June 29, 2006

Mr. Michael A. Balduzzi
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC INSPECTION REPORT 05000293/2006006

Dear Mr. Balduzzi:

On May 19, 2006, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim reactor facility. The enclosed inspection report documents the inspection results, which were discussed on May 19, 2006, with Mr. Kevin Bronson and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed Entergy's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two findings of very low safety significance (Green), one of which involved a violation of NRC requirements. However, because of the very low safety significance and because the issue has been entered into your corrective action program, the NRC is treating the issue as a non-cited violation (NCV), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest the NCV in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Pilgrim.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 50-293/06-06
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2006006
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates:	April 10 - May 19, 2006
Inspectors:	J. Schoppy, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader J. Krafty, Reactor Inspector, DRS A. Patel, Reactor Inspector, DRS S. Pindale, Senior Reactor Inspector, DRS B. Parks, Nuclear Safety Professional Development Program (NSPDP) Participant (Trainee) H. Campbell, Mechanical Contractor O. Mazzoni, Ph.D., Electrical Contractor
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000293/2006-006; 04/10/2006 - 05/19/2006; Pilgrim Nuclear Power Station; Component Design Bases Inspection.

This inspection was conducted by a team of four NRC inspectors and two NRC contractors. Two findings of very low risk significance (Green) were identified, one of which involved a violation of NRC requirements. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control. Entergy used a non-conservative calculation method to determine the critical condensate storage tank (CST) water level which would preclude vortex formation at the suction of the high pressure coolant injection (HPCI) pump.

  The finding was more than minor because the formation of vortexing at the intake of the HPCI suction line could result in air entrainment, which in turn, could cause pulsating pump flow and/or reduction in pump performance. It was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. The team reviewed this finding using the Phase 1 SDP worksheet for Mitigating Systems and determined the finding was of very low safety significance (Green), because it did not represent a loss of safety function. (Section 1R21.2.1.1)

- Green. The team identified a finding regarding Entergy’s operability determination for a HPCI trip solenoid valve failure. Specifically, Entergy’s operability evaluation technical basis did not support the specific technical specification (TS) requirement of ensuring that the HPCI system automatically isolates on a reactor vessel high water level signal.

  The finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, Entergy did not ensure HPCI’s continued reliability and capability to isolate automatically as designed during reactor vessel high water level conditions. The team reviewed this finding using the Phase 1 SDP worksheet and determined the finding was of very low safety significance (Green), because it did not represent a loss of safety function for greater than its TS allowed outage time. (Section 1R21.2.1.2)

B. Licensee Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in Entergy's Probabilistic Risk Analysis (PRA), the U. S. Nuclear Regulatory Commission's (NRC's) Standardized Plant Analysis Risk (SPAR) model, and the Significance Determination Process (SDP) Risk Informed Inspection Notebook, Revision 2, for Pilgrim Station. In general, this included components and operator actions that had a risk achievement worth (RAW) of greater than two. The components selected were located within both safety related and non-safety related systems and included a variety of components such as electrical buses, pumps, motors, diesel generators, heat exchangers, transformers and valves.

An initial list, consisting of over 50 components, was created based on risk considerations. The team performed a margin assessment to narrow this list down to 17 components for a detailed design review. This design margin assessment considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. Issues impacting design margin included failed performance test results, significant corrective action history, repeated maintenance, Maintenance Rule (MR) (a)(1) status, operability reviews for degraded conditions, NRC resident inspector input of problem equipment, system health reports and industry operating experience (OE). Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report. Specific documents reviewed are listed in the attachment to this report.

.2 Results of Detailed Reviews

.2.1 Detailed Component and System Reviews

.2.1.1 Condensate Storage Tanks (T-105A & B)

a. Inspection Scope

The team selected the condensate storage tanks (CSTs) because of their function as the preferred source of water for the high pressure coolant injection (HPCI) system pump. The team reviewed the HPCI system design basis document (DBD), the Technical Specifications (TSs), the Updated Final Safety Analysis Report (UFSAR), CST drawings, associated instrumentation, and supporting calculations. The team also performed independent calculations of critical submergence level (vortexing) and instrument uncertainty.

b. Findings

Introduction: The team identified a non-cited violation of very low significance (Green) of 10 CFR 50, Appendix B, Criterion III, Design Control, associated with inadequate calculations for a condition that could have impacted HPCI pump performance under certain accident scenarios.

Description: In the event of a small break loss of coolant accident (LOCA), the HPCI and reactor core isolation cooling (RCIC) systems may receive automatic actuation signals on either low reactor vessel level or high drywell pressure and initially take suction from the CSTs. At a tank level determined to protect the HPCI pump from air ingestion via vortex formation, pressure switches initiate an automatic transfer of suction to the torus. The team observed that Entergy determined that the critical tank level to preclude vortex formation was 14 inches, based on a graph given in a 1969 Oil and Gas Trade Journal article. The team questioned the validity and application of this approach, particularly with respect to the scaling of the test results given in the 1969 article. Whereas the CST diameter is 15 feet and the HPCI suction line has an internal diameter of approximately 17.5 inches; the 1969 article test results were obtained on a 3 foot diameter tank with a 1.5 inch outlet pipe. Consequently, the team performed an independent evaluation of this vortex potential.

The team noted that there are several different approaches addressing the formation of tank vortices reported in trade journals and research publications. However, there is no universally accepted industry standard or calculational methodology available to ensure, with a relatively high probability, that a specific tank-piping-pump combination will not develop vortices and consequential pump degradation. Notwithstanding, the team's three independent approaches and resultant calculations yielded results significantly different from Entergy's. Specifically, at a combined HPCI/RCIC total flow of 4650 gpm flowing through the CST suction line of 17.5 inches internal diameter, the Reddy-Pickford method yielded a critical water depth of approximately 33 inches. Second, the analysis and results given in the Gould Pump Manual gave the required head of water above the suction line at close to 54 inches. Finally, the Hydraulic Institute method (ANSI/HI 9.8-1998) yielded the result that the suction line should be submerged to a depth of approximately 53 inches to preclude vortex formation. Based on the preponderance of calculated results derived from more recent and more readily accepted methodologies, the team determined that Entergy's use of a water depth of 14 inches was not sufficiently conservative to protect the HPCI pump from potential degradation due to vortice-induced air ingestion.

In recognition of the above concern, Entergy personnel initiated condition report (CR) 2006-01699 to evaluate the condition fully and to implement corrective actions to conservatively bound the vortexing concern in the interim. The associated Entergy operability determination directed procedure changes to manually swap the HPCI suction from the CST to the torus at a tank level of eight feet and required operators to maintained both CSTs cross-tied and available (maximizing CST available inventory and minimizing vortexing potential). The team determined that this design control deficiency did not result in a loss of HPCI's safety function based on the actual installed swap-over setpoint (36 inches), normal administratively controlled CST levels, normal operation with cross-tied CSTs, automatic swap-over on high torus level, CST low level alarms (12.5 feet), and available operator action time.

Analysis: The team determined that this issue was a performance deficiency because Entergy used a non-conservative method to demonstrate that the HPCI pump was protected from the potential of vortex-induced air ingestion and subsequent pump degradation. The team determined that this design control deficiency was reasonably within Entergy's ability to identify and correct prior to April 2006 based on related industry OE since 2000.

The finding was more than minor because the formation of vortexing at the intake of the HPCI suction line could potentially result in air entrainment, which could cause pulsating pump flow and/or reduction in pump performance. This finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team used IMC 0609, Appendix A, to determine the risk significance of this finding. Using the Phase 1 SDP screening worksheet, the team determined that the finding was of very low safety significance (Green) because this design control deficiency did not result in a loss of safety function.

Enforcement: Title 10 to CFR Part 50, Appendix B, Criterion III, Design Control, states, in part, that measures shall be established for the selection and review for suitability of application of processes that are essential to the safety-related functions of the structures, systems and components. Contrary to the above, as of April 2006, Entergy failed to select and review for suitability an adequately conservative method for calculating the onset of vortexing at the intake of the HPCI suction line from the CST. Specifically Calculation M501, Minimum CST Level for Transfer of HPCI Pump Suction to Torus, Rev. 0, completed on November 19, 1991, and Attachment M501-0-1, dated November 4, 1999, used a method to calculate the critical submergence depth which yielded a result lower, by factor of at least two, than more rigorous and recent approach methodologies. Because this design control deficiency is of very low significance and has been entered into Entergy's corrective action program (CR 2006-01699), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). **(NCV 05000293/2006006-01, Less Than Adequate Design Control Associated with Potential CST Vortexing)**

.2.1.2 High Pressure Coolant Injection Trip Solenoid Valve (SV-2301-246)

a. Inspection Scope

The team reviewed the design of the HPCI trip solenoid valve (SV-2301-246). The solenoid valve functions to dump auxiliary oil causing the HPCI turbine steam inlet stop valve to close on valid isolation signals. One of the valid isolation signals is a reactor vessel high water level condition. The solenoid valve functions to isolate HPCI to prevent the water level from increasing further which could result in water entering the main steam lines and the HPCI steam supply line. As described in UFSAR Section 7.4.3.2.4, the reactor water level trip function is to protect the HPCI turbine from damage which can be caused by gross carryover of moisture. The team reviewed the maintenance history, design changes, condition reports (CRs), design specifications, drawings, and surveillance tests (STs). The team also reviewed the voltage drop calculations to verify that adequate voltage was provided to the solenoid valve under worst-case conditions.

b. Findings

Introduction: The team identified a finding of very low safety significance involving Entergy's less than adequate operability determination for a HPCI trip solenoid valve failure.

Description: On November 21, 2005, at 1:21 a.m., operators discovered that the HPCI turbine steam inlet stop valve failed to close during a quarterly HPCI test. Entergy performed troubleshooting and determined the cause was a failure of the HPCI turbine trip solenoid valve to function. As a result of the solenoid valve failure, several HPCI turbine protective features were affected, such as the HPCI isolation during reactor vessel high water level conditions. Entergy initiated CR 2005-05040 to evaluate the condition and track corrective actions. Entergy promptly completely an operability determination and declared HPCI operable, based primarily on its continued capability to perform its design basis function (deliver 4250 gpm to the reactor vessel over a range of reactor pressure from 150 psig to 1000 psig). Even though they had considered HPCI operable, Entergy prioritized the trip valve corrective maintenance and removed HPCI from service on November 22 at 12:10 p.m. Maintenance replaced the failed HPCI turbine trip solenoid valve under a priority 1 work order. Subsequently, operators successfully tested the valve and declared HPCI operable on November 23 at 12:39 a.m.

The team determined that Entergy did not adequately evaluate how the inoperable solenoid valve affected overall HPCI system operability. Specifically, TS 3.2.B requires instrumentation to actuate under reactor vessel high water level conditions to automatically isolate HPCI. This instrumentation must be operable whenever the HPCI system is required to be operable as specified in TS 3.5.C. The team noted that the instrumentation would not have completed its function of isolating HPCI on a reactor vessel high water level signal with the solenoid valve failed. In response to the team's questions, Entergy determined that they should have declared HPCI inoperable and entered a 14 day Limiting Condition for Operation(LCO) in accordance with TS 3.5.C. Entergy initiated CR 2006-01460 to address this performance deficiency.

Analysis: The performance deficiency associated with this finding was that Entergy had failed to adequately evaluate the impact of a HPCI trip solenoid valve failure on HPCI system operability. Specifically, Entergy's operability evaluation technical basis did not support the specific TS design feature of the HPCI system to automatically isolate on a reactor vessel high water level signal. The finding was more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, Entergy did not ensure HPCI's continued reliability and capability to isolate automatically as designed during reactor vessel high water level conditions. The team used IMC 0609, Appendix A, to determine the risk significance of this finding. Using the Phase 1 SDP screening worksheet, the team determined that the finding was of very low safety significance (Green) because it did not represent a loss of safety function for greater than its TS allowed outage time. **(FIN 05000293/2006006-02, Failure to Perform an Adequate Operability Determination for the HPCI System)**

Enforcement: The team did not identify any violation of regulatory requirements of significance associated with this finding.

.2.1.3 Automatic Depressurization System Accumulators (Tanks T-221A, B, C, & D)

a. Inspection Scope

The team reviewed the automatic depressurization system (ADS) DBD, piping and instrumentation diagrams, and relevant portions of the UFSAR and TSs. The team reviewed the accumulator leakage and capacity calculations to analytically verify their capability to perform at least 20 actuations of the safety relief valves (SRVs). In addition, the team reviewed completed ADS surveillances and Entergy's response to ADS related industry OE involving potential calculational errors.

b. Findings

No findings of significance were identified.

.2.1.4 Low Pressure Coolant Injection Inboard Injection Valve

a. Inspection Scope

The team selected the B low pressure coolant injection (LPCI) inboard injection valve as a representative sample of safety-related, motor operated valves (MOVs). The valve is normally closed and is required to open to permit injection from the residual heat removal (RHR) pumps into the B recirculation line during a LOCA. The valve is also required to close in order to prevent diverting LPCI flow out of a ruptured recirculation loop or to provide containment isolation. The team reviewed DBDs, piping and instrumentation diagrams, calculations, maintenance requests, CRs, stroke-time and diagnostic test trending data, operational data, and Entergy's responses and modifications resultant to NRC Generic Letters 89-10, 95-07, and 96-05.

b. Findings

No findings of significance were identified.

.2.1.5 C Residual Heat Removal Pump

a. Inspection Scope

The team reviewed the design of the C RHR pump, as presented in the RHR DBD and selected UFSAR sections. This review included system flow, minimum flow capability, and net positive suction head (NPSH) calculations related to the pump's operation under various transient and accident conditions. The team conducted a walkdown of the pump and reviewed recent test results, plant design changes (PDCs), CRs, system health reports, flow instrumentation uncertainty calculations, and other available documentation. The team also evaluated the capability of the RHR system testing and instrumentation to adequately demonstrate TS compliance.

b. Findings

No findings of significance were identified.

.2.1.6 4KV Undervoltage and Degraded Grid Relays

a. Inspection Scope

The team reviewed the undervoltage (UV) and degraded grid relays on the safety-related 4160 volt buses to ensure that variations in voltages during design basis events would not degrade any loads. The team reviewed maintenance history, CRs, ST procedures, and acceptance criteria to verify the TS-required relays were appropriately set. The team also reviewed calculations and drawings to determine if the startup transformer (SUT) and 4160 volt bus protective relaying was designed as described in the UFSAR and tested in accordance with the TS requirements. The team also conducted walkdowns of the relays to determine their material condition and operating environment.

b. Findings

No findings of significance were identified.

.2.1.7 Bussmann Fuses

a. Inspection Scope

The team reviewed the Bussmann fuses in safety system control power circuitry. These fuses were a concern due to a manufacturing defect, as described in NRC Information Notice (IN) 2006-05, and several related fuse failures in Pilgrim safety systems. The team reviewed the status of corrective actions to replace the fuses, test procedures, CRs, acceptance criteria to verify fuse quality, drawings, and Entergy's Bussmann fuse replacement plan.

b. Findings

No findings of significance were identified.

2.1.8 Automatic Depressurization System Solenoid Valve (SV-203-3B)

a. Inspection Scope

The team reviewed the design of the 3B ADS solenoid valve (SV-203-3B) as a representative sample of the ADS solenoid valves. The ADS SRVs are dual purpose valves; the valves are pilot-operated to automatically open at a certain reactor pressure or operators can manually open them from a remote switch. The control system consists of a solenoid valve which controls pneumatic pressure directly to the SRV actuator. The team reviewed the maintenance history, design changes, CRs, design calculations, design specifications, drawings, and STs. The team also reviewed the voltage drop calculations to verify that adequate voltage was provided to the solenoid valve under worst-case conditions.

b. Findings

No findings of significance were identified.

.2.1.9 B Reactor Building Closed Cooling Water Heat Exchanger

a. Inspection Scope

The team evaluated the E-209B reactor building closed cooling water (RBCCW) heat exchanger to assess whether it was capable of removing sufficient heat from the RBCCW system during normal and accident conditions. The inspection consisted of a walkdown of the equipment; interviews with the system and design engineers; and a review of the RBCCW DBD, calculations, system health reports, CRs, and STs. The team also reviewed Entergy's Generic Letter 89-13 response, inspection, maintenance, and testing program. The review of Entergy's Generic Letter 89-13 program included thermal performance testing, visual inspection and cleaning, eddy current testing, weekly backflushing, and differential pressure trending of the salt service water side of the RBCCW heat exchanger.

b. Findings

No findings of significance were identified.

.2.1.10 High Pressure Coolant Injection Pump

a. Inspection Scope

The team evaluated the HPCI pump to assess whether it was capable of providing the required flow to the reactor vessel during a LOCA and when required as a backup to the RCIC system. The inspection consisted of a walkdown of the equipment; interviews with the system engineer; and review of the HPCI DBD, General Electric (GE) LOCA analysis, STs, safety evaluations, inservice testing data, and CRs.

b. Findings

No findings of significance were identified.

.2.1.11 X-107A Emergency Diesel Generator Air Start Motors

a. Inspection Scope

The team evaluated the X-107A emergency diesel generator (EDG) air start motors to assess whether the EDG was capable of meeting its safety function of being able to accept loads within ten seconds of receiving a start signal. The inspection consisted of a walkdown of the equipment; interviews with the system and design engineers; and review of the EDG DBD, calculations, modifications, STs, and CRs. The team also observed the start and portions of the monthly ST on the A EDG on May 10, 2006.

b. Findings

No findings of significance were identified.

.2.1.12 Automatic Depressurization System Safety Relief Valves

a. Inspection Scope

The team evaluated the capacity and setpoint drift of the four two-stage Target Rock SRVs to assess whether they met the design basis of protecting the reactor vessel from overpressure. The inspection consisted of interviews with the system and design engineers; and a review of the ADS DBD, GE calculations and analyses, licensee event reports (LERs), root cause investigations, Maintenance Rule (MR) action plan, and CRs.

b. Findings

No findings of significance were identified.

.2.1.13 4160 Vac Bus A6 and Associated Breakers

a. Inspection Scope

The team selected 4160Vac bus A6 because of its importance to safety system reliability and as a representative sample of safety buses A5 and A6, which showed very low margin relative to their momentary current rating. The team performed a system walkdown and reviewed the associated drawings, DBD, calculations, and STs.

b. Findings

No findings of significance were identified.

.2.1.14 480 Vac Bus B14 and Associated Breakers

a. Inspection Scope

The team selected 480Vac bus B14 because of its importance to safety and relatively low margin to its design loading. The team reviewed the one line diagram (E9, E10), DBD, loading, and protective setting calculations. The team reviewed the adequacy of the bus normal and short circuit ratings, the voltage profile adequacy, and the acceptability of connections to non-Class 1E buses. The team also performed a system walkdown.

b. Findings

No findings of significance were identified.

.2.1.15 A Emergency Diesel Generator

a. Inspection Scope

The team selected the A EDG because of its marginal loading and performance. The team reviewed the associated system drawings, DBD, calculations, and loading. The team also reviewed the adequacy of the bus normal and short circuit ratings, the voltage profile adequacy, the protective relaying, and STs. The team performed several walkdowns of the A EDG and its support equipment. In addition, the team observed the start and portions of the monthly ST on the A EDG on May 10, 2006.

b. Findings

No findings of significance were identified.

2.1.16 Startup Transformer

a. Inspection Scope

The team selected the SUT because of its importance as the preferred offsite power source, operating history, and potential vulnerabilities. The team reviewed the surge protection, the voltage profile adequacy, the protective relaying, and the acceptability of connections to the Class 1E buses. The team performed a system walkdown and reviewed the associated drawings, surge arrester vendor data, DBD, calculations, and STs.

b. Findings

No findings of significance were identified.

2.1.17 Station Blackout Diesel Generator

a. Inspection Scope

The team reviewed the reliability, availability, and capability of the station blackout (SBO) diesel generator and its supporting equipment. The team reviewed the one line diagram (E6, sh 2), DBD, operating procedure, testing, voltage profile, loading, and protective setting calculations. The team also reviewed the adequacy and the acceptability of connections to the Class 1E buses. The team performed several walkdowns of the SBO diesel and observed portions of the SBO testing on May 17, 2006.

b. Findings

No findings of significance were identified.

.2.2 Review of Low Margin Operator Actions

The team performed a margin assessment of expected operator actions, and selected a sample of operator actions for detailed review based upon risk significance and time dependency of the actions. The operator actions were selected from PRA rankings of human action importance based on RAW values and other PRA insights.

Low margin issues were generally characterized as having one or more of the following attributes:

- Low margin between the time required and time available to perform the actions;
- Complexity of the actions;
- Reliability or redundancy of the components associated with the actions; and
- Procedure or training challenges that may impact the operators' ability to perform the actions.

.2.2.1 Starting and Controlling the Suppression Pool Cooling Mode of RHR; Starting and Controlling the Containment Spray Cooling Mode of RHR

a. Inspection Scope

The team selected the manual actions to start and control two specific modes of RHR; suppression pool cooling and containment spray cooling. These manual operator actions are related in that they both affect containment integrity, and failure of these actions can result in containment failure. The Pilgrim PRA assigned a stress level for these actions between nominal and moderately high (due to an assumed simultaneous unavailability of multiple RHR modes of operation).

In order to evaluate the time required to perform the manual actions, the team interviewed licensed operators, non-licensed operators and training personnel. The team performed field, main control room and simulator walkdowns to independently identify operator task complexity. The team evaluated the available time margins to perform the operator actions to verify Entergy's operating and risk model assumptions; and reviewed the applicable procedures. The team also observed a simulator training scenario involving these operator actions.

b. Findings

No findings of significance were identified.

.2.2.2 Venting the Containment via the Direct Hardened (Torus) Vent

a. Inspection Scope

The team selected the manual actions associated with venting the primary containment via the direct hardened vent. The failure of these actions could result in inadequate venting of containment to remove pressure and heat, and could lead to containment failure. The Pilgrim PRA assigned a stress level with these actions as moderately high. While the need to vent the containment is not expected to occur until many hours into the applicable postulated events (over 20 hours), the associated actions are moderately complex and involve exercising several decision points and various alternate venting options prior to establishing the hardened vent option.

In order to evaluate the time required to perform the manual actions, the team interviewed licensed operators, non-licensed operators and training personnel. The team performed main control room and simulator walkdowns to independently identify operator task complexity. The team evaluated the available time margins to perform the operator actions to verify Entergy's operating and risk model assumptions; and reviewed the applicable procedures.

b. Findings

No findings of significance were identified.

.2.2.3 Start and Align the Station Blackout Diesel Generator to Bus A5 or A6

a. Inspection Scope

The team selected the manual actions associated with starting and aligning the SBO diesel generator to Bus A5 or A6. The failure of these actions can result in continued loss of all AC power, and, without subsequent recovery of offsite power, can result in core damage. The Pilgrim PRA assigned a stress level with these actions as extremely high. Further, the time needed to start and align the SBO diesel generator is relatively short (ten minutes).

In order to evaluate the time required to perform the manual actions, the team interviewed licensed operators, non-licensed operators and training personnel. The team performed field, main control room and simulator walkdowns to independently identify operator task complexity. The team evaluated the available time margins to perform the operator actions to verify Entergy's operating and risk model assumptions; and reviewed the applicable procedures. The team also observed a simulator training scenario involving these operator actions.

b. Findings

No findings of significance were identified.

.2.2.4 Align Fire Water Cross-tie to Drywell Sprays; Align Fire Water Cross-tie for Reactor Pressure Vessel Injection via LPCI

a. Inspection Scope

The team selected the manual actions to align the fire water cross-tie for two modes of RHR (drywell spray and LPCI). These manual operator actions are related in that they both provide an alternate source of water to the RHR system, and the failure of these actions can result in containment failure. The Pilgrim PRA assigned a stress level with these actions between moderate to high, and a task complexity of moderate.

In order to evaluate the time required to perform the manual actions, the team interviewed licensed operators, non-licensed operators and training personnel. The team performed a field walkdown to independently identify operator task complexity. The team evaluated the available time margins to perform the operator actions to verify Entergy's operating and risk model assumptions; and reviewed the applicable procedures.

b. Findings

No findings of significance were identified.

.2.2.5 Align Alternate Switchgear Room Ventilation

a. Inspection Scope

The team selected the manual actions to align alternate switchgear room ventilation in the event of a loss of normal room cooling. Several of the PRA sequences involved an internal flooding scenario where the equipment in one of the two rooms is rendered inoperable due to the flooding, and the ventilation fails in the remaining switchgear room. The Pilgrim PRA assigned a stress level with these actions as moderately high, and a task complexity of low to moderate.

In order to evaluate the time required to perform the manual actions, the team interviewed licensed operators, non-licensed operators and training personnel. The team performed a field walkdown to independently identify operator task complexity. The team evaluated the available time margins to perform the operator actions to verify Entergy's operating and risk model assumptions; and reviewed the applicable procedures.

b. Findings

No findings of significance were identified.

.2.3 Review of Industry Operating Experience and Generic Issues

a. Inspection Scope

The team reviewed selected OE issues that had occurred at domestic and foreign nuclear facilities for applicability at Pilgrim. The team performed an independent applicability review and selected issues with apparent applicability to Pilgrim. The team performed a detailed review of the following OE issues:

- GE SC06-01: Worst Case Single Failure for Long Term Torus Heatup

  The team reviewed the GE Safety Communication addressing a potentially new worst case single active failure which could affect long term torus heatup. Specifically, whereas the design basis accident (DBA) LOCA/LOOP with an EDG failure was believed to result in the highest torus temperature, GE notified boiling water reactor (BWR) owners that the above DBA concurrent with loss of a functioning RHR heat exchanger, instead of an EDG, could possibly result in a higher torus temperature. The essential difference in the two cases is that in the new scenario additional pump heat is being added to the coolant without additional heat being removed. The team reviewed Entergy's resultant operability evaluation, compensatory actions, and associated operating procedures.

- NRC Information Notice (IN) 2000-08: Inadequate Assessment of the Effect of Differential Temperatures on Safety-Related Pumps

  The team assessed Entergy's applicability review and disposition of two industry events that appear to have been caused by inadequate engineering design assessment of the effect of differential temperatures on safety-related pumps. In particular, IN 2000-08 communicated the potential impact to safety-related pump bearings with regard to cooling system differential temperatures (e.g., seal water supply) and changes in lubricating oil viscosity.

- NRC Information Notice 97-21: Availability of Alternate AC Power Source Designed for Station Blackout Events

  The team reviewed the applicability and disposition of NRC IN 97-21, Availability of Alternate AC Power Source Designed for Station Blackout Events. The basis of IN 97-21 was a concern for SBO source susceptibility to a failure to start as a result of loss of non-safety related power to the SBO diesel auxiliaries. This issue was selected because the OE described an actual failure at Pilgrim.

- NRC Bulletin 88-04: Potential Safety-Related Pump Loss

  The team reviewed the potential loss of safety-related pumps due to two conditions: dead-heading of a weaker pump through a common minimum flow line and inadequate minimum flow capacity for single pump operation. The inspection consisted of a walkdown of the RHR and core spray pumps; interviews with the system and design engineers; and review of Entergy's

Bulletin 88-04 response, DBDs, calculations, and correspondence from GE, Bechtel, Bingham Willamette, and the Boiling Water Reactor Owners' Group (BWROG).

- NRC Information Notice 2002-12: Submerged Safety-Related Electrical Cables

  The team assessed Entergy's applicability review and disposition of NRC IN 2002-12: Submerged Safety-Related Electrical Cables. The team selected IN 2002-12 due to its potential applicability to the underground cables that connect the SUT to the safety buses. The team reviewed the underground duct drawings and preventive maintenance procedures. The team also performed a walkdown of the underground systems, inspected the inside of the associated SUT cable manholes, and checked the operation of the sump pump.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design and qualification issues. In addition, action requests written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On May 19, 2006, the team presented the inspection results to Mr. K. Bronson and other members of Entergy management. The team verified that no proprietary information is documented in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

B. Ahern	DC System Engineer
A. Biswas	I&C Engineer
G. Bradley	ADS System Engineer
K. Bronson	General Manager Plant Operations
T. Collis	EDG System Engineer
S. Das	Electrical Engineer
P. Doody	Mechanical Engineer
N. Eisenmann	I&C Design Supervisor
B. Ford	Licensing Manager
M. Green	Engineer, Motor Operated Valves
P. Harizi	Mechanical Engineer
K. Kennedy	Senior Operations Instructor
C. Littleton	Lead PSA Engineer
J. Macdonald	Shift Manager
M. McClellan	System Engineer, Motor Operated Valves
D. Noyes	Assistant Operations Manager
R. Pace	Mechanical Systems Supervisor
D. Richardson	I&C Engineer
D. Rydmann	RHR System Engineer
J. Sabino	Mechanical Engineer
E. Sanchez	Licensing Engineer
B. Sullivan	Component and Programs Manager
T. White	Design Engineering Manager

NRC personnel:

W. Raymond, Senior Resident Inspector
C. Welch, Resident Inspector
W. Cook, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open and Closed

05000293/2006006-01	NCV	Less than adequate design control associated with potential CST vortexing. (Section 1R21.2.1.1)
05000293/2006006-02	FIN	Failure to perform an adequate operability determination for the HPCI system. (Section 1R21.2.1.2)

LIST OF DOCUMENTS REVIEWED

Calculations
EN-DC-195, Margin Management, Rev. 0
M501, Minimum CST Level for Transfer of HPCI Pump Suction to Torus, Rev. 0
M581, Stroke Time Requirements for MO2301-6, MO2301-35, & MO2301-36, Rev. 0
M497, Minimum HPCI Suction Line Pressure at PS2360-1 Tap During HPCI Operation, Rev. 0
M577, Time to Drain HPCI Suction Piping Prior to Swapover From CST to Torus, Rev. 0
I-N1-59, Setpoint Calculation for HPCI Pump Suction Low Pressure Trip, Rev. 2
M667, RHR System Hydraulic Analysis using Proto-FLO Version 1.02, Rev. 2
M734, RHR and Core Spray Pump Suction Strainer Debris Head Loss NPSH Evaluation, Rev. 2
M-662, RHR and Core Spray Pump NPSH and Suction Pressure Drop, Rev. E4
I-N1-215, Uncertainty Calculation-RHR Flow Computer Points RHR022 & RHR024, Rev. 0
I-N1-215, Uncertainty Calculation, RHR Flow Indicators FI-1040-11A & B, and EPIC Pts. RHR-022 & 024, (flow span 0 – 10,000 gpm), Rev. 3
I-N1-215, Attachment 1, Uncertainty Calculation, RHR Flow Indicators FI-1040-11A & B, and EPIC Pts. RHR-022 & 024, (flow span 0 – 6,000 gpm), Rev. 3
I-N1-245, Setpoint Calculation, (E634-3) Setpoints for PS-2390A and PS-2390B of the HPCI System, Rev. 1
S-046, ADS Accumulator Post Accident Operability Time, Rev. 0
M77, Check Valve Leakage on SRV Air Operator, Rev. 0
M85, Leakage Check Valve & Reservoir on SRV Air Operator, Rev. 0
M121, Leakage-Reservoir Tank T-221A thru T-221D, Rev. 0
M405, Assessment of ADS Accumulators T-221A, B, C, D with Relief Valves set at 130 psig, Rev. 0
M563J, AC MOV Design Basis Review, RHR Inboard Injection Valves, Rev. 8
M1100, MOV Periodic Verification Program, Rev. 0
M1118, Thrust and Torque Calculation for MO-1001-29B, Rev. 0 and Rev. 1
PS 217, Setpoint Calculation for 127-504/1,2 & 127-604/1,2 Startup Transformer Undervoltage Relays, dated 9/24/93
PS127, Setpoint Calculation for Bus A5/A6 Loss of Voltage Relays, dated 4/20/95
PS148, Degraded Voltage Alarm Relays - Revised Voltage Setpoint, dated 10/25/94
PS147, Degraded Voltage Trip Relays - Revised Voltage Setpoint, dated 10/25/94
PS88, Voltage Profile & Loading Study for New Security Power System, dated 8/28/90
GE-NE-0000-0000-6533, GE Task T0902 - Anticipated Transients Without Scram, Rev. 0
M-517-1, RHR Pump Min Flow Line Flow Rate, dated 8/22/86
M-517-2, Examine RHR Minimum Flow Requirements OD, dated 2/3/87
M-587, HVAC Pressure Drop TGB Vent Upgrade (SREVS), Rev. 0
M-824, Temperature Limits of Operation for Pilgrim Station Emergency Diesel Generators, Rev. 0
M-991, X-107A/B High Temperature Design Limit (PDC 99-12), Rev. 0.
M-1276, EDG X-107A/B Design Basis Thermal Operating Limits, Rev. 0
N120, HPCI Pump Room Heatup Without Unit Coolers
NEDC-31852P, SAFER/GESTR-LOCA, Loss of Coolant Analysis for Pilgrim Nuclear Power Station, Rev.3
NEDE-30476, Set point Drift Investigation of Target Rock Two-Stage Safety/Relief Valves, dated 2/84

NEDO-22159, Increased Safety/Relief Valve Simmer Margin Analysis for Pilgrim Nuclear Power Station Unit 1, dated 6/82.
PS104, Heat Losses from Electrical Equipment in the Upper & Lower Switchgear Rooms and Battery Rooms A&B for LOCA W/LOOP, LOCA W/O LOOP and 100% Power, Rev. 0
CDCN 04-437, RBCCW Flow Uncertainty Calc., Flow Loops FT6263 & FT-6265
CALC. No. 537-35-17322, EQ Analysis of RCIC Breaks in the RCIC Valve Station and in the RCIC Pump Room, Rev. 2

<u>Completed Surveillance Test Procedures</u>
8.5.2.2.1, LPCI Loop A Operability, dated 5/05 & 1/06
8.7.1.10, ADS Accumulator Pressure Drop and Check Valve Operability Tests, dated 5/2/05
8.M.2-2.10.9, Depressurization System Actuation Logic when Reactor is Shutdown, dated 5/5/05
8.5.6.4, ADS Operability from Alternate Shutdown Panel, dated 4/30/05
8.5.6.2, Special Test for ADS System Manual Opening of Relief Valves, dated 5/14/05
8.M.2-2.10.8.6, Diesel Generator “B” Initiation by Loss of Offsite Power Logic, dated 4/26/05
8.M.2-2.10.8.5, Diesel Generator “A” Initiation by Loss of Offsite Power Logic, dated 10/12/05
8.M.2-2.1.10, 4160 Volt Emergency Buses A5 and A6 Loss of Voltage and Degraded Voltage Relays, dated 4/11/05
8.M.2-2.1.11, Emergency Buses A5 and A6 4.16kV Startup Transformer Undervoltage and Degraded Voltage Relays, dated 4/11/05
8.M.2-2.10.12, HPCI High Water Trip Logic, dated 11/22/04
8.M.2-2.10.5, HPCI Auto-Isolation System Logic, dated 8/2/04
8.5.4.1, High Pressure Coolant Injection (HPCI) System Pump and Valve Quarterly and Biennial Comprehensive Operability, dated 2/24/06.
8.5.4.4, HPCI Valve (Quarterly) Operability Test, dated 2/24/06.
8.9.1, A Emergency Diesel Generator and Associated Emergency Bus Surveillance, dated 3/16/06
8.9.1, B Emergency Diesel Generator and Associated Emergency Bus Surveillance, dated 3/29/06
8.M.3-1, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test, dated 5/8/03
8.M.3-1, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test, dated 5/8/05
2.2.32, Salt Service Water System, Attachment 5, Backwashing the RBCCW Heat Exchangers, dated 4/23/06
2.2.32, Salt Service Water System, Attachment 7, RBCCW/TBCCW Heat Exchanger Differential Pressure Evaluation, dated 4/12/06 and 4/19/06
8.E.30, RBCCW Instrument Calibration; dated 3/24/03, 4/14/03, 3/1/05, and 3/25/05
8.5.3.1, Reactor Building Closed Cooling Water Quarterly Operability, dated 10/31/05 and 12/19/05
8.5.3.1, Reactor Building Closed Cooling Water Quarterly and Biennial Comprehensive Operability; dated 1/16/06, 2/06/06, and 2/28/06
8.5.3.10, RBCCW Motor Operated Valve Operability Test, dated 1/18/06 and 3/01/06
8.5.3.14.1, RBCCW Thermal Performance Test, dated 4/20/03 and 4/18/05

Condition Reports

PR01.0452	2003-02086	2005-00392	2006-00396	2006-01770
PR01.9038	2003-02685	2005-00517	2006-00464	2006-01773*
PR01.9646	2003-02895	2005-01177	2006-01365	2006-01782*
PR01.9712	2003-02905	2005-02132	2006-01460*	2006-01802*
1997-09182	2003-04493	2005-02559	2006-01515	2006-01817
2002-10273	2003-04497	2005-03151	2006-01537*	2006-01848*
2002-11734	2004-00143	2005-03495	2006-01546*	2006-01851*
2002-11878	2004-00212	2005-03643	2006-01560*	2006-01879*
2002-12310	2004-00864	2005-03751	2006-01587	2006-01889*
2002-12925	2004-01368	2005-04115	2006-01592*	2006-01897*
2003-00151	2004-01684	2005-04353	2006-01682*	2006-01902
2003-00193	2004-01987	2005-04631	2006-01695*	2006-01909*
2003-00386	2004-02367	2005-04716	2006-01696*	2006-01911*
2003-00669	2004-03265	2005-05040	2006-01699*	2006-01922*
2003-00870	2005-00256	2006-00121	2006-01719*	2006-01924*
2003-01526	2005-00308	2006-00246	2006-01761*	
2003-01664	2005-00341	2006-00254		

* NRC identified during this inspection

Design Basis Documents

SDBD-01, Automatic Depressurization System (ADS)/Main Steam System (MSS), Rev. E0
SBDB-10, Residual Heat Removal (RHR) System, Rev. 1 (with 11/09/05 Update)
SBDB-23, High Pressure Coolant Injection (HPCI) System, Rev. E0
SDBD-30A, System Design Basis Document for Reactor Building Closed Cooling Water (RBCCW) System, Rev. E0
SDBD-14, System Design Basis Document for the Core Spray System, Rev. E0
SDBD-31, System Design Basis Document for the Compressed Air System, Rev. E0
SDBD-61, System Design Basis Document for Emergency Diesel Generator (EDG) and Auxiliary Systems, Rev. E0
TDBD-111, Topical Design Basis Document for Pipe Break Analysis (HELB, PBOC), Rev. E0
TDBD-105, Topical Design Basis Document for Fire Protection/Appendix R Program, Rev. E0
TDBD-103, Topical Design Basis Document for Environmental Qualification, Rev. E0

Drawings

C341, Miscellaneous Structures Condensate and Demineralized Water Piping, Rev. E1
C338, Miscellaneous Structures Condensate Tank T-105A, T105B & T212 Details, Rev. E3
M241, Residual Heat Removal System, (Sheet 1, Rev. 81; Sheet 2, Rev. 47)
M205G1, Three Way Valve Solenoid Operated, Rev. E4
M205G2, 3 Way Solenoid Valve Manifold Assembly, Rev. E2
M205G3, Air Operator High Pneumatic Pressure, Rev. E1
E9, Single Line Meter & Relay Diagram 480 Volt System-Load Centers & Motor Control Center B10 & B20, Rev. E55
E38, Schematic Diagram 4160 Volt System Breakers 152-504 & 152-604, Rev. E13
E18, Schematic Diagram Diesel Generator Load Shedding, Rev. E18
E17 Sh 1, Schematic Meter & Relay Diagram 4160 Volt System, Rev. E14
M244 Sh 2, HPCI System Turbine Lube and Control Oil Subsytem, Rev. 9
M1J14-14 Sh 1, Elementary Diagram HPCI System, Rev. E26

M1J14-14 Sh 2, Elementary Diagram HPCI System DC MOV & Control Center, Rev. E4
M1J15-10 Sh 2, Elementary Diagram HPCI System, Rev. E21
M1J16-10 Sh 3, Elementary Diagram HPCI System, Rev. E25
M1J17-12 Sh 4, Elementary Diagram HPCI System, Rev. E24
M1J18-11 Sh 5, Elementary Diagram HPCI System, Rev. E21
M1J32 Sh 9, Elementary Diagram HPCI System, Rev. E7
M-223, Diesel Fuel Oil & Transfer System P&ID, Rev. E29
M220 Sh 3, Compressed Air System Essential Instrument Air P&ID, Rev. E69
M227 Sh 2, Containment Atmospheric Control System P&ID, Rev. E48
M-242, Core Spray P&ID, Rev. E50
M-244 Sheet 1, HPCI P&ID, Rev. E30
M252 Sh 1, Nuclear Boiler P&ID, Rev. 63
M289, Reactor Building Air Flow Diagram HPCI, Rev. E17
M-311, Heating Ventilation & Air Conditioning Reactor Building Below EL 23'-0" Plans & Sections, Rev. E6
M-348, RHR 3" Common Minimum Flow Line Flow Resistance, Rev. 0
70-66, Fuel Oil Day Tanks for Bechtel Corporation, Rev. 11
M11-26-2, Sh 2, RBCCW E-209A Tube Layout as of April 2003, Rev. E8
M11-26-2, Sh 3, RBCCW E-209B Tube Layout as of April 2003, Rev. E8

Engineering Evaluations
Operability Evaluation CR-PNP-2006-00254, Rev. 0 and Rev. 1
Operability Evaluation CR-PNP-2006-1802
Safety Evaluation No. 2983, Evaluation of Maximum Salt Service Water Temperature of 75EF
SE 1830, Change HPCI required Start Time from 25 Seconds to 90 Seconds, dated 5/24/85
SE 3260, Revised HPCI and RCIC Pump Test Acceptance Criteria Based on Design Basis Hydraulic Analysis, dated 6/15/99
SE 3317, Re-introduce Original HPCI System Analysis and Requirements in FSAR Section 6, dated 8/22/00

Maintenance Work Orders

MR01108097	MR02108711	MR06103920	MRP9700406
MR01109827	MR05108234	MRE9700167	
MR01121592	MR06103919		

Miscellaneous
Operations Section Standing Order No. 05-11, HPCI Turbine Trip Function, dated 11/21/06
NEDWI No. 394, Methodology for Calculation of Instrument Setpoints, Rev. 3
PDC 94-18A, Modification to MO-1001-29B for Generic Letter 89-10, dated 4/8/94
Gould Pump Manual, 7th Edition, (pages 728 - 733)
Y. A. Reddy and J. A. Pickford, Vortices at Intakes in Conventional Sumps
ANSI/HI 9.8-1998, American National Standard for Pump Intake Design
F. M. Patterson, Vortexing Can Be Prevented, The Oil and Gas Journal, 1969
Module No. O-RQ-06-02-51, Scenario No. 11: Station Blackout Lab, Rev. 0
PRA - Appendix H, Post-Accident Human Reliability Analysis, Rev. 1
Standing Order No. 06-01, Bussmann Fuses, Rev. 1, dated 2/23/06
PDC 97-14, Revised Power Supply for Station Blackout Diesel Generator Auxiliaries, dated 4/22/99
EN-EE-S-012-P, Pilgrim Station Bussmann Fuse Replacement, Rev. 0

1.83.324, NRC Letter Issuing Amendment No. 73, dated 12/29/83
21A1110, Specification for Reactor Pressure Vessel, Rev. 2
257HA354, High Pressure Coolant Injection System Design Specification, Rev. 2
86/0165, Bechtel Letter, dated 12/12/86
87/0054, Bechtel Letter, RHR Minimum Flow, dated 2/10/87
93J802-C01, Fan Performance Curves for Switchgear Room Emergency Ventilation System, Rev. 0.
BECo 2.99.041, Request for Technical Specification Change Concerning HPCI and RCIC Surveillance Testing, dated 5/11/99
BECo 83-85, Proposed Change to Technical Specifications, dated 4/5/83
BECo Ltr. No. 82-51, Report on Target Rock Safety/Relief Valves, dated 2/11/82
Bingham Willlamette Letter, dated 9/2/86
CR04-0212, Maintenance Rule Action Plan for Main Steam SRV, Rev. 2
EDG Reliability Spreadsheets
FDI 93/78003, Minimum Flow Bypass Orifices, RHR System, dated 10/25/71
G-HK-7-420, GE Letter, Adequacy of Pilgrim ECCS Minimum Flow Capacity, dated 9/30/87
LER 2004-001-00, Target Rock Relief Valves' Test Pressures Exceeded Technical Specification Tolerance Limit.
LER 2004-003-00, Target Rock Relief Valves' Test Pressures Exceeded Technical Specification Tolerance Limit
LER 2005-003-00, Target Rock Relief Valves' Test Pressures Exceeded Technical Specification Tolerance Limit
MPL 1401, QC Records Core Spray Pumps
NEA-02-044, Entergy Memo: The Temperature Profile of the HVAC Systems for the Pilgrim IPE. An Update for the Life Extension Project, dated 2/13/02
PDC 86-53, Backup Nitrogen Supply, Rev. 0
PDC 86-95, Increase RHR Minimum Flow, dated 3/20/87
PDC 87-55, Emergency Diesel Generator Radiator Cooling Fan Modification, Rev. 0
PDC 89-52, Nitrogen System Upgrade, Rev. 0
PDC 96-19, A-46 Nitrogen Supply for Pressure Control, Rev. 0
PDC 98-097, Upgrade of RHR Flow Instrument Loops
PDC 99-12, EDG Ventilation and Radiator Fan Modifications, Rev. 0
PDC 00-12, EDG Air Starter Motor Replacement, Rev. 0
PNPS-RPT-04-004, PNPS Air Operated Valve Categorization, Rev. 0
RHR Pump Curves
FE-6240, Specification Data
FE-6265, Specification Data
FT-6240, Specification Data Sheet
FT-6240, Setpoint, Loop Accuracy and Calibration Data
FT-6265, Specification Data Sheet
FT-6265, Setpoint, Loop Accuracy and Calibration Data
M591, SSW & RBCCW Safety-Related Piping & Heat Exchanger Inspection, Maintenance, & Test Requirements in Response to Generic Letter 89-13, Rev. E7
NOP02E1, Service Water Inspections, Maintenance and Testing in Response to Generic Letter 89-13, Rev. 1
PNPS-PSA, Pilgrim Nuclear Power Station Probabilistic Safety Assessment, Rev. 1
Risk-Informed Inspection Notebook for Pilgrim Nuclear Power Station Unit 1, Rev. 2
NRC Inspection Report 50-293/84-39, dated 2/7/85
NRC Inspection Report 50-293/85-17, dated 8/6/85

Margin Review Board Meeting Minutes (First Quarter 2006), dated 4/3/06
Environmental Qualification Master List, Rev. E51
V-0321, High Pressure Coolant Injection System Vendor Manual, Rev. 2
PNPS Top Ten Equipment Reliability Issues, dated 3/21/06

Normal and Special (Abnormal) Operations Procedures
1.3.34, Operations Administrative Policies and Processes, Rev. 106
8.M.2-2.5.7, Instrument Functional/Calibration Test For HPCI Suppression Chamber Water Level, Rev. 32
8.M.5.2.2.2, LPCI System Loop B Operability – Pump Quarterly and Biennial (Comprehensive) Flow Rate Test and Valve Tests, Rev. 38
3.M.4-6, Removal, Installation, Test, Disassembly, Inspection, and Reassembly of Main Steam Relief Valves, Rev. 41
3.M.4-69, Soft Seat Replacement For ADS Air Supply Check Valve, Rev. 8
8.5.4.1-1, High Pressure Coolant Injection (HPCI) Simulated Automatic Actuation, Flow Rate and Cold Quickstart Test, Rev. 18
5.3.35, Operations Management Emergency and Transient Response Expectations for Operating Crews, Rev. 8
8.5.4.1, HPCI System Pump and Valve Quarterly and Biennial Comprehensive Operability, Rev. 100
8.7.1.10, ADS Accumulator Pressure Drop and Check Valve Operability Tests, Rev. 17
2.1.42, Operation During Severe Weather, Rev. 4
2.2.146, Station Blackout Diesel Generator, Rev. 38
2.2.146, Emergency Diesel Generator Daily Surveillance, Rev. 57
2.2.19.5, RHR Modes of Operation for Transients, Rev. 14
2.2.21, High Pressure Coolant Injection System, Rev. 66
2.2.21.5, HPCI Injection and Pressure Control, Rev. 12
2.2.105, Backup Nitrogen Supply System, Rev. 29
2.2.70, Primary Containment Atmospheric Control System, Rev. 96
2.4.144, Degraded Voltage, Rev. 33
2.4.153, Loss of Turbine Building/Aux Bay Area Ventilation, Rev. 15
2.4.16, Distribution Alignment Electrical System Malfunctions, Rev. 31
2.4.21, Double-Ended Break of the 3-inch Instrument Air/Nitrogen Line in the Drywell, Rev. 10
2.4.29, Stuck Open Safety Relief Valve, Rev. 19
2.4.35, Inadvertent Initiation of Core Standby Cooling Systems, Rev. 19
2.4.42, Loss of RBCCW, Rev. 26
5.3.21, Bypassing Selected Interlocks, Rev. 18
5.3.26, RPV Injection During Emergencies, Rev. 19
5.3.31, Station Blackout, Rev. 10
5.4.6, Primary Containment Venting and Purging Under Emergency Conditions, Rev. 30
7.1.87, Diesel Fuel Oil Storage Tank Sampling, Rev. 12
7.8.1, Chemistry Sample and Analysis Program, Rev. 40
8.5.2.10, RHR Temperature and Pressure Monitoring, Rev. 13
8.5.2.7, Hydrodynamic Test for Measuring Leakage Through RHR System
8.5.6.4, ADS Operability from Alternate Shutdown Panel, Rev. 10
8.M.2-2.5.6, HPCI Condensate Storage Tank Level, Rev. 31
ARP-905L-B5, Alarm Response Procedure, Rev. 8
ARP-C903C, Alarm Response Procedure, Rev. 12
ARP-C103B, Alarm Response Procedure, Rev. 8

ARP-C903R, Alarm Response Procedure, Rev. 13
ARP-C904LC, Alarm Response Procedure, Rev. 17
EN-LI-100 Process Applicability Determination, Revision 1
EN-LI-101 10 CFR 50.59 Review Program, Revision 2
EN-OP-104, Operability Determinations, Rev. 1
EN-OP-115, Conduct of Operations, Rev. 0
EOP-01, RPV Control, Rev. 9
EOP-02, Failure to Scram, Rev. 9
EOP-03, Primary Containment Control, Rev. 8
EOP-04, Secondary Containment Control, Rev. 7
EOP-05, Radioactivity Release Control, Rev. 4
EOP-11, Figures, Cautions and Icons, Rev. 2
EOP-16, RPV Flooding, Rev. 5
EOP-17, Emergency RPV Depressurization, Rev. 5
JPM-200-08, Licensed Operator Job Performance Measure - Direct Torus Vent (In Plant), Rev. 0
JPM-200-10, Nuclear Plant Operator Job Performance Measure - Direct Torus Vent (Simulator), Rev. 2
JPM-205-01, Licensed Operator Job Performance Measure - Suppression Pool Cooling, Rev. 6
JPM-205-04, Nuclear Plant Operator Job Performance Measure - Initiate Torus Cooling and Torus Sprays, Rev. 3
JPM-205-11, Nuclear Plant Operator Job Performance Measure - Fire Water Cross-Tie to RHR, Rev. 6
JPM-205-15, Nuclear Plant Operator Job Performance Measure - RHR Operations (Containment Spray), Rev. 0
JPM-290-02, Nuclear Plant Operator Job Performance Measure - Local Operation of the SBO DG During Station Blackout, Rev. 4

<u>Operating Experience</u>
OE19161, LTOP Nitrogen Backup Calculation Deficiencies, dated 8/04/04
OE22390, Non-Conservative Vortexing Methodology Used for Calculating BWST Isolation Level during an Event, dated 3/10/06
OE00.0017, Review of NRC Information Notice 2000-08, Inadequate Assessment of the Effect of Differential Temperatures on Safety- Related Pumps
NRC Information Notice 86-51: Excessive Leakage in the Automatic Depressurization System, dated 6/18/86
NRC Information Notice 94-06: Potential Degradation of Long-Term Emergency Nitrogen Supply for the Automatic Depressurization System Valves, dated 1/28/94
Information Notice 97-21, Availability of Alternate AC Power Source Designed for Station Blackout Event, dated 4/18/97
Information Notice 97-78, Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times, dated 10/23/97
Information Notice 2006-05, Possible Defect in Bussmann KWN-R and KTN-R Fuses, dated 3/3/06
OE94.0022.01, PNPS Response to NRC Information Notice 94-06, dated 5/2/94
Generic Letter 96-01, Testing of Safety-Related Logic Circuits, dated 4/30/87
1.89.252, Response to NRC Information Notice 86-51, dated 7/6/89
BECo 88-110, Response to NRC Bulletin 88-04, dated 7/13/88
BECo Response to NRC Bulletin 88-04, dated 7/13/88

BWROG-8836Response to NRC Bulletin 88-04, dated 6/29/88
BECo 90-047, Response to NRC Generic Letter 89-13 Service Water System Problems Affecting Safety-Related Equipment, dated 4/2/90
BECo 97-095, Update of GL 89-13 Response, dated 9/18/97
BECo Ltr. No. 81-72, Response to IE Bulletin No. 80-25, Operating Problems with Target Rock Safety/Relief Valves at BWR's, dated 4/8/81

System Health Reports & Trending
RHR System Health Report, 1st Quarter 2006
Stroke Time Testing Trending for MO-1001-29B
05E - Emergency Lighting, 1st Quarter 2006
SBO DG System Health Report, 1st Qtr 2006
HPCI System Health Report, 4th Qtr 2005
Core Spray Pumps IST Trend Spreadsheets
HPCI Turbine Pump IST Spreadsheets
Reactor Building Closed Cooling Water System Health Report, 4th Qtr 2005
Reactor Building Closed Cooling Water Pump IST Trend Spreadsheet

LIST OF ACRONYMS

ADAMS	Agency-Wide Documents Access and Management System
ADS	Automatic Depressurization System
BIL	Basic Impulse Level
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CR	Condition Report
CST	Condensate Storage Tank
DBA	Design Basis Accident
DBD	Design Basis Document
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
GE	General Electric
GPM	Gallons Per Minute
HPCI	High Pressure Coolant Injection
IN	Information Notice
IST	In-Service Test
JPM	Job Performance Measure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
MOV	Motor Operated Valve
MR	Maintenance Rule
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission

OE	Operating Experience
PARS	Publicly Available Records
PDC	Plant Design Change
PNPS	Pilgrim Nuclear Power Station
PRA	Probabilistic Risk Analysis
RAW	Risk Achievement Worth
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SBO	Station Blackout
SDP	Significance Determination Process
SPAR	Simplified Plant Analysis Risk
SRV	Safety Relief Valve
ST	Surveillance Test
SUT	Startup Transformer
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UV	Undervoltage