regulation valve.

The inspector noted that the controls in place and the planning for the replacement of the "B" feed regulating valve (FRV) transmitter was very thorough. The I&C department performed a mock-up of the job to minimize the time spent in a high radiation area. In addition, operators were deliberate in gagging and positioning the FRV as not to challenge the plant during the maintenance activity and the retest of the valve. The retest of the valve was performed successfully.

The inspector reviewed portions of the online maintenance to replace the channel head on the "B" TBCCW heat exchanger. Generally, the work progressed as planned and the SSW system interfaces between the TBCCW and reactor building closed cooling water (RBCCW) systems were well understood. During review of the controls at the worksite in the auxiliary bay, the inspector observed a pipe flange and valve (i.e., 29-HO-3887) on a SSW spoolpiece was removed from the system and placed in a bucket with a caution tag attached. The caution tag was rolled up and inserted into the valve handwheel. The tag was for a nonsafety related valve.

The inspector interviewed the maintenance supervisor who indicated that the tag should have been cleared prior to removing the valve from the system. The supervisor contacted the control room and cleared the caution tag from 29-HO-3887. The caution tag related to establishing a drain path for the maintenance work on the channel head. The inspector determined that removal of valve 29-HO-3887 from the system without clearing the caution tag was a minor oversight. The licensee initiated an apparent cause analysis as part of PR 98.0497 which was initiated to document and evaluate the problem. The apparent cause was indeterminate. This failure constitutes a violation of minor significance and is not subject to found enforcement action.

c. Conclusion

Thorough planning and controls were used during a planned downpower for work on the "B" feed water system regulating valve. Surveillance test data was properly evaluated to ensure acceptance criteria were met. The inspector identified one minor oversight during the TBCCW heat exchanger work when a nonsafety related SSW system valve was removed from the system with a caution tag attached.

M1.2 Scheduled and Emergent Work Activities

a. Inspection Scope (62707)

The inspector reviewed BECo's process to asses the impact maintenance activities have on overall plant safety.

b. Observations and Findings

Maintenance on equipment is planned through a rotating 12-week schedule matrix. The matrix lists various combinations of equipment that may be taken out of service during the scheduled work week. BECo has evaluated the risk associated with these combinations of equipment out of service using probabilistic risk assessment (PRA). In addition, as of December 1997, BECo made available the equipment out of service (EOOS) database for employee use to assess the cumulative risk impact associated with taking equipment out of service. As of March 12, 1998, the work control department work week managers, as well as two of the six operating crews had been trained on the use of EOOS. Procedural changes have been made that require scheduled work activities be evaluated using the EOOS database. For the observed work activities, the inspector verified that the EOOS database was used to asses the risk to the plant.

On March 12, 1998, at 4:00 a.m. operators tagged out the high pressure coolant injection (HPCI) system for scheduled maintenance. Shortly thereafter, plant operators found the station blackout (SBO) diesel with no visible glycol in the sight glass and declared the SBO diesel inoperable. The inspector questioned BECo regarding whether the operating crew evaluated the increased risk or safety impact with both of these components out of service, and of the need to return the HPCI system back to service if the risk was determined to be unacceptable. The Assistant Operations Department Manager stated that the Nuclear Watch Engineer (NWE) did a qualitative assessment of the risk increase based on his training and experience as a licenced rector operator, and concluded it to be acceptable. The NWE on watch had not yet received the EOOS training. The inspector questioned the PRA engineer on what is the increase in risk with both the HPCI and SBO diesel out of service and was informed that it was minimal. The risk increase, based on the EOOS model, went from a risk factor of 8.2 to 8.8. This increase in risk is within that allowed by BECo. There is presently no procedural guidance associated with assessing the risk to the plant associated with emergent work activities. The licensee indicated that the remaining operating crews will be trained on the use of EOOS and that EOOS will be a tool available for operators to use when assessing the risk impact for emergent work activities.

c. Conclusion

The process for assessing risk associated with scheduled on-line maintenance work activities was good.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Maintenance Related Inspection Observations

a. Inspection Scope (62707)

During plant tours, the inspector noted several degraded equipment conditions which were reviewed to determine whether the conditions were known by the licensee and entered into the maintenance or corrective action system. Also, the inspector reviewed other recent NRC reports for similar findings.

b. Observations and Findings

During a tour of the spent fuel pool skimmer surge tank area in the 91 foot elevation of the reactor building, the inspector identified an air leak in the air supply line to the spent fuel pool system valves. The leakage was detected audibly and originated at a threaded fitting joint in the air line. No work request tag (WRT) was hanging at the joint indicating the leak was not known. The inspector reported this adverse condition to operations personnel who subsequently initiated WRT 052072 to enter this into the licensee's corrective maintenance log. No immediate operability issues were evident.

In the high pressure coolant injection (HPCI) turbine room, located off of the "B" residual heat removal (RHR) quadrant room, the inspector identified a broken conduit on the turbine end of the HPCI pump. The conduit housed several electrical leads related to a high pressure bearing oil temperature alarm. The inspector expressed concern that the insulation on the leads may become damaged if rubbed against the broken electrical conduit. Problem report 98.9175 and maintenance request 19800702 were initiated to review and correct the adverse condition. The operability determination concluded that the HPCI system remained operable as the electrical leads were for an alarm only function and the insulation was not worn through.

The inspector identified two material condition problems related to the "A" reactor recirculation system motor-generator (M/G) set. The reactor recirculation system was covered as a system monitored by the maintenance rule. First, the oil piping for valve PCV-3730A which supplies oil to bearings was vibrating and the inspector a loose U-bolt pipe support. The inspector informed operations personnel of the adverse condition. An operator tightened the U-bolt nuts on the pipe support which reduced the pipe vibration levels. A second problem identified by the inspector involved potential through wall piping leakage. Although some mechanical joints exhibited minor oil leakage, a pipe tee welded in the line was identified to be leaking. Subsequent inspection by quality assurance personnel confirmed this observation and determined that a defect existed in the toe of a weld. Problem report (PR) 98.0888 was initiated and the corrective maintenance planned for the cycle 12 refueling outage. This leak does not adversely affect operation of the M/G set. The inspector determined that the two degraded equipment conditions were not known by the licensee.

Also in the reactor recirculation system M/G set room, the inspector identified the loss of the operating status lights for the "B" fluid drive pump for the "B" M/G set. The fluid drive pump was operating, but its associated red operating light was not illuminated. This adverse condition existed for several days. The inspector informed operations personnel who corrected the condition by changing the light bulb. The inspector determined that the loss of this local status light was missed by operators on tour.

In addition to the identification of the above adverse material equipment conditions during this inspection period, the inspector reviewed several recent NRC inspection reports for similar NRC identified problems. NRC Inspection Report (IR) 50-293/98-01 documented mechanical joint leakage downstream of standby liquid control valve 1101-49 as evidenced by the build-up of boron residue. NRC IR 50-293/97-13 documented a partially clogged air cooling inlet screen on the safety related "A" RHR pump. Also, a missing cotter pin was identified in the mechanical piping snubber supporting the west scram discharge volume small bore piping. Lastly, NRC IR 50-293/97-07 documented loose fasteners on a safety related HPCI system pipe support. None of these degraded equipment conditions had been identified by the plant staff.

These NRC observations cumulatively represent a failure by the plant staff to assure that conditions adverse to quality are promptly identified and corrected via an established quality assurance process. The failure to identify and promptly corrected the material deficiencies and non-conforming conditions stated above, is contrary to 10CFR50, Appendix B, Criterion XVI, "Corrective Action," and is a violation. **(VIO 50-293/98-02-03)**

c. Conclusion

A violation was found by the NRC involving several lower level equipment problems in the plant which collectively indicated that some conditions adverse to quality are not promptly identified and corrected.

M3 Maintenance Procedures and Documentation

M3.1 Safety-Related Maintenance Rework

a. Inspection Scope (62707)

The inspector observed online maintenance performed on the "A" core spray pump, P-215A, motor to replace two flexible hoses that connect the RBCCW cooling water to the motor.

b. Observations and Findings

Mechanics replaced the hoses from the RBCCW system to the P-215A motor in accordance with the work package instructions. Caution was used not to exceed the minimum bend radius of the hoses that was specified in the work package. After the physical work was completed, the tagout was cleared and the flexible hoses pressurized from the RBCCW system. In preparation for the post work test (PWT), an electrician identified a mixture of oil/water leaking from the P-215A motor. The PWT to run the pump was placed on hold and maintenance troubleshooting revealed that an internal copper coil in the top of the motor became damaged during replacement of the flexible hoses. Water leakage from the internal motor cooling coil displaced lubricating oil contained in the upper motor. A maintenance supervisor initiated PR 98.9164 to document and evaluate the problem. An apparent cause analysis was initiated but was not completed by the end of this inspection period.

The work scope was modified to disassemble the top of the motor and replace the damaged copper cooling coil. In addition, a boroscope inspection of the motor internals was performed and electricians measured the insulation resistance to confirm that the oil/water mixture did not adversely effect the motor electrical windings. No damage was noted. The new cooling coil was installed and passed the PWT. Approximately 530 additional man rem of radiation exposure was used to complete the increased work scope of replacing the motor cooling coil. The rework also resulted in 48 addition hours of safety system LCO time.

Discussions with the system engineer revealed that when the mechanics installed the new hoses, the lines from the motor were turned to obtain the proper bend radius. Unknown to the mechanics at that time, the line was not a slip fit and any movement transferred directly to the internal cooling coil. The inspector reviewed the work package and the vendor manual and noted that the work package provided instructions on the minimum bend radius, but did not provide instructions on how to make the adjustments. The vendor manual did not clearly illustrate the internal cooling coil and how the coil protrudes thru the motor casing; however, a vendor service information letter contained relevant information which may have prevented this event. Also, this work was non-routine and had not been performed in the past at Pilgrim. The inspector noted that no vendor input was solicited in developing the initial work package instructions.

The inspector concluded that the work package deficiency was a violation **(VIO 50-293/98-02-02)** of procedure 1.5.20, "Work Control Process," step 7.4.2[5], which requires that work plan details shall be commensurate with the complexity of the task.

c. Conclusions

The NRC found a violation involving an inadequate work plan for LCO maintenance that resulted in damaging an internal cooling coil in the "A" core spray pump motor. The increased work scope resulted in an additional 530 mr of radiation exposure and 48 hours additional safety system unavailability time.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 RHR Room Cooler Seismic Inadequacy; (Closed) LER 50-293/98-05 Inadequate Design of RHR Cooler Supports

a. Inspection Scope (37551)

The inspector reviewed the engineering staff involvement in the evaluation and resolution of inadequate RHR cooler supports.

b. Observations and Findings

One non-conforming equipment condition involved the RHR quadrant room coolers (VAC-204A&B) which function as a RHR support system as defined in NRC Generic Letter 91-18. Each RHR train has two coolers with one stacked on top of the other. On March 31, 1998, BECo engineering personnel identified a design deficiency with the support feet of the RHR coolers. This could cause the coolers to fail during a seismic event which would render the safety related equipment in the room inoperable due to overheating. Engineers initiated problem report 98.9160 and informed operations personnel who declared the coolers inoperable and placed the plant in a 24 hour shutdown LCO. BECo initiated a 10 CFR 50.73 review and determined that an LER would be required to be reported to the NRC.

As corrective action, engineering personnel developed temporary modification (TM) 98-06 to add additional cooler supports to limit potential movement during a seismic event. The inspector noted that engineers worked closely with the maintenance staff during the implementation phase. The coolers were subsequently declared operable and the 24 hour shutdown LCO was exited. The inspector notes that these corrective measures were interim and long term corrective actions are planned for the next refueling outage (i.e., RFO12) to restore full conformance with FSAR requirements.

The inspector visually examined the TM made to the RHR coolers. No problems were noted. Also, an in-office review of licensee event report (LER) 98-05, " Supports For RHR Quadrant Coolers Were Found Not To Be Seismically Qualified," was performed. The LER met the reporting requirements of 10 CFR 50.73 and is considered closed. BECo determined the root cause was inadequate original design of the seismic stability of the area coolers. The licensee determined that no adverse safety consequence resulted from this design deficiency. The inspector noted that the postulated failure of the cooler supports would require a seismic event combined with a loss of coolant accident which was of low probability. Further, the inspector noted that there were various attachments, such as ventilation ducting, to the coolers which mitigate the potential failure mode. There was also other equipment nearby which would limit the postulated movement of the coolers.

This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation **(NCV 50-293/98-02-04)**, consistent with Section VII.B.I of the NRC Enforcement Policy. The inspector determined that the engineering staff identified a significant nonconforming condition, followed TS requirements, and implemented timely and sound interim corrective actions with the final resolution planned for RFO12.

E2.2 SSW Piping Integrity Issues

a. Inspection Scope (37551)

A review was performed of through wall piping weld leakage in the SSW side outlet piping from the TBCCW heat exchanger.

b. Observations and Findings

During this inspection period, the engineering staff evaluated two small (1/8 inch in diameter) through wall pipe leaks in the discharge side of the "B" TBCCW heat exchanger. Operators identified the through wall pipe leaks on March 22, 1998, and initiated PR 98.9143. A written operability evaluation was performed using the guidance contained in NRC Generic Letter 90-05, "Guidance For Performing Temporary Non Code Repair of ASME Code Class 1,2 and 3 Piping." The inspector independently examined the orientation of the two through wall leaks and identified no other similar leaks. The operability evaluation concluded that the structural integrity of the piping remained intact and the RBCCW system was operable. The licensee briefed the NRC and submitted a written repair plan to the NRC which was subsequently approved. The repair plan included interim corrective actions to weld a metal patch over the leaks. Longer term corrective actions include replacement of the piping spoolpiece during RFO12. Quality assurance personnel conducted augmented inspections of five other locations by ultrasonic tests (UT). No additional wall thinning problems were found. The inspector determined that the licensee followed the regulatory guidance provided in NRC Generic Letter 90-05 and implemented an interim modification that welded on a metal patch.

c. Conclusions

The NRC concluded that engineers resolved two degraded equipment issues involving the RHR quadrant room cooler supports and thru wall pipe leakage in the outlet of the "B" TBCCW heat exchanger. In both cases, prompt interim corrective actions included physical modifications which demonstrated effective teamwork between engineers and the maintenance staff.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) LER 50-293/97-10-01: Postulated Pipe Break Pressure Relieving Pathway Boundary Door-9 Found Open During Operation.

This LER documented that a high energy line break (HELB) door, that forms part of the pressure relieving boundary for the high pressure coolant injection (HPCI) system turbine steam supply pipe break (door-9), was found open at power. Investigation revealed that the door had been removed for a maintenance activity during the 1997 refueling outage. Upon completion of the work activity, the door was reinstalled but not closed. The placard requiring NWE permission to open the door had been removed and was not re-attached after the maintenance activity. The door was open for four days with temperature and pressure above 200°F and 275 psig respectively. If a HELB were to occur during this four day period that door-9 was open, the equipment located in the auxiliary bay could have been affected.

Corrective actions taken included shutting and stenciling the door to indicate its function. BECo also committed to develop a program to identify and control compartment barriers. The inspector conduced an on-site review of the LER and visually verified that the door was shut and labeled appropriately, and verified that BECo was in the process of developing a program to control compartment barriers. The failure to adequately maintain the design of the plant was a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." 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 50-293/98-02-05**). This LER is closed.

E8.2 (Open) VIO 98-02-07: Inadequate Design Control On SSW Pump Motor Shaft; (Closed) URI 50-293/97-11-03: Salt Service Water (SSW) Motor Shaft Corrective Actions

a. Inspection Scope (92902)

The inspector reviewed the cause and corrective actions taken for the SSW motor shaft breakage. The motor shaft broke on two occasions within 2 months of each other.

b. Observations and Findings

The first SSW motor shaft broke on August 26, 1997, after being in service for about 18 months. Investigation by BECo included disassembly and inspection of the motor shaft gib key and keyway, and the fracture surface for any signs of deformation that would result from torsional/shear loading; none was found. A review of the pump history was also performed and did not identify any similar history of motor shaft failure.

During removal of the SSW motor shaft, BECo identified that the coupling between the motor and pump shaft was not fully engaged; and the threads on the motor shaft were bottomed out. Discussions with the pump vendor revealed that shaft fatigue failures have been observed and were attributed to additional flexing/bending when the motor and pump shaft ends do not butt-up inside of the coupling. Based on the investigation of the SSW motor shaft failure, BECo reworked the replacement motor shaft to add additional threads to the coupling end to ensure the motor and pump shafts butt-up upon assembly. A liquid penetrant test of the shaft and keyway area was then performed to verify that there was no pre-existing cracks in the replacement shaft. The pump was returned to service and vibration reading were taken and verified within the specified acceptance criteria.

The replacement SSW motor shaft failed 54 days later after being placed into service. BECo investigation into this failure identified improper alignment between the SSW pump and motor support stand as a result of a previously performed modification. This resulted in misalignment between the pump motor and pump which caused excessive shaft deflection and subsequent fatigue failure. The improper fitup was not identified during the maintenance activity since the maintenance procedure and vendor manual did not have any discussions on assembly alignment checks for the motor to the pump. The SSW pump is a vertical pump and has a hollow shaft motor; proper alignment is based on the tongue and groove fit between the pump motor and stand. BECo subsequently developed a method to check alignment.

The "A" pump motor was replaced with a new motor per Field Revision Notice (FRN) 94-02-27 to the electrical standing plant design change in 1994. A review of the FRN indicated that the unique configuration of the "A" pump motor stand was not addressed in the package. Weld pads had been installed to take up clearance due to metal loss (corrosion) from the old pump motor during a previous overhaul. When the motor was replaced per the FRN, the normal dimensions were restored and the weld pads did not allow proper fitup. The existence of the weld pads was identified on the drawing, but the drawing was not used during the motor replacement.

The BECo root cause evaluation identified the cause of the shaft fractures was the failure to remove spacer weld tabs on the motor stand during installation of the new pump motor. The corrective actions taken by BECo to address the issue included revising the SSW maintenance procedure to include a method to check SSW pump alignment. The inspectors review of the evaluation identified that the licensee did not addres. vhy a design change to the pump pedestal had been overlooked in the design and/o work control process when installing the new motor. The failure to consider the actual configuration of the SSW pump motor and pedestal was a violation of design control requirements pursuant to 10CFR50, Appendix A, Criterion III, Design Control. **(VIO 50-293/98-02-07).**

c. Conclusion

Engineering review of the failure of a salt service water pump shaft did not detect a subtle pump pedestal alignment problem until a second shaft failure occurred. The licensee's root cause evaluation after the second shaft failure addressed the specific concern for the SSW pumps, but did not evaluate the oversight in the design and/or work control process which overlooked a previous modification.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry Controls

R1.1 Dose Reduction Efforts

a. Inspection Scope (71750)

A review was performed to assess the results from two radiological dose reduction efforts involving the "D" thru "G" condensate demineralizer room and the 51 foot elevation floor drains located in the reactor building.

b. Observations and Findings

The clean-up of the "D" thru "G"condensate demineralizer room focused on the removal of eight barrels of spent resin stored in the room which were reading 3 to 5 rem/hour on contact. The general area radiation dose rates in the area were 500 to 1000 mr/hour. In the past, resin had to be loaded into barrels for manual transfer to the spent resin storage tank. This method of transferring resin was both time and dose intensive. During refueling outage no. 11 in early 1997, the licensee implemented a design change, field revision (FRN) 96-10-165. This FRN added two new connections in the condensate demineralizer spent resin transfer line which allowed pumping resin into the transfer line thus eliminating the need for manual drum transfer.

During this inspection period, the drums filled with spent resin were pumped into the new connections for the transfer line to the spent resin storage tank. The work was completed with no problems and under the ALARA dose estimates for the work. The ALARA review planned 100 hours of work resulting in 1.085 REM of exposure. The actual work was completed in 86 hours and resulted in only 0.317 REM of exposure. The general area dose rates in the "D" thru "G" condensate demineralizer room dropped from 500 - 1000 mr/hour to 10 - 15 mr/hour which was a significant reduction in source term for the room. Dewatering the barrels using the new transfer line connections, instead of the old manual drum transfer method, resulted in completing the work in less time with only 1/3 of the planned radiation exposure.

Also during plant tours, the inspector observed another dose reduction activity involving the floor drains on the 51 foot elevation of the reactor building. A portable machine (i.e., hydrolazer) was used to inject high pressure water into the floor drains to reduce the build-up of radioactive crud. Radiological protection technicians measured the floor drain piping dose rates to determine the effectiveness of the dose reduction activity. The dose rates in the piping dropped considerably.

c. Conclusions

A significant dose reduction activity in the "D" - "G" condensate demineralizer room was effectively performed. The use of a design change in the resin transfer line to the spent resin storage tank eliminated the need for manual transfer of drums filled with spent resin. The dose rates in the room dropped from 500 - 1000 mr/hr to 10 - 15 mr/hr which was a significant reduction. In a second dose reduction activity, a portable high pressure water source was used to clean the 51 foot elevation floor drains in the reactor building.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspector presented the inspection findings to members of the licensee management after the conclusion of the inspection on May 8, 1998. The licensee acknowledged the findings presented.

X3 Management Meeting Summary

Mr. Curtis Cowgill, the NRC Region I Branch Chief, visited the site on March 25 and 26, 1998, for plant tours, interviews with members of the licensee workers and management, and discussions with the resident inspector staff.

X4 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedure and/or parameter to the UFSAR descriptions.

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspector verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters.

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 40500: Effectiveness of Lice..see 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
Pilgrim Nuclear Power Station
Docket No. 50-293

Opened

VIO 98-02-02	Inadequate work package for core spray pump motor work.
VIO 98-02-07	Inadequate design control on salt service water system pump pedestal.
VIO 98-02-03	Failure to have identified and promptly corrected the material deficiencies and non-conforming conditions

Closed

IFI 97-03-01	Locked Valve List
LER 97-10-01	Postulated Pipe Break Pressure Relieving Pathway Boundary Door-9 Found Open During Operation
LER 98-05	Inadequate Design of RHR Cooling Supports
NCV 98-02-01	Unexpected RHR Pump Trip
NCV 98-02-04	Inadequate Design of RHR Cooler Supports
NCV 98-02-05	Postulated Pipe Break Pressure Relieving Pathway Boundary Door-9 Found Open During Operation
NCV 98-02-06	Salt Service Water Motor Shaft Corrective Actions
URI 97-11-03	Salt Service Water (SSW) Motor Shaft Corrective Actions
URI 97-07-01	Unexpected RHR Pump Trip

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
APRMs	Average Power Range Monitors
BECo	Boston Edison Company
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
EEI	Escalated Enforcement Issue
EP	Emergency Preparedness
EPIC	Emergency and Plant Information Computer
ESF	Engineered Safety Feature
gpm	gallons per minute
I&C	Instrumentation and Controls
IFI	Inspection Follow-Up Item
IR	Inspection Report
LER	Licensee Event Report
MG	Motor Generator
MR	Maintenance Request
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NWE	Nuclear Watch Engineer
PNPS	Pilgrim Nuclear Power Station
PR	Problem Report
RHR	Residual Heat Removal
RP	Radiological Protection
SALP	Systematic Assessment of Licensee Performance
SRO	Senior Reactor Operator
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation
WWM	Work Week Manager

Meeting at NRC Region 1 to Discuss the Pilgrim Nuclear Power Station Design Bases Project

March 10, 1998

AGENDA

- Meeting Objectives
  - Vern Oheim
- Overview of Design Bases Information Program
  Fred Mogolesko
  Chris Wilson
- Overview of Docket Correspondence Review
  Jim Keyes
- Overview of UFSAR Information Upgrade
  Jim Keyes
- Closing Summary
  Vern Oheim

MEETING OBJECTIVES

- Describe and share ideas on Pilgrim's approaches for:
  - Design bases information documentation
  - Docket correspondence review
  - UFSAR information upgrade
- Living process - will be maintained and improved
- Show our plans, schedule, progress to date
- Address any questions regarding the above efforts

DESIGN BASES INFORMATION PROGRAM

Program Vision

Documented, maintained, and well-understood design and licensing bases that are easily accessible and retrievable.

DESIGN BASES INFORMATION PROGRAM

Program Objectives

- DBI is focused to establish confidence that:
  - Pilgrim is operated in accordance with the current plant design and licensing bases
  - The plant physical and functional characteristics are maintained and are consistent with the design bases
  - Systems, structures, and components can perform their intended functions
- Develop an improved understanding of the design bases
- Provide the "why's" of the design bases

DESIGN BASES INFORMATION PROGRAM

Program Objectives (cont.)

- Reconcile identified deviations
- Recover and organize design documents in a central location to support
  - design bases
  - UFSAR
  - Technical Specifications conversion
  - operating procedures
- Design bases documents will be
  - easily retrievable and in a user friendly format
  - maintained up-to-date
  - used and useful

DESIGN BASES INFORMATION PROGRAM

PROGRAM FRAMEWORK

- Program plan, process development, and prototyping
- Data collection and indexing
- DBD preparation and review
- Validation

DESIGN BASES INFORMATION PROGRAM

PROGRAM PLAN

- Plan developed and approved and and will be maintained as a living document
  - incorporating lessons learned from the industry and NRC
  - team assembly
  - developing most effective data assembly process
- Processes have been developed addressing:
  - writer's guide
  - open items
  - validation
- Prototyping of DBDs
  - EDG draft being reviewed
  - RHR expected in May

DESIGN BASES INFORMATION PROGRAM

DATA COLLECTION AND INDEXING

- Document Management Plan
  - Document Collection
  - Document Storage and Retrieval
  - Document Screening and Indexing

DESIGN BASES INFORMATION PROGRAM

DATA COLLECTION AND INDEXING (cont.)

- Document collection
  - primary data sources
    - BECo microfilm library
    - GE archives
    - Bechtel archives
- Storage and retrieval
  - evaluating electronic imaging vs. paper files
    - digital conversion of microfilm library
    - digital imaging and indexing of paper files
    - link digital images to computerized document index

DESIGN BASES INFORMATION PROGRAM

DATA COLLECTION AND INDEXING (cont.)

- Document screening and indexing
  - systematic, directed search
  - screening for applicability
  - enhancement of computerized index

DESIGN BASES INFORMATION PROGRAM

DBD PREPARATION AND REVIEW

- Issue DBD templates
  - captures changes to design bases during the DBI program
  - Priority "A" documents
  - immediately assimilates DBDs into work process
- DBDs prepared using a writer's guide
  - format and content consistency
  - based on reviews of other utility DBD's and writer's guides
  - organizational buy-in
- Scope
  - approximately 50 system DBDs
  - approximately 13 topical DBDs
    - plant level DBD

DESIGN BASES INFORMATION PROGRAM

DBD PREPARATION AND REVIEW (cont.)

- Existing corrective action process will be used for discrepancies
  - tracking and resolution
  - operability evaluations
  - reportability evaluations
- Review and verification
  - design bases information has been correctly and consistently translated from the source documents (NUREG 1397)
  - completeness
  - compliance with the writer's guide format and content

DESIGN BASES INFORMATION PROGRAM

DBD PREPARATION AND REVIEW (cont.)

- Initial validation
  - UFSAR will be updated after each DBD is completed
  - operating procedures will be validated against DBDs
- Publication - the "Vision"
  - electronic documents - easily accessed and retrieved
  - supporting documents will also be retrievable electronically
  - hypertext linking between DBDs as well as supporting documents

DESIGN BASES INFORMATION PROGRAM

VERTICAL SLICE VALIDATION

- DBI program is implementing a validation process that:
  - provides confidence that Pilgrim is operated in accordance with current design and licensing bases
  - plant physical and functional characertistics are maintained and consistent with the design bases
  - SSCs can perform their intended functions
- Process follows NRC Inspection Guide 93801
- Validation will be on a sampling basis
- Feedback important to process and product quality

DESIGN BASES INFORMATION PROGRAM

CURRENT STATUS

- DBD production
  - EDG (draft under team review)
  - RHR (NSSS vendor portion mid-May)
  - HPCI (working)
  - Topicals
    - radiological consequences of accidents
    - LOCA/ECCS
    - anticipated operational transients (w/o control rod withdrawal)

DESIGN BASES INFORMATION PROGRAM

CURRENT STATUS (cont.)

- Remaining schedule for 1998
  - DBDs
    - ADS/main steam
    - Core Spray
    - RCIC
    - Non-NSSS (to be determined)
    - Plant level DBD

DESIGN BASES INFORMATION PROGRAM

CURRENT STATUS (cont.)

- Topicals
  - SWOPI
  - RBCCW
  - anticipated operational transients (w/control rod withdrawal)
  - containment response
  - fuel handling accidents
  - core thermal hydraulic stability
  - DC end device study

DESIGN BASES INFORMATION PROGRAM

CURRENT STATUS (cont.)

- Procedures/processes
  - writers guide (issued)
  - DBD validation (procedure issued)
- Staffing
  - full time dedicated staff
  - 9 engineers (BECo, 1 contractor) representing
    - engineering (all)
    - licensing
    - operations (SRO)
    - maintenance
    - quality assurance
  - new office space created for team

DESIGN BASES INFORMATION PROGRAM

DBI CONCLUSIONS

- Vision
- DBR experts
- Team buy-in
- NUORG anxious for output

DOCKET CORRESPONDENCE REVIEW

OBJECTIVES

- Perform a comprehensive review of the Pilgrim docketed correspondence from 1966 to present
- Identify commitments and information affecting plant processes, design, and the FSAR
- Complete the review by December 31, 1998

DOCKET CORRESPONDENCE REVIEW

PROCESS OVERVIEW

- Document retrieval
  - Retrieve docket file listing from PDR (complete)
  - Retrieve docket file listing from PNPS DCC (complete)
- Review both docket file listings for missing correspondence and retrieve as needed
- Revise correspondence log structure/format (e.g., x-referencing and closure references)

DOCKET CORRESPONDENCE REVIEW

PROCESS OVERVIEW (cont.)

- Develop correspondence review checklist to identify and "tag" correspondence pertaining to the design bases and licensing bases
- Develop data sheets to collect key information from "tagged" correspondence
  - UFSAR
  - Technical Specifications
  - hardware/program/procedure related
  - topical subjects
- Use commitment management procedure as necessary

DOCKET CORRESPONDENCE REVIEW

PROCESS OVERVIEW (cont.)

- Create licensing basis narratives for inclusion in the design basis reports following completion of the correspondence review
  - NUREG 1412, Foundation for the Adequacy of the Licensing Bases
- Enhance commitment management processes to allow highlighting future commitments to NRC as either design or licensing bases commitments (Millstone Lessons-Learned Short Term Action Item #1)

UFSAR INFORMATION UPGRADE

OBJECTIVE

- Ensure the UFSAR is accurate, usable, and up-to-date
- Establish UFSAR information standards

UFSAR INFORMATION UPGRADE

PROCESS OVERVIEW

- Revise key processes supporting UFSAR maintenance (April 1998)
  - update timeliness tightened (complete)
  - PDC process revised (complete)
  - address other processes that can impact UFSAR
- Assign owners to each section of the UFSAR (complete)
- Develop living UFSAR process
  - electronic key word searchable version of UFSAR (June 1998)

UFSAR INFORMATION UPGRADE

PROCESS OVERVIEW

- Define standard for UFSAR format and information content (June 1998)
  - information that describes the plant
  - licensing commitments, NRC SERs and exemptions
  - information that presents design bases and operation limits
    - limits, minimum values, nominal values, analytical values
- Enhance organization personnel knowledge of UFSAR contents (July 1998)
- After standards are established and understood, revise FSAR using information from docket review
- Keep abreast of NRC effort to revise UFSAR information standard per NRC Millstone Lessons-Learned Short Term Item #17 & Long Term Item #18

Pilgrim Has An Aggressive Design And Licensing Program Underway

- In our 12/8/97, 50.54(f) update letter we committed to:
  - Formal DBI Program
  - Docket Correspondence Review
  - Operating Procedure Review to DBD
- Our program will meet these commitments and create a user-friendly, retrievable design basis
- Our program will complete in 2001 and cost approximately $17 million
- We are committed to build a quality program

ENTRANCE MEETING REPORT NO. ______

SUBJECT: Pilgrim Design Basis Project DATE: 3-10-98

	NAME (please print)	TITLE/OFFICE
1.	Len Prividy	Sr. Reactor Engr, USNRC, Region I
2.	Nancy Desmond	Regulatory Relations Mgr Pilgrim
3.	Verl Oheim	General Manager-Tech. Sect. (Pilgrim)
4.	John P. Gerety	Nuclear Engineering Mgr
5.	Fred J. Mogolesko	Design Basis Program Mgr. - PNPS
6.	Chris Wilson	Design Basis Program Engineer
7.	Jim Keyes	Regulatory Affairs PR. Eng (Pilgrim)
8.	Gene Kelly	Systems Branch Chief, USNRC, Region I
9.	Jim Wiggins	DRS Division Director, USNRC, Region I
10.	Curt Cowgill	NRC
11.	Alan Wang	NRC
12.		
13.		
14.		
15.		
16.		
17.		
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19.		
20.		