January 17, 2001

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

SUBJECT: PILGRIM STATION - NRC INSPECTION REPORT NO. 05000293/2000-012

Dear Mr. Bellamy:

On December 15, 2000, the NRC completed a team inspection of the high pressure coolant injection system, the ability of primary and secondary containment to function, and the availability of these to perform during a loss of offsite power. The team also inspected your evaluations of plant changes, tests and experiments. The enclosed report presents the results of that inspection. The findings were discussed with you, and other members of your staff, on December 14, 2000.

This inspection was an examination of activities conducted under your license as related to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observation of activities, and interviews with personnel.

The team identified seven issues that were evaluated under the risk significance determination process (SDP) and were determined to be of very low safety significance (Green). These issues have been entered in your corrective action program and are discussed in the enclosed inspection report. If you contest the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and the NRC Resident Inspector at the Pilgrim Nuclear Power Station.

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

Sincerely,

/RA J. C. Linville Acting for/

Wayne D. Lanning, Director
Division of Reactor Safety

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

Enclosure: Pilgrim - NRC Inspection Report 05000293/2000-012

Attachments: (1) NRC's Revised Reactor Oversight Process
(2) Supplemental Information
(3) List of Documents Reviewed
(4) List of Acronyms Used

cc w/encl:
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J. Alexander, Director, Nuclear Assessment Group
D. Tarantino, Nuclear Information Manager
S. Brennion, Regulatory Affairs Department Manager
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R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
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P. Gromer, Massachusetts Secretary of Energy Resources
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	05000293
License No:	DPR-35
Report No:	05000293/2000-012
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Dates:	November 27- December 1 and December 11-15, 2000
Inspectors:	E. Harold Gray, Senior Reactor Inspector, Team Leader, DRS R. Arrighi, Resident Inspector, DRP F. Arner, Reactor Inspector, DRS G. Cranston, Reactor Inspector, DRS G. Morris, Reactor Inspector, DRS J. Talieri, Reactor Inspector, (Trainee), DRS
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 050000293-00-12, on 11/27-12/15/2000, Entergy Nuclear Generation Company, Pilgrim Nuclear Power Station (PNPS). Inspection procedure attachments titled “Safety System Design and Performance Capability” and “Evaluations of Changes, Tests, or Experiments” were associated with findings during this Engineering inspection of a Mitigating System and Barrier Integrity.

This report includes the results of a team inspection, by resident and region-based inspectors, of the Primary and Secondary containment functions, high pressure coolant injection (HPCI) system, related direct current (DC) systems, and the conduct of evaluations of changes, tests and experiments under the 10 CFR 50.59 process. This inspection applied the NRC Baseline Inspection Procedure 71111.21, “Safety System Design and Performance Capability.” The team also reviewed the PNPS’s evaluation of changes, tests and experiments under the 10 CFR 50.59 process using NRC Baseline Inspection Procedure 71111.02, “Evaluations of Changes, Tests, or Experiments.” The significance of the seven issues identified is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process (SDP) in the NRC Inspection Manual Chapter 609 (see Attachment 1).

Cornerstone: Mitigating Systems

1. Green. The team identified that the cooling water flowrate to the high pressure coolant injection (HPCI) lube oil cooling heat exchanger was not measured. A flowrate of 70 gallons per minute is required to ensure adequate cooling with the maximum temperature cooling water. During normal HPCI turbine operation the cooling water supply is from the condensate storage tank (CST) which supplies water at temperatures well below the design maximum torus temperature of 140 degrees Fahrenheit (^{o}F). Degradation in cooling water supply flow may not be readily apparent during normal system testing.

   This issue has a credible impact on safety and is more than a minor issue. Existing surveillance testing is performed with suction from the CST but testing with the 140^{o}F cooling source or a flow verification have not been performed. Worst case design suction temperatures of 140^{o}F from the torus could occur in only a few events including station blackout or an anticipated transient without a scram (ATWS). Therefore, even if the cooling water supply flow was degraded, the risk aspect of this issue would be considered to be of very low safety significance (Green). The licensee initiated problem report PR00.3467 to evaluate monitoring of lube oil cooler performance and measuring the lube oil cooler water flowrate in order to verify that the minimum design flowrate and cooling capacity would be achieved. (Section 1R21.1.1, HPCI Lube Oil Cooling)

- Green. On June 17, 1999, a procedure revision was performed which removed the HPCI pump discharge pressure criteria. By not having a test acceptance criteria of either pressure or speed along with flow, degradation of the speed controller limiter could go undetected. The surveillance test acceptance criteria no longer ensured operability of all HPCI equipment at the remote shutdown panel. The failure to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was considered a 10 CFR Part 50, Appendix B, Criterion XI, Test Control, violation. The issue was determined to be of very low safety significance (Green) because a review of the most recent test results from December 1999, showed that the controller speed limiter had not been degraded and rated speed had been achieved. The issue was a non-cited violation and was entered into the PNPS corrective action program. (Section 1R21.1.2, Surveillance Procedures)

- Green. The team found that the licensee failed to provide adequate review of the acceptance criteria for the battery surveillance discharge tests. The problems identified included incorrect minimum voltage for the service test acceptance criteria for both the HPCI and RCIC station battery technical specification surveillance tests. The failure to correctly provide adequate acceptance criteria in battery surveillance procedures and failure to verify or check the adequacy of test procedures was the first example of a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was determined to be of very low safety significance (Green) because it had negligible impact on the operability of the systems based on the latest test, since voltage did not drop to the correct minimum voltage level. These issues were a non-cited violation and were entered into the PNPS corrective action program. (Section 1R21.3.1, Design - Mechanical, Electrical and Instrumentation and Controls)

- Green. The team found that the licensee failed to control the inputs and assumptions used in the calculations for determining large battery sizing. The failure to correctly provide adequate design inputs and assumptions for the design margin correction factor in the battery sizing calculations had very low safety significance (Green) because there was negligible impact to the operability of the system based on compensating margins included in the body of the calculation and rounding-up margins to the next larger number of positive plates. The licensee’s failure to verify or check the adequacy of design calculations was the second example of a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was a non-cited violation and was entered into the PNPS corrective action program. (Section 1R21.3.1, Design-Mechanical, Electrical and Instrumentation and Controls)

- Green. The team identified that the 125 Volt “B” battery service test documentation failed to record the battery voltages during the critical first hour of the test when the load currents were the highest and the voltage dips the lowest. (During the review of these concerns, the licensee discovered that the 125 V “A” battery, the power supply for the reactor core isolation cooling system (RCIC) motor operated valves (MOVs), also had not been correctly discharged according to its service test duty cycle.) The failure to correctly test and record the appropriate data was concluded to have very low safety significance (Green) based on the results of special tests performed by PNPS during the inspection. The failure to adequately review the results of the service test data was a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control. The issue was a non-cited violation and was entered into the PNPS corrective action program. (Section 1R21.3.2, Operations and Maintenance)

- Green. Performance tests to the manufacturer’s rated eight hour discharge current for the NCN-35 cells were performed in March 1997 for the 250 V battery and in September 1996 for the 125 V “B” battery. Both test records recorded capacities over 100%. However, the team found that there was no data included in the QA test record that substantiated the recorded results for the “B” battery. Therefore the team was unable to independently verify the recorded capacity because of the missing data. The failure to adequately maintain the required records was considered to have very low safety significance (Green) because the results of those tests are used for aging determination and the batteries are relatively new. Failure to properly maintain required QA records was a violation of 10 CFR 50, Appendix B, Criterion XVII, QA Records. The issue was a non-cited violation and was entered into the PNPS corrective action program. (Section 1R21.3.2, Operations and Maintenance)

- Green. The team determined that several opportunities had existed for the licensee to identify and correct a degraded HPCI flow controller prior to a failure to meet surveillance test acceptance criteria during September 1, 2000 testing. Based on the result that although degraded, the system could have met its safety function regarding the transient and loop scenarios, this issue was determined to have very low safety significance (Green). Notwithstanding this determination, the team concluded that the licensee’s handling of this issue showed weak problem identification and corrective action relative to a risk significant component and system.

  The team determined this issue to be a violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action, based on the failure to identify the degraded speed controller and correct it prior to failing the surveillance test criteria. The issue was a non-cited violation and was entered into the PNPS corrective action program. (Section 4OA1.1, Identification and Resolution of Problems)

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1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (IP 71111.21)

Introduction

The team selected the high pressure coolant injection (HPCI) system and the adequacy of the primary and secondary containment to function for their review of the design and performance capability of safety systems at the Pilgrim Plant. The two systems were selected because of their risk significance in event mitigation, barrier integrity, and core damage prevention. The primary function of the HPCI system is to prevent excessive fuel clad temperatures by providing high pressure coolant to the reactor core in the event of a small break loss of coolant accident (LOCA) that does not result in the rapid depressurization of the reactor vessel. The main function of the primary containment system is to provide a barrier to prevent a radiological release during operation or in the event of an accident. The secondary containment system was selected as a barrier system complementary to the primary containment. In addition the inspection evaluated the capability of the HPCI system to work properly in the case of a loss of offsite power. The Inspection Procedure used for this effort was IP 71111, Attachment 21.

.1 System Design - High Pressure Injection System/Mechanical, Instrumentation and Controls

a. Inspection Scope

The team reviewed the High Pressure Coolant Injection (HPCI) system design and licensing basis documents, including applicable portions of the Updated Final Safety Analysis Report (UFSAR), the plant Technical Specifications (TS), the design basis document (DBD), and the Individual Plant Examination (IPE). This review was performed to determine the system and component functional requirements during transient and accident conditions. The reviews also verified that selected design documents, such as drawings and design calculations, were correct. The documents reviewed included engineering analyses, calculations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument set points. Additionally, plant procedures including surveillance tests were reviewed to ensure they supported the system licensing and design bases.

In accordance with Inspection Procedure 71111.21, the team selected several major components to perform an in-depth inspection. These components included the HPCI main and booster pumps, the HPCI turbine, and the turbine control system. The team also focused on verifying that risk significant equipment which was not routinely tested during surveillance testing, such as the motor operated injection valve, would function under accident or transient conditions.

For selected mechanical calculations and analyses, the team verified that the assumptions were appropriate and agreed with current plant configurations, that proper engineering methods and models were used, and that there were adequate technical bases to support the conclusions. When appropriate, the team performed independent calculations to evaluate the document adequacy. For selected plant modifications including changes to procedures, the team verified that the ability of the HPCI system to perform its design function was not adversely affected by the change.

The team also reviewed selected portions of design documents of interfacing systems, such as the condensate storage tanks. For these systems, the team assessed the capability of the supporting systems to satisfy the design functions of the HPCI system.

Finally, the team performed plant walkdowns of the HPCI system, specifically focusing on the HPCI turbine, pump and control system. The team also performed a walkdown of the alternate shutdown panel which contained HPCI instrumentation utilized to achieve a shutdown in the event the control room instrumentation could not be used.

b. Findings

.1.1 HPCI Lube Oil Cooling

The HPCI system was designed to operate without the use of an external cooling water system. To accomplish this, the HPCI booster pump should supply a nominal 70 gallons per minute of cooling water to the turbine lube oil cooler and the gland seal condenser. This flowrate is required to ensure adequate cooling at worst case temperature conditions of 140 degrees Fahrenheit (^{o}F) with maximum turbine heat rejection rates. A flow restricting orifice and a pressure control valve are provided to supply the required flow. The oil cooler is a straight tube type heat exchanger which maintains the oil temperature to the turbine and main pump bearings between 90 and 150^{o}F. An alarm on control room panel C-921 actuates when a temperature switch senses a 155^{o}F cooler outlet oil temperature. The team noted that HPCI pump, valve and flow quarterly Surveillance Test 8.5.4.1, Revision 59, contained a step to verify forward flow of the HPCI turbine lube oil cooler return check valve. This step was performed by verifying that the oil cooler discharge temperature was at or below 130^{o}F as indicated by temperature recorder, TR-263-104. The basis of the 130^{o}F value was unclear at the time of the inspection and the team questioned the ability of this value to indicate that the minimum expected cooling water flowrate was supplied to the cooler. During normal turbine operation the water supply is from the condensate storage tank (CST) which supplies water at temperatures well below the worst case design maximum torus temperature of 140^{o}F. Therefore, degradation in cooling water supply flow may not be readily apparent during normal system testing.

Without an evaluation or analysis of the current cooling water flow, the team considered this an issue which had a credible impact on safety and therefore more than a minor issue. Existing surveillance testing is performed with suction from the CST and the team had no indication that high oil temperatures had occurred in this configuration. Worst case design suction temperatures of 140^{o}F from the torus would not be expected to occur except for a few events such as station blackout or an anticipated transient without a scram (ATWS). Even if the cooling water supply flow would be degraded, the risk significance of this issue would be considered to be of very low significance

(Green). The licensee initiated problem report PR00.3467 to evaluate monitoring of lube oil cooler performance and measuring the lube oil cooler water flowrate in order to verify that the minimum design flowrate and cooling capacity would be achieved.

.1.2 Surveillance Procedures

During a review of surveillance test procedure 8.5.4.6, Revision 27, "HPCI Pump and Valve Operability From Alternate Shutdown Panel," the team determined that an appropriate acceptance criterion was not established to prove operability of all HPCI equipment operated from the Alternate Shutdown Panel. The purpose of the procedure is to demonstrate that panels C155 and C158 are operable for the HPCI system in accordance with Technical Specification surveillance requirement Section 4.12.5 (Alternate Shutdown Panel Capability for HPCI). Licensee Safety Evaluation 3317, dated August 22, 2000, described the design basis maximum operating pressure of the HPCI pump as the maximum pressure associated with the lowest analytical setpoint of the safety relief valves (SRV) operating in the spring safety mode. PNPS has one group of four safety relief valves with an upper analytical setpoint of 1,126 psig as specified in Final Safety Analysis Report Table (UFSAR) 4.4-1. The team reviewed past performances of the surveillance test (revision 24) and noted that it previously contained pump flow acceptance criteria along with pump discharge pressure acceptance criteria of greater than or equal to 1,225 psig. The 1,225 psig was based on the pressure required at the pump discharge to inject against the upper analytical SRV setpoint when factoring in elevation and dynamic losses.

On June 17, 1999, a procedure revision was made that removed the pump discharge pressure criteria. The basis for removing this criterion was that the test was not a pump operability test and the change did not impact the function of satisfying Alternate Shutdown Panel Capability. The team concurred with the conclusion that this procedure was not intended to be an HPCI pump operability test and was not associated with ASME Code Section XI test requirements. However, the team did not agree that removing the previous pressure criteria could not impact the function of determining the ability of HPCI equipment from this alternate shutdown panel to perform its function.

Specifically, the procedure instructed the operator to throttle a test return valve and/or adjust the flow controller FIC-2340-2 on panel C155 to obtain a flowrate of 4,250 gpm or greater. However, the flow controller has a high limit setting associated with the controller which determines the maximum signal that the controller will output. Section 4OA1of this report documents a recent failure of this limiter with a similar HPCI controller in the main control room. With only flow acceptance criteria, failure or degradation of this limit associated with the alternate shutdown panel flow controller could go undetected during performance of this test. If the flow controller high limit (speed signal) was degraded below design speed, the test allowed for throttling open the return valve, which would lower system pressure, in order to achieve the required flow of 4,250 gpm. By not having a test acceptance criteria of either pressure or speed along with flow, degradation of the controller limiter would go undetected. Therefore, the surveillance test acceptance criteria did not ensure operability of all HPCI equipment at the remote shutdown panel. The failure to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service was considered a 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," violation. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) because test results from December 1999, showed that the controller speed limiter had not been degraded and rated speed had been achieved. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue associated with this violation is in the PNPS corrective action system as PR 00.3464. **(NCV 05000293/2000-012-01)**

.1.3 Quality Assurance Effectiveness

a. Inspection Scope

The team reviewed Quality Assurance (QA) Audit Report 97-06, "High Pressure Coolant Injection (HPCI) System," surveillance reports 99-065 and 99-168, and several installation reports related to various electrical components. In addition, the team interviewed the QA manager and reviewed licensee procedure regarding QA findings and selection of components for review during surveillance and audit activities.

b. Findings

There were no findings identified.

.2 Containment Function

Cornerstone: Barrier Integrity

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report (UFSAR) and system design basis documents to establish the design and licensing basis for the primary and secondary containment systems and the attributes needed for proper functioning of containment. The piping and instrumentation drawings (P&ID), electrical drawings and the installed configuration outside containment were reviewed to assess the capability of containment to satisfy the design intent.

.2.1 Primary Containment System

.2.1.1 Design - Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed the primary containment system design and licensing basis documents for containment leak detection (drywell floor sump pumps), containment vacuum breakers (torus-to-drywell and reactor building-to-torus), and selected containment isolation valves to determine the system and component functional requirements during normal, abnormal and accident conditions. For the documents reviewed, which included component vendor manuals, calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used and that there was an adequate technical basis to support the conclusions. The reviews were performed to determine that: (1) the design basis was in accordance with the licensing commitments and regulatory requirements; (2) the design output documents such as drawings were correct; and (3) the installed system and components were tested to verify the design bases were met.

b. Findings

There were no findings identified.

.2.1.2 Operations and Maintenance

a. Inspection Scope

The team reviewed a number of activities to verify that the selected primary containment components were installed, operated and maintained consistent with the design and licensing basis. The team reviewed the design basis documents, system modifications, technical specifications, P&ID, UFSAR, and applicable operating, surveillance, maintenance and alarm response procedures. The team also interviewed a licensed reactor operator and the system engineer assigned to the primary containment system. In addition, the team reviewed the licensee’s response and corrective actions to address GL 96-06, “Assurance of Equipment Operability and Integrity During Design-Basis Accident Conditions,” as it applied to the containment leak detection system. As part of the review, the team also evaluated a sample of licensee-identified problems in the licensee’s corrective action system to assess the effectiveness of the licensee’s corrective actions.

b. Findings

There were no findings identified.

.2.1.3 Surveillance and Testing

a. Inspection Scope

The team reviewed technical specifications, test procedures and recent performance data to verify that the following components met their design and licensing bases:

- Drywell Floor Sump Pumps P-305 A/B
- Reactor Building-to-Torus Vacuum Breakers AO-5040A/B and X-212 A/B
- Torus-to-Drywell Vacuum Breakers X-201 A thru K

The team also reviewed the primary containment isolation valve local leakage rate test results for all containment isolation valves for the past three refueling outages (RFO 10, 11 and 12). The inspector noted that failure of the overall as-found leakage rate to meet the Technical Specifications allowable leakage rates during the 10 CRF 50, Appendix J, Local Leak Rate Testing (LLRT) was due to leakage of one or more of the main steam isolation valves (MSIV) or the feedwater (FW) check isolation valves. There are four main steam lines and two feedwater lines penetrating the primary containment with two containment isolation valves in series in each line for redundancy. Except for RFO 12, no two isolation valves in the same line failed to pass the LLRT acceptance criteria. During RFO 12, MSIVs AO-203-1C and 2C both failed the LLRT. The inspector reviewed Licensee Event Report 1999-003-00 which was associated with the LLRT failure of both MSIVs. Also, four (4) problem reports associated with the MSIV or the FW isolation valves were reviewed (PR 99.9216, PR 99.9226, PR 97.9137, and PR 97.9136). Associated cause evaluations and corrective actions were examined. Discussions with the system engineer responsible for LLRTs were conducted regarding his knowledge of the problems, historical trends, maintenance history, previous repairs, and proposed actions to prevent recurrence of excessive MSIV and FW isolation valve leakage.

b. Findings

There were no findings identified.

.2.2 Secondary Containment Systems

.2.2.1 Design - Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed the secondary containment Standby Gas Treatment System (SBGT) fan/filter system and the proposed corrective action identified in problem report PR 00.9143, regarding the undersized pre-heaters for reactor building air entering the charcoal filter beds. The inspector also reviewed the licensee's basis for continued operation (Operability Evaluation OE00-009) with the existing undersized heaters and discussed with the system engineer the proposed plant design change to upgrade the heater capacity to ensure all air entering the charcoal filter beds is less than 70% relative humidity as required by the Technical Specifications.

The team reviewed the proposed modification to the SBGT system isolation dampers air supply. The licensee identified a problem of under sizing of the existing safety-related accumulators, which supply air to allow post accident operation of the dampers, in problem report PR00.9130. Also reviewed was the pressure drop test procedure for the SBGT dampers and the safety evaluation (SE 3322), which were prepared to allow continued operation with the existing design. Plant design change PDC 00-25, issued to correct the problem, and associated calculation M679, were reviewed to evaluate the appropriateness of the proposed corrective action to prevent recurrence. The PDC consisted of increasing the size of the safety-related accumulators, raising the low pressure setpoint for recharging the accumulators, and adding a non-safety related air compressor to maintain the accumulators fully charged.

b. Findings

There were no findings identified.

.2.2.2 Operations and Maintenance

a. Inspection Scope

The team reviewed the secondary containment isolation system, including building leakage test results, challenges to maintaining containment, the reactor building ventilation isolation dampers and air locks, and the standby gas treatment system (SBGT), including SBGT isolation dampers and filter performance.

The team reviewed engineering evaluation (EE 00-011), safety evaluation (SE 3296) and controls in place to maintain secondary containment while repairing the reactor building (secondary containment) roof since abandoned screw holes through the roof would be temporarily exposed while portions of the old roof were removed and the new roof was installed.

Also, the team reviewed engineering evaluation (EE 00-037) and controls in place to maintain secondary containment prior to and during repair of a leak path through the reactor building truck air lock doors due to door seal problems identified in problem reports PR00.9287 and PR00.9083.

Discussions were held with the system engineer responsible for secondary containment regarding the problems associated with the secondary containment integrity, previous test results, and proposed actions to correct the existing problems.

b. Findings

There were no findings identified.

.2.2.3 Surveillance and Testing

a. Inspection Scope

The team reviewed the secondary containment structure and associated isolation system dampers, personnel air locks and truck-bay air lock barrier leakage test results and challenges to maintaining containment. PNPS Procedure No. 8.7.3, for conducting secondary containment tests and the results of the most recent test were reviewed.

b. Findings

There were no findings identified.

.3 Adequacies of the Electrical System to Support Operation of the HPCI System in the Event of a Loss of Offsite Power

Cornerstone: Mitigating System

Introduction

The team reviewed the instrumentation, control and power supplies associated with the High Pressure Coolant Injection (HPCI) system.

In the area of instrumentation, the team reviewed selected instrument setpoint calculations and instrument calibration dates. In the area of controls, the team reviewed initiation logic, interlocks, Functional Control Diagrams and Elementary Diagrams associated with selected motor operated valves. In the area of power supplies, the team reviewed the 125 Volt (V) and 250V direct current (dc) supplies associated with the HPCI motor operated valves (MOVs), dc pump motors and the HPCI inverters. The team also reviewed the electrical protection provided for selected HPCI MOVs.

.3.1 Design - Mechanical, Electrical and Instrumentation and Controls

a. Inspection Scope

The HPCI system is powered from the 125 V “B” battery and the 250 V battery. In 1994 and 1995, the licensee replaced the existing batteries with GNB model NCN-35 cells rated 2550 Amp-Hours at the eight-hour rate. These cells were used in all the 125 V (60 cells) and the 250 V (120 cells) batteries. According to the system engineer, these cells were selected for all the HPCI and RCIC batteries to provide additional margins and to reduce the required inventory of spare cells. The team reviewed the battery sizing calculations to determine the calculated sizing margin under design basis conditions. The team reviewed other DC system analyses to determine the adequacy of the voltage supplied to safety-related HPCI loads. The team reviewed the results of recent surveillances and tests to determine the adequacy of assumptions used in the analysis.

b. Findings

The team observed that the licensee had established an acceptance criterion for battery voltage during the 250V battery service test of 227 volts. This value was obtained from their calculated battery voltage at the end of the eight-hour discharge. The team found, by independent calculation, that the voltage at other periods during the discharge would be both higher and lower than the acceptance criterion. PNPS acknowledged that these other values had been used in the voltage drop analysis and issued PR00.9503 to address the use of the incorrect acceptance criteria. This was a concern applicable to all the batteries in the PNPS 125V and 250V systems. Failure to correctly translate the design into the test procedure was the first example of a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) because the latest test voltage did not drop to the correct minimum voltage level. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue associated with this violation is in the PNPS corrective action system as problem report PR00.9503. **(NCV 05000293/2000-012-02)**

Industry standard IEEE 485, Sizing Large Station Batteries, provides guidelines for calculating battery size. This standard is referenced in the Pilgrim dc analyses. The standard identifies different factors which should be evaluated in selecting the design margin correction factor, including less than optimum battery maintenance. The team identified that the licensee had used a design margin correction factor of 1.0 in the battery sizing and voltage drop analyses. The team observed that the acceptable minimum specific gravity specified in the battery quarterly surveillance was 1.205. The nominal specific gravity of the fully charged Pilgrim's batteries is 1.215. Using the industry rule-of-thumb that three points (.003) of specific gravity equate to 1% capacity, a design margin correction factor of at least 1.03 should have been used in the battery sizing and voltage drop calculations. The minimum margin indicated in the calculations for the HPCI batteries was 4.9% based upon rounding up the required number of positive plates to the next integral number. The licensee issued problem report 00.3478 to address this issue. The failure to adequately account for all the parameters that make up the design margin correction factor was a second example of a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) because of adequate compensating margins included in the body of the calculation. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue associated with this violation is in the PNPS corrective action system as PR 00.3478. **(NCV 05000293/2000-012-02)**

.3.2 Operations and Maintenance

a. Inspection Scope

The team reviewed selected equipment technical manuals and operating, surveillance and preventive maintenance procedures to assess whether the selected systems and components were being operated, tested and maintained in accordance with the system requirements. The team performed a walkdown of the dc power supplies with the system engineer and the electrical engineering manager. The team reviewed the QA records for the latest battery service duty and performance discharge tests for the batteries powering the 125 and 250 V dc HPCI motor operated valves.

b. Findings

The team observed that the batteries were maintained in excellent physical condition.

The service duty tests on the HPCI batteries were performed in May 1999. The test procedures indicated the required discharge profiles. However, the team found that the test discharge record of the battery voltage and current for the 250 V battery indicated that zero current was recorded during the eight-hour discharge test. The 250 V battery is also the source of power for other dc loads such as shutdown cooling system isolation valve MO-1000-47 in the residual heat removal (RHR) system, the recirculation pump auxiliary dc lube oil pumps, the dc supply to the vital MG set, the main turbine emergency oil pump and the main generator emergency seal oil pump. The licensee entered this concern into the corrective action system as problem report PR00.9477.

The team also identified that the 125 V "B" battery service test failed to record the battery voltages during the critical first hour of the test when the load currents were the highest and the voltage dips the lowest. (During the review of these concerns, the licensee discovered that the 125 V "A" battery, the power supply for the RCIC MOVs, also had not been correctly discharged according to its service test duty cycle.) The failure to adequately review the results of the service test data was a violation of 10 CFR

50, Appendix B, Criterion XI, Test Control. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) based on the results of special tests performed by PNPS during the inspection. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue associated with this violation is in the PNPS corrective action system as PR00.9494. **(NCV 05000293/2000-012-03)**

Performance tests to the manufacturer's rated eight hour discharge current for the NCN-35 cells were performed in March 1997 for the 250 V battery and in September 1996 for the 125 V "B" battery. Both test records recorded capacities over 100%. However, the team found that there was no data included in the QA test record that substantiated the recorded results for the "B" battery. Therefore the team was unable to independently verify the recorded capacity because of the missing data. Failure to properly maintain required QA records was a violation of 10 CFR 50, Appendix B, Criterion XVII, QA Records. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) because the results of those tests are used solely for aging determination and the batteries are relatively new. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue associated with this violation is in the PNPS corrective action system as PR00.9503.
(NCV 05000293/2000-012-04)

Overall, in spite of the concerns identified with the HPCI power supplies, nothing that the team found suggested that the power supplies for the HPCI valves would not be able to perform their safety function. This conclusion was based on the excellent physical condition of the batteries, the large capacity and conservative sizing calculation, the special tests performed during the inspection on representative single cells in the warehouse, and the results from the 1995 service duty and 1997 performance tests.

1R02 Evaluations of Changes, Tests, or Experiments (IP71111.02)

a. Inspection Scope

The team reviewed selected safety evaluations (SEs) performed by the Pilgrim staff. The SEs were selected from a list of changes relating to the HPCI and Containment systems and from other plant changes implemented during the last year. The selections took into consideration safety significance of the change, risk to the structures, systems, and components affected, and impact on three reactor safety cornerstones (initiating events, mitigating systems, and barrier integrity).

The review was conducted to verify those changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and test and experiments not described in the UFSAR, were reviewed and documented by the licensee in accordance with 10 CFR 50.59. The review also verified that any safety issues pertinent to the changes, tests and experiments had been properly resolved. Additionally, the team verified the licensee's conclusions that the changes, tests and experiments did not require prior NRC approval or a license amendment was appropriate. The team conducted discussions with cognizant engineers, as required, and evaluated supporting technical information, including calculations, analyses, and design requirements.

The team also reviewed a sample of changes, tests and experiments for which the Pilgrim staff determined that a safety evaluation was not required. This review was performed to verify that the threshold for performing safety evaluations was consistent with the requirements of 10 CFR 50.59. Lastly, the team verified that problems associated with the implementation of the safety evaluation program were entered into the corrective action program.

b. Findings

The team found that a 10CFR 50.59 screening review associated with a revision to a HPCI system surveillance test had inappropriately justified a change to the pump discharge acceptance criteria within the test. This issue is described in Section 1R21.1.2 of this report. No other findings were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed the licensee's effectiveness in identifying problems associated with the high pressure coolant injection and containment systems and the electrical features that enable these systems to function during a loss of offsite power. The team also reviewed a sample of event reports related to the selected systems to evaluate the adequacy and timeliness of the corrective actions resulting from the identified problems. For selected event reports the team reviewed the adequacy of the operability determinations and verified the completion of the corrective actions.

b. Findings

Background

The team reviewed problem report (PR) 00.9347 associated with HPCI failing to meet acceptance criteria during a HPCI pump and valve operability test performed on September 1, 2000. Safety Evaluation 3317, dated August 22, 2000, defined the design basis performance requirements for the HPCI system, and supported a revision to the pump test surveillance requirements. Prior to the safety evaluation being approved, the HPCI pump was tested to provide 4,250 gallons per minute (gpm) of flow against a system head corresponding to a reactor pressure of 1,126 psig. A discharge pressure of 1,230 psig at the pump discharge was required to account for elevation and dynamic losses at this flowrate. The reactor pressure of 1,126 psig was the upper analytical setpoint of the SRV's. The safety evaluation provided justification for lowering the test requirements to 4,250 gpm for a system head that corresponds to normal reactor pressure, consistent with Technical Specification 4.5.C requirements. The previous criterion of testing the capability of injection against a pressure of the SRV setpoint was revised to reduce the flow requirement at this pressure to 3,000 gpm, thereby resulting in two acceptance criteria established within the surveillance test.

On September 1, 2000, the HPCI pump satisfied the first criterion but failed to achieve the second criterion of 3,000 gpm against an equivalent pump discharge pressure (1,230 psig) which correlated to reactor pressure of 1,126 psig when considering elevation and system dynamic losses. The test results therefore demonstrated that the pump met the technical specification requirement by providing 4,250 gpm at 1,130 psig pump discharge pressure, but failed the criteria established to show injection flowrate capabilities against an SRV setpoint. This second criterion demonstrated the ability of the HPCI system to provide adequate core cooling in the event of reactor isolation with a loss of feedwater.

During the investigation of the failure, PNPS found that the HPCI flow controller, FIC-2340-1, high limit setting was out-of-specification low. The controller potentiometer was adjusted in accordance with procedure 8.E.23.1, "HPCI Instrumentation Calibration", and the surveillance test was subsequently performed on September 3, 2000, and resulted in the pump achieving 4,250 gpm at 1,250 discharge pressure, satisfying both acceptance criteria.

.1 HPCI Speed Control Degradation

The team reviewed the last completed HPCI instrumentation calibration procedure results performed on February 29, 2000, and noted that during the calibration check of the flow controller the as-found setting of the high speed limit had been 48.7 milliamps. The normal range documented in the procedure was between 51 and 53 milliamps. A setting of 50 milliamps corresponds to the nominal rated HPCI turbine speed of 4,000 rpm. The team noted that although the data indicated the flow controller was degraded and would have limited speed below design, that no problem report was generated by the licensee. The setting was adjusted back within the range and left at

52 milliamps which corresponds to a nominal maximum turbine speed setting of 4160 rpm. The team noted that operators had trouble achieving the required flow and pressure criteria during a subsequent pump, valve and flow test performed in June 2000. Problem Report 00.9221.00 was initiated at the time, which suggested in the apparent cause section that the flow controller was not satisfied thereby asking for more flow but flow was limited by the turbine upper speed clamp. The team compared the computer traces from this run in June to other successful runs performed and concluded that there was degradation with the flow and speed parameters. Several groups had been involved with the review of the issue including I&C, Mechanical Systems, and Safety and System Analysis; however, no further corrective actions were initiated and the problem report was closed. As documented above, the subsequent test in September failed to achieve a reduced flow criteria of 3,000 gpm against a discharge pressure of an SRV setpoint.

The team determined that several opportunities had existed for the licensee to identify and correct the degraded condition of the flow controller. Specifically, in February 2000, the controller setting was found degraded and adjusted back into the correct range. The team determined that the June testing results indicated the controller had degraded to a setting below the range of 51-53 ma based on speed output and flow degradation as shown by the computer traces. Finally, in September, the turbine was limited to a speed of 3,820 rpm which was not able to satisfy the acceptance criteria of 3,000 gpm against a discharge pressure equivalent to the SRV lift setpoint. The licensee had determined in September that with the degraded speed, a flow of 1,000 gpm would be achieved at a maximum pressure of 1,200 psig pump discharge, thereby potentially not achieving the function of injection against an SRV.

Based on performance issues associated with weak problem identification and corrective action concerning the HPCI pump speed flow controller, the team evaluated this issue utilizing the SDP. Although the system had achieved the criterion associated with a small break LOCA accident of 4,250 gpm against normal reactor pressure, the team noted that the HPCI system, if it did not meet its function associated with the Transient and loss of offsite power (LOOP) scenarios, had more than very low risk significance (non-Green). This function consists of the ability to maintain core coverage during loss of feedwater events similar to the RCIC function. The design response for these events would result in injection against the lowest SRV setpoint.

In response to the team’s questions during this inspection, the licensee performed analysis EE No. 00-072 which developed a pump curve based on a degraded speed condition of 3,800 rpm. The licensee used best estimate dynamic losses and elevation differences assuming the lowest water level in the condensate storage tanks (CST) to evaluate whether they could have achieved 320 gpm against the safety relief valve (SRV) setpoint. After a loss of feedwater and vessel isolation event, with the SRV setpoint at the upper analytical value of 1,126 psig, 320 gpm makeup from the RCIC or HPCI system is sufficient to maintain reactor water level above the top of active fuel. The hydraulic analysis determined that the flowrate developed would have stayed above the minimum flow valve setpoint and would have been a nominal 1,500 gpm against the SRV upper analytical setpoint.

Based on the determination that although degraded, the system could have met its safety function regarding the Transient and LOOP scenarios, the team screened this issue out as Green (very low safety significance) in phase 1 of the SDP. Notwithstanding this determination, the team concluded that the licensee's handling of this issue showed weak problem identification and corrective action relative to a risk significant component and system. Additionally, the failure in September had initially been classified as requiring a root cause but had been subsequently revised to a direct cause determination and therefore no root causes for the failure had been identified. The due date had already been extended once and was not due until December 29, 2000. The team noted that this revision had taken place prior to the licensee determining that they would still have satisfied the required function during a Transient and LOOP event when the controller had been degraded. This signified a poor understanding of the potential risk implications of allowing degradation with the speed control of this risk significant system.

The team reviewed the latest, November 2000, surveillance test results via a detailed examination of the computer traces, and noted that the speed was a nominal 4,150 rpm with the flow setpoint of 4,250 gpm achieved at a pump discharge pressure of 1,230 psig. There were no indications of any abnormalities in the speed signal noted. This indicated that no further degradation had taken place with the controller since I&C had performed potentiometer adjustments in September. Although no direct cause had ever been identified and documented with the HPCI pump speed control prior to this inspection, based on the recent test results the team concluded that the speed control remained operable. The licensee had believed the problem to be a sticky potentiometer which was "wiped" by exercising the component in September. The licensee stated they are evaluating replacing the controller in the near future.

The team determined these issues to be a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, based on the failure to identify the degraded speed controller and correct it prior to failing the surveillance test criteria. The issue was evaluated using the Significance Determination Process (SDP) and determined to be Green (very low safety significance) because system operability was demonstrated. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issues associated with this violation are in the PNPS corrective action system as PR00.3298, PR00.9121 and PR00.3382. The licensee initiated PR00.3298 which acknowledged deficiencies in their evaluation of this issue concerning not entering problem reports when not within expected ranges and their flawed evaluation of the degradation in June during their review of PR00.9221.00. Additionally, the licensee initiated PR00.3382 to evaluate the lack of long term risk assessment associated with changes that are made to procedures, calculations and design documents. **(NCV 05000293/2000-012-05)**

No other findings were identified.

4OA6 Meetings, Including Exit

.1 Management Meeting

The team presented the inspection results to Mr. R. M. Bellamy, and other members of licensee management at an exit meeting on December 14, 2000. The licensee acknowledged the inspection findings presented.

(1) NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.
More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

(2) SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Pilgrim Nuclear Plant

R. Bellamy	Vice President
J. Bonner	Manager, Electrical Design Engineering
R. Byrne	Sr. Engineer
W. Carroll	System Engineer
D. Cooper	Electrical Maintenance Lab
J. Coughlin	Electrical Design Engineer
L. Darsney	System Engineer
S. Das	Electrical Design Engineer
R. DiVario	Electrical Design Engineer
J. Falconieri	Maintenance Support Engineer
R. Gay	DC System Engineer
K. Kampschneider	I&C System Engineer
J. Keyes	Manager of Corrective Action
T. McElhinney	Systems Engineering Manager
F. Mogolesko	Project Manager
K. O'Brien	Electrical Maintenance Lab
V. Oheim	Director, Design Engineering
D. Richard	I&C Design Engineer
W. Riggs	Engineering Director
R. Sheridan	QA Manager
D. Sitkowski	Electrical Design Engineer
T. White	Manager, Design Enginering
M. Williams	System Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

--05000293/1998-203-05	IFI, EDG Analysis. This item is in the PNPS tracking system and is being completed. There are no unresolved regulatory issues.
--05000271/1998-203-06	IFI, Backfeed Analysis. This item is in the PNPS tracking system with a plan for completion.

Opened/Closed

(NCV 05000293/2000-012-01) The failure to assure that surveillance test acceptance criteria ensured operability of HPCI equipment at the remote shutdown panel was considered a 10 CFR Part 50, Appendix B, Criterion XI, Test Control, non-cited violation. The issue was entered into the licensee's corrective action program as PR 00.3464.

(NCV 05000293/2000-012-02) Failure to correctly translate the design battery acceptance criteria into the test procedure was the first example of a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, in the electrical area.

(NCV 05000293/2000-012-02) The failure to adequately account for all the parameters that make up the design margin correction factor was a second example of a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, in the electrical area.

(NCV 05000293/2000-012-03) The failure to adequately review the results of the battery service test data was a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control.

(NCV 05000293/2000-012-04) Failure to properly maintain required QA records of battery test parameters was a non-cited violation of 10 CFR 50, Appendix B, Criterion 17, QA Records.

(NCV 05000293/2000-012-05) The failure to identify the degraded HPCI speed controller and correct it prior to failing the surveillance test criteria was a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action.

(3) LIST OF DOCUMENTS REVIEWED

Design Drawings

E1	Station Single Line Diagram
E7	4160 Volt System Single Line Diagram
E8	480 V Load Center Single Line Diagram
E9	480 V Load Center and MCC Single Line Diagram
E10	480 V MCC Single Line Diagram
E13	125 &250 V DC Single Line Diagram
E14	120 V Instrument AC Single Line Diagram
E404	Containment Atmosphere Control System Schematic Diagram
E727	ECCS Analog Trip Unit Elementary Diagram
E728	ECCS Analog Trip Unit Elementary Diagram
E729	ECCS Analog Trip Unit Elementary Diagram
E730	ECCS Analog Trip Unit Elementary Diagram
MIA-15-7	Nuclear Boiler Functional Control Diagram
MIA16-5	Nuclear Boiler Functional Control Diagram
MIA69-1	Nuclear Boiler Functional Control Diagram
MIH7-12	Residual Heat Removal Elementary Diagram
MIH9-12	Residual Heat Removal Elementary Diagram
MIH13-6	Residual Heat Removal Elementary Diagram
MIJ15-10	HPCI Elementary Diagram
MIJ16-10	HPCI Elementary Diagram
MIJ19-9	HPCI Elementary Diagram
MIJ18-11	HPCI Elementary Diagram
MIJ20-5	HPCI Elementary Diagram
MIJ22-5	HPCI Functional Control Diagram
MIJ23-4	HPCI Functional Control Diagram
MIJ24-4	HPCI Functional Control Diagram
MIJ32	HPCI Elementary Diagram
MIK4-11	Core Spray Elementary Diagram
MIN33-10	Primary Containment Isolation System Elementary Diagram
MIN34-9	Primary Containment Isolation System Elementary Diagram
MIN39-13	Primary Containment Isolation System Elementary Diagram
M63A3	Pneumatic Details, Secondary Containment Isolation Dampers
M209	Condensate and Demineralized Water Storage P & ID
M220	Compressed Air System
M227	Containment Atmospheric Control System
M232	Radwaste Collection System
M241	Residual Heat Removal System P & ID
M243	HPCI P & ID
M244	HPCI P & ID
M253	Nuclear Boiler Vessel Instrumentation
M280	HVAC Temperature Control Diagram for Turbine, RW, Intake & EDG Bldg.
M283	Secondary Containment Isolation Control Diagram
M294	Standby Gas Treatment System

Engineering Calculations

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

00025-0042-001	DC System Analysis, Methodology and Scenario Development
00025-0042-002	250 Volt Battery System Voltages
00025-0042-004	125 Volt Battery B System Voltages
I-N1-57	Setpoint calculation for HPCI Pump Suction Low Pressure Turbine Trip
I-N1-59	Setpoint verification for Torus High Level Switches
I-N1-96	Setpoint calculation for LIS263-72A, B, C & D
I-N1-97	Setpoint calculation for LIS263-72A-1, B-1, C-1 & D-1
I-N1-98	Setpoint calculation for LIS263-72A-2, B-2, C-2 & D-2
I-N1-137	Setpoint calculation for PIS1001-89A-3, B-3, C-3 & D-3
M416	Calculation Drywell-to-Torus Leak Rate Test Curve
M553	Maximum Pressure Differential for DC MOVs, Rev. 4
M600	MOV Pressure Locking and Thermal Binding Evaluation, Rev. 3
M679	Maximum Allowable Daily Leakage Rate of the SBGT Air Accumulators, Rev. 0
M911	HPCI System Hydraulic Calculation, Rev. 1
M895	RCIC And HPCI NPSH and Suction Line Pressure Drop, Rev. 0
PS31-5	DC System Over-current Protection Coordination Study
PS79	Emergency Diesel Generator Loading
PS79-5-1	Emergency Diesel Generator Loading
PS140	Thermal Overload Sizing for Priority 1 MOVs(DC)
PS142	Thermal Overload Sizing for Priority 2, 3, 4, 5 and 6 MOVs
PS233C	125 Volt Battery B System Voltage Calculation
PS233D	250 Volt Battery System Voltage Calculation

Design Bases Documents

Reference Text	Primary Containment Isolation System, Dec. 1998
SDBD-09	Primary Containment Pressure Suppression
SDBD-23	HPCI System Design Basis Document
SDBD-48	Standby Gas Treatment System (SBGT)
SDBD-50	Primary Containment Structure
UFSAR 4.10	Nuclear System Leakage Rate Limits
UFSAR 5.0	Containment
UFSAR 5.3	Secondary Containment System
UFSAR 6.5.2.2.	HPCI
UFSAR 7.3	Primary Containment and Reactor Vessel Isolation Control System
UFSAR 7.4.3.2	HPCI Control and Instrumentation
UFSAR 12.2	Structural Design
UFSAR, App. 0	Analysis of the Consequences of High Energy Piping Failures Outside the Primary Containment.
UFSAR, App. R.3	Analysis of Design Basis Accidents

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

Engineering Design Changes and Design Change Requests

FRN-93-37-02	DC MCC Breaker and Combination Starter Replacement
FRN-93-38-01	Replacement of GE Thermal Overload Relays with Westinghouse Units
FRN-93-38-08	Resizing of Thermal Overload Heaters
PDC Reg 45	Removal of Flow Switch From HPCI Cooling Water and Remove Alarm
PDC 00-25	New Accumulator Air Compressor and SBGT Capacity Expansion
TM 00-29	Temporary Modification C19A/B (East/West)

Problem Reports (PR)

PR 97.9136	6-CK-58B Failed RFO #11 As Found LLRT
PR 97.9137	6-CK-58A Failed RFO #11 As Found LLRT
PR 99.0054	HPCI and RCIC ST Outside Design Bases
PR 99.9216	Local Leak Rate Testing of Main Steam Isolation
PR 99.9226	During LLRT of the FW Outboard Containment Isolation Check Valve, 6-CK-62B, an Unacceptable Leakage Rate of 145.6 Standard Liters/Minute Was Found
PR 00.3464	HPCI Alternate Shutdown Panel Acceptance Criteria
PR 00.3465	HPCI Flow Controller High and Low Limit Outside Range
PR 00.3477	Unconservative Reset tolerance for Alt Shutdown HPCI Inverter Trip
PR 00.3478	Inadequate Design Margin Correction Factor in DC System Calculations
PR 00.3479	Incorrect Alarm Setpoint Used in 250 V Battery Service Discharge Test
PR 00.9083	There is a Gap Between the Floor and the Bottom Seal of the Outer Reactor Truck-lock Door
PR 00.9130	SBGT Air Accumulator Leakage
PR 00.9143	SBGT System Charcoal Efficiency at >70% Relative Humidity
PR 00.9221	HPCI CV Indicated Mid-position While Flow Controller Demand Called for Full Open
PR 00.9287	While Performing Preventative Maintenance on Inner Reactor Building Truck-lock Door #45 Found Rubber Bulb Seal Damaged in Two Areas
PR 00.9347	HPCI Failed to Meet Acceptance Criteria During 8.5.4.1
PR 00.9477	Inadequate Testing and Documentation of 250 V Battery Service Test
PR 00.9480	Infrequently Performed PM on Alternate Shutdown HPCI Inverter
PR 00.9494	125 Volt A Battery Service Missing 33 minutes of Required Discharge
PR 00.9501	Questionable Operation of the Battery Test Equipment
PR 00.9503	125 Volt B Battery Documentation Missing the First Hour of Records

Licensee Event Reports (LER)

LER 99-003-00	Local Leak Rate Test Results Exceeding Allowable Technical Specification Leakage Rates

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

Station Procedures

2.1.15	Daily Surveillance Log (Technical Specification Regulatory Agencies)
2.2.70	Primary Containment Atmospheric Control System
2.2.77	Drywell Leak Detection Systems
2.2.21	HPCI, Rev. 55
2.2.21.5	HPCI Injection and Pressure Control, Rev. 10
2.5.2.71	Radwaste Collection Facility
5.3.12	Loss of Essential DC Bus
5.3.21	Bypassing Selected Interlocks
5.3.31	Station Blackout
8.5.4.1	HPCI System Pump and Valve Quarterly Operability
8.5.4.1-1	HPCI Simulated Automatic Actuation, Flow Rate and Cold Quickstart Test
8.5.4.4	HPCI Valve Operability Test
8.5.4.6	HPCI Pump and Valve Operability From Alternate Shutdown Panel
8.7.1.19	Pressure Drop Test of Air Supply for Standby Gas Treatment System Dampers
8.7.1.9	Local Leak Rate Test of Primary Containment Penetrations, Isolation Valves and Inspection of Containment Structure
8.7.3	Secondary Containment Leak Rate Test, 8/29/1998
8.7.4.9	Suppression Chamber to Reactor Building Vacuum Breaker Quarterly Operability
8.9.8.1	125 V DC “A” Battery Performance Test
8.9.8.2	125 V DC “B” Battery Performance Test
8.9.8.3	250 V DC Battery Performance Test
8.A.1	Drywell to Torus Vacuum Breaker Monthly/Quarterly Operability
8.A.2	Drywell to Suppression Chamber Vacuum Breaker Leak Rate Test
8.C.14	Weekly Pilot Cell, Overall Battery Check and Battery Charger Test
8.C.16.1	250 Volt Battery Quarterly Inspection/Surveillance
8.C.16.2	125 Volt A Battery Quarterly Inspection/Surveillance
8.C.16.3	125 Volt Battery Quarterly Inspection/Surveillance
8.C.36	Suppression Chamber to Reactor Building Vacuum Breaker Accumulator Leakage Monitoring
8.E.23	HPCI System Instrumentation Calibration
8.E.23.1	HPCI Turbine Speed Control System Calibration
8.I.1.1	Inservice Pump and Valve Testing Program
3.M.3-42	Battery Charger Maintenance and Calibration
8.M.1-32.5	Analog Trip Cabinet C2233A File Calibration
8.M.1-32.7	Analog Trip Cabinet C2233B File Calibration
8.M.2-1.5.8.3	Logic System Functional Test of System SBGT Initiation, Reactor Building Isolation and Inboard Drywell Isolation Valves
8.M.2-5	Drywell Drain Sump Integrator
8.M.2-8.1	Analog Trip System Rack C2205 Transmitter Calibration
8.M.3-4	Reactor Building to Suppression Chamber Vacuum Breaker Calibration
8.Q.3-10	Maintenance of Step-down Transformers
ARP-C20L	Alarm Response Procedure
ARP-C3RC	DC Annunciator Alarm Response Procedure
ARP-C904LC	Alarm Response Procedure

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

NOP83E5	Safety Reviews
TP99-026	Temporary Procedure for “A” 125 V DC Battery Service Test
TP99-027	Temporary Procedure for “B” 125 V DC Battery Service Test
TP99-056	Temporary Procedure for 250 V DC Battery Service Test

Safety Evaluations

3317	Establishment of Design Bases and HPCI Performance Requirements
3294	Test HPCI, RCIC-Measure Pump performance at Reduced Flowrate
3260	Revised Pump Test A.C. Based on Design Bases Hydraulic Analyses and System Performance Requirements
3275	Installation of Blocking Diodes on HPCI Drain Pot Valves
3308	Temp Mod 00-015, HPCI Inverter 2340-13 Input Monitor
3322	Addition of Fifth Standby Gas Treatment System Air Accumulator Tank and Installation of a Non Safety-Related Air Compressor
3324	Addition of Four 1" Isolation Valves in N2 Supply Lines to Inboard MSIVs
3255	Evaluate Applicability of Conditions Evaluated in OP 98.069 to Design/Licensing Basis of RBCCW and Primary Containment and Clarify FSAR Accordingly for 50.59
3296	Remove and Replace Reactor Building Roofing Material

Preliminary Evaluations (50.59 Screens)

Calc. M-998	Design Basis for CST Reserve Capacity for HPCI/RCIC
Calc. IN 1-269	Loop Accuracy Calculation, 33' El. Meteorological Tower Air Temperature, EPIC Point MTR-006, Rev. 0
EPIP 210	Control Room Procedure Augmentation
FRN-93-37-02	DC MCC Breaker and Combination Starter Replacement
FRN-93-38-01	Replacement of GE Thermal Overload Relays with Westinghouse Units
FRN-93-38-08	Resizing of Thermal Overload Heaters
FRN-96-21	Install Sentry Analyzers for "A" RHR/CS Quadrant
FRN-97-01-11C	Installation of Drywell Sump Pump Relief Assembly
FRN 98-21-10	Relocation of Current Transformer "D" of the Quick Test Cable Assembly for MO-1001-28A
FRN-98-21-11	Limit Switch Setting for MO-1400-25A
FRN-00-02-29	Replacement 480V MCC Starters for "A" RHR LCO
FRN-00-04-06	Replace RBCCW Flow Transmitters FT-6263 & FT-6265
FRN-00-04-22	Replacement of Obsolete ENTEK IRD Signal Conditioning Card With Universal Card P/N 34294 at C2263 for Recirculation Pump 'B' Vibration Monitor VIS-202-1B
PCF-99-12915	HPCI Pump and Valve Operability from Alternate Shutdown Panel
Procedure 8.5.4.6	HPCI Pump and Valve Operability from Alternate Shutdown Panel
Procedure 8.E.23	HPCI System Instrumentation Calibration, 12/30/1998
Procedure 8.E.23	HPCI System Instrumentation Calibration, 3/30/1999

Engineering Evaluations (EE) and Operability Evaluations (OE, OP)

(3) LIST OF DOCUMENTS REVIEWED (Cont.)

EE-00-011	Gaps in Reactor Building Truck Lock Outer Door Seal for Secondary Containment, Rev. 2, 5/23/2000
EE-00-068	Engineering Evaluation 250 Volt DC Power Battery
EE-00-070	Engineering Evaluation 125 Volt DC Train A System
OE-00-008	SBGT Air Accumulator Leakage Testing
OE-00-009	SBGT heater capacity less than required to maintain 70% RH
OE-00-005	Reactor Building Truck-Lock Door Lower Seal Air Gap
OE-00-029	Reactor Building Inner Truck-Lock Door
OE 00-029	Gaskets at Secondary Containment System Truck-lock Door 45.
OE-98.0025	A potential condition exists that could create a direct vent path from the containment to atmosphere
OE-00-040	RCIC Valve MO-1301-17, Body to Bonnet Steam Leak to Reactor Building
OE-00-032	C19B (East)Flow Control Valve Removed
OP00.0017.00	X57 Transformer Out-of-Spec Low

Technical Specifications

TS 3.2	Protective Instrumentation
TS 3.6	Primary System Boundary
TS 3.7	Containment Systems
T.S. 4.12.5	Surveillance Requirement 4.12.5, Alternate Shutdown Panel
T.S. Bases 3/4.5	High Pressure Coolant Injection System

Miscellaneous

GL 96-06	Assurance of Equipment Operability and Integrity During Design-Basis Accident Conditions
QA16.03	Quality Assurance Audit on Documentation, Tracking, and Follow-up of QA Identified Problems
SEE-953	Substitution Equivalency Evaluation, 11/6/2000
SEE-955	Substitution Equivalency Evaluation, 12/13/2000
V-0257	HPCI Turbine Vendor Manual Rev. 29
V-0451	General Precision System Vacuum Breaker Valves

Maintenance Rule Summary for the Month of October 2000

(4) LIST OF ACRONYMS USED

AC or ac	Alternating Current
CFR	Code of Federal Regulation
cfm	Cubic feet per minute
CST	Condensate Storage Tank
DBD	Design Basis Document
DC or dc	Direct Current
ER	Event Report
FRN	Field Revision Notice
FW	Feed Water
GE	General Electric
gpm	gallons per minute
HPCI	High Pressure Coolant Injection
LCO	Limiting Condition for Operation
LLRT	Local Leakage Rate Testing
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MTR	Material Test Report
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
OE	Operability Evaluation
PDC	Plant Design Change
PNPS	Pilgrim Nuclear Power Station
PR	Problem Reports
psi	Pounds per square inch
psig	pounds per square inch gauge
RCIC	Reactor Coolant Injection System (Reactor Core Isolation Cooling)
RH	Relative Humidity
RHR	Residual Heat Removal
SBGT	Standby Gas Treatment System
SDBD	System Design Basis Document
SDP	Significancy Determination Process
SE	Safety Evaluation
SGTS	Standby Gas Treatment System
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report