May 4, 2006

Mr. L. William Pearce
Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
10 Center Road, A290
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC COMPONENT DESIGN BASES INSPECTION (CDBI)
INSPECTION REPORT 05000440/2006009(DRS)

Dear Mr. Pearce:

On March 23, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on March 3, 2006, with you, and on March 23, 2006, with Mr. J. Shaw, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Specifically, this inspection focused on the design of components that are risk significant and have low design margin.

Based on the results of this inspection, three NRC-identified findings of very low safety significance, which involved violations of NRC requirements were identified. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Perry Nuclear Power Plant.

In accordance with 10 CFR 2.390 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosure: Inspection Report 05000440/2006009(DRS)
w/Attachment: Supplemental Information

cc w/encl: G. Leidich, President - FENOC
J. Hagan, Chief Operating Officer, FENOC
D. Pace, Senior Vice President Engineering and Services, FENOC
Director, Site Operations
Director, Regulatory Affairs
M. Wayland, Director, Maintenance Department
Manager, Regulatory Compliance
T. Lentz, Director, Performance Improvement
J. Shaw, Director, Nuclear Engineering Department
D. Jenkins, Attorney, FirstEnergy
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

In accordance with 10 CFR 2.390 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosure: Inspection Report 05000440/2006009(DRS)
w/Attachment: Supplemental Information

cc w/encl: G. Leidich, President - FENOC
J. Hagan, Chief Operating Officer, FENOC
D. Pace, Senior Vice President Engineering and Services, FENOC
Director, Site Operations
Director, Regulatory Affairs
M. Wayland, Director, Maintenance Department
Manager, Regulatory Compliance
T. Lentz, Director, Performance Improvement
J. Shaw, Director, Nuclear Engineering Department
D. Jenkins, Attorney, First Energy
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

DOCUMENT NAME:ML061250451.wpd
X Publicly Available ☐ Non-Publicly Available ☐ Sensitive X Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII		RIII			
NAME	ADunlop: ls		EDuncan		AMStone			
DATE	5/03/06		5/05/06		5/04/06			

OFFICIAL RECORD COPY

ADAMS Distribution:
GYS
WAM1
RidsNrrDipmIipb
GEG
JAC
LTD
DPN
CRO
KGO
RJP
CAA1
DRPIII
DRSIII
PLB1
JRK1
WDL (IR's only)
ROPreports@nrc.gov

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	Inspection Report 05000440/2006009(DRS)
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Perry Nuclear Power Plant
Location:	PO Box 331 10 Center Road Perry, Ohio 44081-0331
Dates:	January 17, 2006, through March 3, 2006, and March 23, 2006
Inspectors:	A. Dunlop, Senior Reactor Engineer, Lead Inspector W. Cook, Senior Reactor Analyst J. Jandovitz, Reactor Engineer D. Reeser, Operations Inspector H. Walker, Senior Reactor Engineer O. Mazzoni, Electrical Contractor W. Sherbin, Mechanical Contractor
Approved by:	A. M. Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000440/2006009(DRS); 01/17/2006 - 03/23/2006; Perry Nuclear Power Plant; Component Design Bases Inspection.

The inspection was a 4-week onsite baseline inspection that focused on the design of components that are risk significant and have low design margin. The inspection was conducted by regional engineering inspectors and two consultants. Three Green Non-Cited Violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving an inadequate stress analysis performed for the automatic depressurization system (ADS) air accumulators. Specifically, the licensee failed to account for all the related stresses in the ADS accumulator stress analysis calculation. Inclusion of these additional stresses resulted in a higher stress than allowed by the American Society of Mechanical Engineers Code. Additionally, the accumulators' certification of design, as required by the Code, Section III, was incorrect as it was not based on the maximum design pressure the accumulators would be subject to under accident conditions. The licensee's corrective actions included performing an operability determination that concluded the ADS accumulators would not fail structurally under design conditions.

  The finding was more than minor because the failure to adequately evaluate the design requirements of the accumulators could have led to structural failure of the tanks, which would have prevented the ADS valves from functioning as designed and could have affected the mitigating systems cornerstone objective of design control. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.1)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the sizing of the main steam isolation valve and automatic depressurization system (ADS) air storage tank. The inspectors identified that the licensee failed to correctly specify in a design calculation the required minimum differential air pressure required to actuate the ADS valves when manually operated. This resulted in a safety-related air system calculation that was non-conservative when determining the long-term air volume requirements in the air storage tank. The licensee's corrective actions included verifying that adequate design margin existed for the air tank capacity and entered this performance deficiency into their corrective action program for resolution.

The finding was more than minor because the failure to adequately evaluate air storage tank sizing could result in over-predicting the tank's capacity as verified by the surveillance test's acceptance criteria (i.e., creating design margin capability that would not exist) and could have affected the mitigating systems cornerstone objective of design control. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.2)

C Green. The inspectors identified a Non-Cited Violation of Technical Specification Requirement 5.4.1, which requires, in part, that written procedures/instructions be established, implemented, and maintained covering the emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1. The anticipated transient without scram (ATWS) special plant instructions issued to provide for injection outside the shroud were inadequate because the procedures inappropriately limited the ability to control reactor water level (or reactor pressure if reactor water level is unknown). The licensee entered this performance deficiency into their corrective action program for resolution.

This finding was more than minor because the procedure deficiency affected the ability of the licensee to use the low pressure coolant injection sub-systems to prevent undesirable consequences of large power excursions associated with an ATWS, and was associated with the mitigating systems procedure quality attribute of the mitigating systems cornerstone objective. The finding was of very low safety significance because no actual initiating event or transient occurred and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.6.b.1)

B. <u>Licensee-Identified Violations</u>

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify that design bases have been correctly implemented for the selected risk significant components and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The Probabilistic Risk Assessment (PRA) model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectible area verifies aspects of the initiating events, mitigating systems and barrier integrity cornerstones for which there are no indicators to measure performance. Specific documents reviewed during the inspection are listed in the attachment to the report.

.2 Inspection Sample Selection Process

The inspectors selected risk significant components and operator actions for review using information contained in the licensee's PRA and the Perry Standardized Plant Analysis Risk (SPAR) Model, Revision 3.21. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 2.0 and/or a risk reduction worth of greater than 1.005. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios. Since all plant components were not modeled in the licensee's PRA, additional resources were used in the selection process such as the licensee's maintenance rule program, where an expert panel identified additional systems/components that also were considered risk significant.

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design, reductions caused by design modifications or power uprates, or reductions due to degraded material conditions. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results, significant corrective action, repeated maintenance activities, maintenance rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem equipment, system health reports, and the plant health committee issues list. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. As practical, the inspectors performed walkdowns of the components to evaluate the as-built design and material condition. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report (USAR), Technical Specifications (TS), component/system design basis documents, drawings, and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code and the Institute of Electrical and Electronics Engineers (IEEE) Standards, to evaluate acceptability of the systems' design. The review was to verify that the selected components would function as required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability were consistent with the design bases and were appropriate may include installed configuration, system operation, detailed design, system testing, equipment/environmental qualification, equipment protection, component inputs/outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, system health report, and condition reports (CRs). Walkdowns were conducted for all accessible components to assess material condition and to verify the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The components (17 samples) listed below were reviewed as part of this inspection effort:

C High Pressure Core Spray (HPCS) Pump: The inspectors reviewed vortexing calculations for HPCS pump suction alignment to the suppression pool and condensate storage tank. Hydraulic calculations were reviewed to ensure design requirements for flow and pressure were translated as acceptance criteria for pump in-service testing (IST). The inspectors reviewed calculations related to pump's net positive suction head (NPSH) to ensure the pump was capable of functioning as required. The pump motor's calculations for voltage/frequency, current ratings, protective relaying settings, and cable feeder size were reviewed to ensure the motor was able to drive the associated pump. Design change history and IST results were reviewed to assess potential component degradation and impact on design margins. In addition, the licensee responses and actions to Bulletin 88-04, "Potential Safety-Related Pump Loss," were reviewed to assess implementation of operating experience. A modification to replace the pump breaker was also reviewed.

C Division 3 Diesel Generator (DG) for HPCS: The inspectors reviewed electrical loading calculations, including voltage and frequency, current ratings, and short circuit ratings for all operating modes. Protective relaying calculations were reviewed to assess adequacy of protection during test mode and during emergency operation. Test results were reviewed to ensure the adequacy of current ratings for full loading and emergency loading. Operating and surveillance test procedures were reviewed to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- HPCS Minimum Flow Valve: The inspectors reviewed the motor-operated valve (MOV) calculations, including required thrust, degraded voltage, maximum differential pressure, setpoint, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. Diagnostic and IST results were reviewed to verify acceptance criteria were met and performance degradation would be identified.

- Division 3 Essential Service Water (ESW) Pump: The inspectors reviewed calculations related to pump flow, head, and NPSH requirements to ensure the pump was capable of functioning as required. Design change history and IST results were reviewed to assess potential component degradation and impact on design margins. The inspectors reviewed the control and power design drawings to verify the availability of both control and power required for operability. In addition, the licensee responses and actions to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," were reviewed to assess implementation of operating experience. A modification to change pump shaft materials and other design improvements for this pump was reviewed to ensure the change did not adversely affect the design function of the pump.

- HPCS DG Heat Exchanger: The inspectors reviewed the ESW hydraulic analysis that verified the DG jacket water heat exchanger would receive minimum design cooling water flow, in accordance with the flow value specified in the thermal evaluation of the heat exchanger. The inspectors reviewed thermal performance testing, and trending of data, that was performed to verify heat transfer capability. The inspectors interviewed maintenance personnel to ascertain whether the condition of the heat exchanger and attached piping was meeting the guidance of GL 89-13. The inspectors reviewed periodic surveillance tests to verify coolant temperatures were within acceptable limits and performance degradation would be identified.

- Swayle Manual Valves: The inspectors reviewed the seismic and hydraulic calculations for the ESW function using the Swalye valves, IST program, operator rounds, and test procedures to ensure the manual Swayle valves were capable of functioning as required. Maintenance activities were reviewed to verify the periodic operation of the valve. Operator actions and operating procedures were reviewed as the Swayle valves' operation was necessary with opening of the intake bay sluice gates under certain conditions to allow the ESW system to continue it's safety-related function. A modification to the sluice gates to install an air seal system was reviewed based on this inter-related function.

- Residual Heat Removal (RHR) Cross-tie to ESW Valves: The inspectors reviewed the hydraulic calculation to verify decay heat can be removed by ESW in the emergency lineup when injecting into the reactor vessel. Since the cross-tie valves were not normally used, the inspectors verified that the cross-tie valves and associated connecting pipe were inspected to verify that the system would pass design flow. The inspectors performed a walk through of the emergency operating procedures associated with these valves to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required. Additionally, the inspectors performed an assessment of the radiological conditions to be expected during the performance of the above emergency operating procedures.

C Reactor Core Isolation Cooling (RCIC) Pump: The inspectors reviewed vortexing calculations for the RCIC pump suction alignment to the suppression pool, and condensate storage tank. The inspectors reviewed hydraulic calculations to ensure design requirements for flow and pressure were translated as acceptance criteria in IST pump testing. The inspectors reviewed calculations related to the pump's NPSH requirements and minimum flow requirements to ensure the pump was capable of functioning as required. Testing was reviewed to assess potential component degradation and impact on design margins. In addition, the licensee responses and actions to Bulletin 88-04 was reviewed to assess implementation of operating experience. A modification to raise the setpoint of the pump's suction relief valve as corrective action to inadvertent valve lifting events was reviewed.

C RCIC Room Cooler: The inspectors reviewed the RCIC room cooler thermal sizing and the loss of ventilation calculations to ensure that the heat gains in the room were properly accounted for and that a conservative starting temperature was used. The inspectors reviewed data to ensure that the room temperature was within acceptable limits. A walkdown was performed to verify cleanliness of the air filters, and preventive maintenance work orders were reviewed to ensure filters were changed on an appropriate frequency.

C RCIC Test Return Valve: The inspectors reviewed MOV calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. Diagnostic and IST test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. The inspectors reviewed test results to verify the valve would receive an automatic close signal during a RCIC pump test if the system received an initiation signal. Regulatory Information Summary (RIS) 2001-15, "Performance of DC-Powered Motor-Operated Valve Actuators," was reviewed to ensure it was properly evaluated and implemented as appropriate.

C Automatic Depressurization System (ADS) Valves: The inspectors reviewed calculations used for sizing of the air storage tank, structural support capability of the individual safety relief valve (SRV) accumulators, ASME Code stress analysis, solenoid valve operating current and voltage, and associated circuit protection to ensure the ADS valves were capable of functioning under design conditions. The inspectors reviewed vendor design specifications for the SRV accumulators to verify compliance with Code requirements. The inspectors also reviewed the air leak rate testing procedure, and recently completed leak rate testing performed for the air system connected to the accumulators to verify that the acceptance criteria were appropriate and data was within the defined criteria.

- C Reactor Water Cleanup Suction Inboard Isolation Valve: The inspectors reviewed the MOV calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. Local leak rate, diagnostic, and IST test results were reviewed to verify acceptance criteria were met and performance degradation would be identified.

- C Division 2 Diesel Generator: The inspectors reviewed calculations for diesel loading, fuel oil consumption, and vortexing for the fuel oil tanks. Seismic qualification documents for the E22-S001 electrical cabinets were also reviewed. The inspectors performed a review of system normal operating procedures and surveillance test procedures to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- C Diesel Generator Output Breakers: The inspectors reviewed the electrical drawings that described the circuits used to control the generator outputs by means of the DG output breakers. Corrective actions for DG output breaker problems were reviewed.

- C Division 1 and 2 DG Air Start Check Valves: The inspectors reviewed the air leak rate testing procedure and recently completed leak rate testing performed for the diesel generator air start check valves to verify acceptance criteria were met and performance degradation would be identified.

- C 4.16 kV 1E Bus EH12: The repair and replacement program established as part of the corrective actions for 4.16 kV breakers problems was reviewed.

- C Unit 1 and 2 Safety-Related Batteries: The inspectors reviewed electrical calculations, including voltage at the valve terminals under various accident scenarios, battery float and equalizing voltages, and overall battery capacity. The inspectors performed a walk through of the off-normal operating procedures associated with cross-tying Unit 1 and 2 safety-related batteries to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required.

The following review did not constitute an inspection sample:

- C Unit 1 and 2 Non-Safety Related Battery: These batteries were initially selected based on an initial review of the licensee's PRA, which indicated a high risk ranking. Subsequent questioning by the inspectors determined that the risk ranking was in error (high ranking was mis-identified, should have been identified as the safety-related batteries), such that inspection activities were limited and this component will not be counted as a completed sample. The inspectors did walkdowns of the non-safety related batteries, reviewed maintenance and surveillance schedules and records, and performed a walk through of the off-normal operating procedures associated with cross-tying Unit 1 and 2 non-safety related batteries.

b. Findings

Two findings of very low safety significance associated with Non-Cited Violations (NCVs) were identified.

b.1 ADS and MSIV Air Accumulators Stress Analysis Deficiencies

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the stress analysis performed for the ADS backup air accumulators. Specifically, the licensee failed to account for all the related stresses in the ADS accumulator stress analysis calculation. Inclusion of these additional stresses resulted in a higher stress than allowed by the American Society of Mechanical Engineers (ASME) Code. Additionally, the accumulators' certification of design, as required by the Code, Section III, was incorrect as it was not based on the maximum design pressure the accumulators would be subject to under accident conditions.

Description: The ADS used a number of the reactor SRVs to reduce reactor pressure during a small break loss-of-coolant-accident (LOCA) in the event of a HPCS failure. Each ADS valve has a backup accumulator to store air to provide the motive force to stroke the valve if the normal non-safety related air system was unavailable. There were no relief valves on the air accumulators to relieve internal pressure, which also have check valves to prevent flow out of the accumulators back into the air system. Since this created a closed volume, any increase in ambient temperature would result in a corresponding increase in internal pressure.

The accumulators were designed per the requirements of ASME Code, Section III, Division 1, 1974 Edition, up to and including the Winter, 1975 Addenda. The inspectors reviewed the original stress analysis performed for the tanks (accumulators) GAI Vendor Print 4549-94Q-76-1-0, "Final Report on the ASME Section III Analysis of Class 3 Nuclear Safety Related Shop Fabricated Tanks for Perry Nuclear Power Plant." This analysis used an accumulator pressure of 150 pounds per square inch (psi) for internal pressure. The inspectors noted that the certification of design, and the vessel nameplate information for the accumulators listed a maximum ambient temperature of 330 degrees Fahrenheit (EF), and a vessel design pressure of 180 pounds per square inch gauge (psig). The internal pressure used in the stress analysis, however, did not take into account the affect of heating the enclosed air volume due to the maximum ambient drywell temperature.

The licensee provided Calculation P52-003, "Overpressurization of Isolated Piping and Components," which evaluated the affect of the pressure increase on the ADS accumulators during a thermal overpressure event caused by a rapid increase in drywell temperature up to 330 EF. This calculation was completed prior to initial plant startup when this issue was initially identified, however, the actions taken did not adequately resolve the concern. The calculation determined that for an ambient drywell temperature of 330 EF the accumulators should have been designed to withstand an internal pressure of 308.4 psig.

The certification of design for the accumulators listed an internal design pressure of 180 psi at a temperature of 330 EF, with a hydrostatic test pressure of 270 psi. Because the maximum pressure of 308.4 psig was not specified in the ASME certification of design and the vessel nameplates as a design condition, the inspectors determined that the air accumulators were not in compliance with ASME Code, Section III. Paragraph NA-3252 of the ASME Code, “Contents of Design Specification”, item (b) required, in part that the design requirements shall be included in design specification. The licensee initiated CR 06-01044 to address this issue.

The tank specification for the accumulators, SP-504-4549-00, “Conformed Specification for the Tanks,” had design requirements in paragraph 2:06.3.1.a.(2) for acceleration loading requirements and combined loading design limits for safe shutdown building acceleration. The paragraph stated that the tanks located inside the reactor building shall withstand the maximum acceleration created by safe shutdown earthquake combined with SRV discharge and small break accident loads. The inspectors noted that the ASME Code, Section III, Division I, Class 3 Stress Report, supplied by the ADS accumulator vendor as part of original construction, performed an analysis that evaluated loading conditions that included both the design pressure of the tanks and the maximum allowable nozzle loads consisting of axial loads and bending moments caused by seismic plus operating loads, as required by the tank specification. The inspectors noted that the stress analysis contained in Calculation P52-003 was non-conservative as it only considered the hoop stresses in the accumulator caused by the increase in internal pressure, but did not include the maximum allowable nozzle loads when calculating the maximum stresses the accumulators would be subject to under design conditions.

The licensee subsequently performed an operability determination that evaluated the effect of not including the nozzle loads in the stress analysis. The operability determination concluded that the accumulators would maintain pressure integrity using the Level D stress criteria for Class 3 vessels from the 2004 Edition of ASME Code. Level D rules were intended to maintain pressure boundary integrity, but will permit structural deformation. The licensee determined that Calculation P52-003 needed to be revised to include all required stresses and issued CR 06-00974 to track resolution. The inspectors and the Office of Nuclear Reactor Regulation reviewed the licensee’s operability determination and concluded that there would be no affect on operability of the ADS system when accounting for the higher stresses in the accumulators. The use of the Level D stress criteria was acceptable for operability, however, the accumulators were considered nonconforming that would require further action by the licensee to return them to within Code compliance.

Since Calculation P52-003 also evaluated the pressure integrity of the main steam isolation valve (MSIV) accumulators, the inspectors had similar concerns with the stress analysis for these accumulators. As a result, the licensee issued CR 06-00997 and an operability determination that concluded the MSIV accumulators were structurally qualified to ASME Code Level D faulted limits. The inspectors reviewed the operability determination, and agreed with the licensee that the accumulators were operable, but nonconforming. The operability determination discussed the piping systems attached to the accumulators, and determined that they would maintain structural integrity as well.

<u>Analysis</u>: The inspectors determined that the failure to account for the stresses related to the dynamic structural loading in the accumulators was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in

accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Dispositioning Screening,” because it was associated with the attribute of design control, which affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the ADS valves to respond to initiating events to prevent undesirable consequences. Specifically, the failure to account for all the stresses related to the dynamic structural loading in the accumulators resulted in not knowing that the material yield stresses exceeded the ASME Code allowable stress limits.

The inspectors evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, “Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety,” did not represent an actual loss of a system’s safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The basis for this conclusion was that despite the loss of design margin in the ADS accumulator structural capability to maintain pressure integrity, the ADS system would have performed its safety function as concluded in the operability determination.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” required, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, as of February 28, 2006, the licensee’s design control measures failed to provide for verifying or checking the adequacy of design by validating that the calculated stress value for ADS air accumulators would be higher than that assumed by the structural analysis. Specifically, Calculation P52-003 did not evaluate the increased pressure stress in the accumulators in combination with stresses due to seismic plus operating loads, and only considered the hoop stresses in the accumulator caused by the increase in internal pressure resulting from an ambient temperature increase. This resulted in the accumulator’s structural analysis being non-conservative, and not in accordance with the requirements of ASME Code, Section III, Paragraph NA-3252. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2006009-01(DRS)). The licensee entered the finding into their corrective action program as CR 06-00974 to revise the affected calculation, and CR 06-01044 to evaluate ASME Code requirements for the accumulators. The licensee performed an operability determination and concluded that the accumulators, while currently nonconforming, will remain operable.

b.2 Non-conservative Safety-Related Air Storage Tank Sizing Calculation

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving the sizing of the P57 air storage tank. Specifically, the inspectors identified that the licensee failed to correctly specify the minimum differential air pressure required to actuate the ADS valves when manually operated.

Description: The inspectors reviewed Calculation P57-13, "Required Air Volume for MSIV and ADS Accumulators," whose purpose was to determine the long-term air volume in the P57 air storage tank required for the ADS and MSIV air actuators. The accumulator sizing calculation was based on 90 psig, the minimum air pressure to stroke the valve. However, General Electric (GE) Design Specification Data Sheet stated in paragraph 3.1.18.2.2, that for the ADS function, the differential air pressure required to stroke the valve at 0 psig inlet pressure was 95 psid to account for drywell backpressure. Drywell pressure would be elevated above ambient pressure during accident conditions and a station blackout event where ADS was credited with manual vessel depressurization. Since drywell pressure during an accident could be as high as 30 psi, 125 psi would be required to operate the valves as they must overcome the exhaust pressure, which would be exposed to the elevated drywell pressure. By not accounting for the drywell backpressure and the use of a lower minimum air pressure (90 versus 95) when evaluating the accumulator capacity, the inspectors determined that the calculation was non-conservative when calculating the long-term air volume requirements in the P57 air storage tank. Although the licensee indicated there may be a supportable basis for the use of the lower minimum air pressure, the inspectors concluded the calculation must account for the backpressure in containment when determining the minimum actuation pressure of the valve.

Additionally, calculation P57-13 determined the leakage rate acceptance criteria for procedure PTI-P57-P0001, "Loss of Air Test for Safety Related Instrument Air System." Since the calculation assumed that only 90 psi were required to stroke an ADS valve, the acceptance criteria would be non-conservative. Since a higher pressure was required to operate the valves, the licensee reviewed the last several completed tests to ensure that sufficient air pressure remained in the tanks to stroke the valves as required during a design basis event. The review determined that the indicated leakage rates were well below the leakage rate acceptance criteria. Based on the normal 150 psig of air maintained in the P57 storage tank and the low indicated leakage rates, the inspectors concluded that the ADS air system had sufficient capacity for operability.

Analysis: The inspectors determined that failure to correctly specify the minimum differential air pressure required to actuate the ADS valves when manually operated was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of design control, which affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the ADS valves to respond to initiating events to prevent undesirable consequences. Specifically, the failure to properly account for minimum air pressure in the calculation that was used as the basis of ADS air system testing could result in over-predicting the air storage tanks' performance (i.e., creating design margin capability that would not exist), which could potentially render the ADS valves incapable of performing its required safety function.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, did not represent an actual loss of a system's safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The basis for this conclusion was that despite the loss of design margin in the air storage tanks' capacity, the ADS system would have performed its safety function as the P57 air storage tank was maintained at 150 psig and recent testing indicated low leakage from the air system.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of February 17, 2006, the licensee failed to assure that the ADS minimum air pressure operability limits were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the minimum air pressure requirements as specified in the GE Design Specification Data Sheet for SRVs were not incorporated into Calculation P57-13, which in turn resulted in non-conservative acceptance criteria for PTI-P57-P0001 air storage tank pressure drop test. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2006009-02(DRS)). The licensee verified the air storage tank contained a sufficient volume and air pressure to support ADS operation and initiated CR 06-00817 to revise the affected documents.

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed four operating experience issues (4 samples) to ensure these issues, either NRC generic concerns or identified at other facilities, had been adequately addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection effort:

C Gas voids due to improper venting or gas intrusion in emergency core cooling systems and other safety related piping systems such as RCIC and RHR. Responses to related experience at Beaver Valley and Limerick were also reviewed for applicability;

C RIS 2001-05, "Performance of DC-Powered Motor-Operated Valve Actuators";

C Bulletin 88-04, "Potential Safety-Related Pump Loss"; and

C GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

b. Findings

One unresolved item was identified associated with NRC Bulletin 88-04 concerning minimum flow for safety-related pumps.

b.1 Inadequate Response for Minimum Pump Flow Settings

Introduction: The inspectors identified an unresolved item (URI) concerning the licensee's response to Bulletin 88-04, "Potential Safety-Related Pump Loss," regarding establishing minimum flow requirements for the safety-related pumps. The licensee recognized that the conditions reported in the bulletin were present in safety-related pumps, but did not determine appropriate minimum pump flow values to minimize and manage, or to eliminate, the potential for pump damage. At the end of the inspection, the licensee was evaluating current minimum flow values with the pump manufacturer. The inspectors needed this information to complete the assessment of this issue.

Description: Bulletin 88-04, in part, identified a concern regarding the adequacy of minimum flow capacities for safety-related centrifugal pumps. The Bulletin required licensees to evaluate the capability of safety-related pumps to run long-term at minimum recirculation flow rates. The Bulletin stated that many licensees had accounted for thermal considerations in setting the minimum recirculation flow rates, but had failed to consider flow instability effects. The latter consideration could necessitate a considerable increase in minimum flow settings, especially for pump operation for extended periods of time. This potential increase occurred because centrifugal pumps demonstrated a flow condition described as hydraulic instability or impeller recirculation at some flow point below approximately 50 percent of the best efficiency point on the characteristic pump curve. These unsteady flow phenomena become progressively more pronounced if flow was further decreased, and could result in pump damage when operated for extended periods of time. The inspectors reviewed the licensee's responses to Bulletin 88-04, which were described in four letters to the NRC; one in 1988, one in 1989, and two in 1990. The latter three responses were specific to the RCIC pump, where no concerns were identified by the inspectors.

The inspectors identified two concerns with the licensee's 1988 response to the Bulletin:

C The licensee did not properly verify the minimum flow settings with the pump manufacturer in accordance with what was stated in their response to the Bulletin. The licensee had concluded that the original, manufacturer-supplied minimum recirculation flows contained in the pump purchase specifications were adequate to meet the issues discussed in Bulletin 88-04.

  The licensee stated on page 2 of the response, "In all cases but one, the minimum flow capacities exceeded the values specified by the manufacturer. The exception is the minimum flow capacity provided for the RCIC pump." The inspectors verified that the licensee had adequately addressed minimum flow for the RCIC pump. The inspectors requested documentation to establish the technical bases for the current minimum flow settings for the RHR, low pressure core spray (LPCS), and HPCS pumps, particularly how they accounted for flow instability. The licensee was unable to provide any documentation that addressed this issue during the inspection. The adequacy of RHR, LPCS, and HPCS pumps' minimum flows was provided to the licensee by GE, as part of the original plant design, and since the licensee had been assigned these settings prior to Bulletin 88-04, it was likely that these settings did not account for flow instability considerations. With the exception of the RCIC pump, the inspectors concluded that the existing minimum flow settings for the safety-related pumps accounted only for thermal effects.

The inspectors questioned whether the current minimum flow settings were reviewed and approved by the pumps' manufacturer (Byron-Jackson), as specified in the licensee's response to the Bulletin. The licensee had not contacted the pump manufacturer and relied upon information provided by GE to conclude that no changes were needed for pumps in these three systems. The licensee contacted the pumps' manufacturer (now Flowserve) to perform a new analysis of the RHR, LPCS, and HPCS pumps' minimum flow settings in response to Bulletin 88-04. This issue was entered into the licensee's corrective action program as CR 06-00813.

- In the licensee's response to the Bulletin, it was stated "SOI/SVI procedure revisions will be provided for those systems which do not presently contain adequate caution. These cautions will limit pump minimum flow operation to a maximum of 30 minutes and assure that pump discharge is transferred to the full flow test line whenever possible." The licensee further stated that the review and approval of necessary procedure changes will be completed by October 5, 1988. When the inspectors reviewed the safety-related pump procedures, there was no evidence that any precautions related to minimum flow were ever implemented in the appropriate RHR, LPCS, or HPCS pump procedures. There were, however, precautions related to minimum flow incorporated into the RCIC pump procedure, which were considered appropriate by the inspectors. This issue was entered into the licensee's corrective action program as CR 06-00703 to determine if the 30 minute limitation still needed to be incorporated into the respective pump procedures.

In response to these concerns, the inspectors prompted the licensee's engineering department to issue on March 2, 2006, a standing order to control room operators to be aware of the concerns for operation of safety-related pumps for extended periods of time while on minimum flow. Since operating a pump on minimum flow was considered a long-term degradation mechanism, issuance of the standing order provided confidence that the pumps would not be damaged prior to the pump manufacturer completing their analysis.

As a result of the response to Bulletin 88-04, the RHR, LPCS, and HPCS pumps were operated since original plant start-up with an increased potential for unusual wear and aging. Based on the licensee's discussion with the pump manufacturer, the NRC concluded that additional review and evaluation were required to assess whether or not the licensee has established adequate minimum flow requirements for the RHR, LPCS, and HPCS pumps since they may operate at minimum flow conditions for extended periods of time under accident conditions. Therefore, this issue is considered an unresolved item (URI 05000440/2006009-03(DRS)) pending completion of an analysis to assess the pumps minimum flow requirements and subsequent NRC review.

.5 Modifications

a. Inspection Scope

The inspectors reviewed four permanent plant modifications related to the selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components have not been degraded through modifications. The Engineering Change Packages (ECPs) listed below were reviewed as part of this inspection effort:

- ECP 00-5013, “Safety-Related Upgrade of the ESW Sluice Gate Sealing System”;
- ECP 02-0283, “RCIC High Pressure Alarm Setpoint 1E51N0852”;
- ECP 03-0013, “High Pressure Core Spray Pump Breaker Modification”; and
- ECP 04-0263, “Emergency Service Water (ESW) “C” Pump Upgrade Modification (1P45C0002).”

b. Findings

No findings of significance were identified.

.6 Risk Significant Operator Actions

a. Inspection Scope

The inspectors performed a margin assessment and detailed review of five risk significant, time critical operator actions (5 samples). These actions were selected from the licensee’s PRA rankings of human action importance based on risk achievement worth and Birnbaum values. Where possible, margins were determined by the review of the assumed design basis and USAR response times and performance times documented by job performance measures results. For the selected operator actions, the inspectors performed a walk through of associated procedures with a plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required. The following operator actions were reviewed:

- Actions to terminate and prevent injection into the reactor pressure vessel (RPV) during an anticipated transient without scram (ATWS);
- Actions to prepare for injection into the RPV during an ATWS;
- Actions to lineup and inject into the RPV with alternate injection systems during an ATWS;
- Actions to initiate standby liquid control during an ATWS; and
- Actions to initiate suppression pool cooling.

b. Findings

One finding of very low safety significance associated with a NCV was identified.

b.1 Inadequate Procedures for Controlling Flow into Reactor Vessel

Introduction: The inspectors identified a NCV of Technical Specification 5.4.1, having very low safety significance (Green), for failing to maintain adequate procedures and/or instructions for controlling injection into the reactor pressure vessel under conditions where the reactor would not shutdown without relying on the injection of boron.

Description: Plant Emergency Instruction (PEI) flowcharts (RPV Control – ATWS and RPV Flooding) align low pressure coolant injection (LPCI) to inject through the RHR shutdown cooling return path (outside the shroud injection) to control either reactor water level or reactor pressure under conditions when the reactor was not shutdown without relying on the injection of boron. Procedures PEI-SPI 6.1(6.2), "LPCI A(B) Outside the Shroud Injection," provided the necessary instructions for controlling this injection flow-path. These procedures were revised in January of 2003 to address corrective actions specified by CR 01-02848. The corrective actions were meant to address the incorrect assumption in Calculation EPGSAG-WS10, "RPV Variables," that the RHR pump minimum flow valves would be closed over the range of reactor pressures used in the calculation. To address this issue, the procedures established a flow band, with a minimum value of 2100 gallons per minute (gpm) to ensure the pumps' minimum flow requirement was met with the minimum flow valves closed, and a maximum value (dependent on whether the flow path was through the RHR heat exchanger or not) of either 7800 or 8500 gpm. There were no provisions in the procedure for reducing flow below 2100 gpm.

The establishment of a minimum injection flow rate of 2100 gpm forced the operator into an "on/off" level control strategy that would lead the operator to terminate flow when flow rates less than 2100 gpm were needed (e.g., RPV level or pressure was at the top of the desired control band) and reestablish flow when RPV level approached -25 inches or RPV pressure approached the minimum steam cooling pressure. This stopping and restarting of injection flow could cause rapid changes in the temperature of the coolant entering the core region, thus increasing the potential for large power excursions that could damage the fuel. The BWR Owner's Group Emergency Procedure and Severe Accident Guidelines as well as the Perry Specific Technical Guidelines specify that injection, into a reactor that is not shutdown without relying on boron injection, must be controlled so as to minimize boron dilution and changes in core inlet subcooling, thus preventing the undesirable consequences of large power excursions. By allowing a continuous injection at flow rates that would maintain RPV water level in the desired level band, colder water entering the RPV would mix with the warmer water in the region outside the shroud. This slow, relatively constant injection rate would minimize the temperature changes of the water entering the core region thus preventing large power excursions.

Analysis: The inspectors determined that the establishment of a minimum injection flow rate, contrary to both the Boiling Water Reactor (BWR) Owner's Group Emergency Procedure and Severe Accident Guidelines as well as the Perry Specific Technical Guidelines, was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of procedure quality, which affected the mitigating systems cornerstone objective of ensuring the ability of the licensee to use the LPCI sub-systems to prevent undesirable consequences associated with an ATWS. Specifically, the procedure's "on/off" level control strategy may not prevent large power excursions caused by rapid changes in the temperature of the coolant entering the core region.

The inspectors evaluated the finding using IMC 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, did not represent an actual loss of a system's safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation.

Enforcement: Technical Specification 5.4.1, required, in part, that written procedures/instructions be established, implemented, and maintained covering the emergency operating procedures required to implement the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0737, Supplement 1. The BWR Owner's Group Emergency Procedure and Severe Accident Guidelines as well as the Perry Specific Technical Guidelines specify that injection, into a reactor that is not shutdown without relying on boron injection, must be controlled so as to minimize boron dilution and changes in core inlet subcooling, thus preventing the undesirable consequences of large power excursions.

Contrary to the above, in January of 2003, the licensee revised the procedures and instructions for controlling injection into the reactor pressure vessel under conditions where the reactor is not shutdown without relying on the injection of boron by establishing a level control strategy that did not minimize changes in core inlet subcooling, thus increasing the potential for large power excursions that could damage the fuel. Because this violation was determined to be of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000440/2006009-04). The licensee entered the finding into their corrective action program as CR 06-00872 to assess the affected documents.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. W. Pearce and other members of licensee management at the conclusion of the inspection on March 3, 2006, and with Mr. J. Shaw on March 23, 2006. Proprietary information was reviewed during the inspection and will be handled in accordance with NRC policy.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Mitigating Systems

Criterion III, “Design Control,” of 10 CFR Part 50, Appendix B requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The licensee did not adequately translate design basis information into a HPCS system calculation. Specifically, Calculation E22-029, “SVI-E22-T2001- HPCS Pump Performance Test Acceptance Criteria,” had an error in its methodology as the latest revision of the calculation did not properly account for the Division III, diesel generator governor ± 2 percent droop adjustment (frequency). The licensee failed to consider how the frequency adjustment could affect the design and licensing basis of the HPCS pump’s capability to perform under reduced DG frequency. This was identified in the licensee’s corrective action program as CR 06-00184 and CR 06-00296. The inspectors reviewed the licensee’s operability determination and verified that the HPCS pump remained capable of performing its design function. The inspectors determined that the finding was more than minor because if the licensee had not recognized the error, the pump could have degraded below its’ design requirement. The inspectors determined that the finding was of very low safety significance because it did not represent an actual loss of system safety function since the HPCS pump met its design requirement.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
W. Pearce, Site Vice President
P. Chatterjee, Electrical Engineer
H. Conrad, Nuclear Engineer
D. Gartner, Lead Instrumentation Engineer
J. Hagan, Senior Vice President Operations
T. Hilston, Design Engineering Supervisor
K. Howard, Design Engineering Manager
D. Jondle, Probabilistic Risk Assessment
J. Lausberg, Regulatory Compliance Manager
D. Pace, Senior Vice President Engineering
J. Powers, Fleet Engineering Director
J. Rinckel, Vice president Oversight
K. Russell, Regulatory Compliance
J. Shaw, Engineering Director
S. Seman, NSSS Supervisor
R. Siembor, Mechanical Engineer
J. Tufts, Operations Superintendent
F. Von Ahn, Plant Manager
J. Zarea, Lead Electrical Engineer

Nuclear Regulatory Commission
C. Pederson, Director, Division of Reactor Safety
A. M. Stone, Chief, Engineering Branch 2, DRS
R. Powell, Senior Resident Inspector
M. Franke, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000440/2006009-01	NCV	ADS and MSIV Air Accumulators Stress Analysis Deficiencies (Section 1R21.3.b.1)
05000440/2006009-02	NCV	Non-conservative Safety-Related Air Storage Tank Sizing Calculation (Section 1R21.3.b.2)
05000440/2006009-03	URI	Inadequate Response for Minimum Pump Flow Settings (Section 1R21.4.b.1)
05000440/2006009-04	NCV	Inadequate Procedures for Controlling Flow into Reactor Vessel (Section 1R21.6.b.1)

Closed

05000440/2006009-01	NCV	ADS and MSIV Air Accumulators Stress Analysis Deficiencies (Section 1R21.3.b.1)
05000440/2006009-02	NCV	Non-conservative Safety-Related Air Storage Tank Sizing Calculation (Section 1R21.3.b.2)
05000440/2006009-04	NCV	Inadequate Procedures for Controlling Flow into Reactor Vessel (Section 1R21.6.b.1)

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

1R21 Component Design Bases Inspection

Calculations

Number	**Title**	**Revision**
860602B	Low Pressure Air Storage Tank Sizing	Revision 1
CL-MOV-1E22-1	1E22-F001, F004, F010, F011, F015 and F023 Max DP	Revision 2
CL-MOV-1E51-3	1E51-F010, F013, F019, F022, F031, F059, F068, F077, and F078 Max DP	Revision 2
CL-MOV-1G33-1	MOV 1G33F0001, 1G33F0004 Max DP	Revision 3
E12-088	Residual Heat Removal System Hydraulic Calculation	Revision 1
E22-C03	1E22N0656 Leave-as-is Zone and Reset Calculation	Revision 0, DCC 1
E22-001	HPCS System NPSH Calculation	Revision 0, DCC 1-5
E22-11	E22 Pump Suction Switchover	Revision 0, DCC-01
E22-025	Minimum Flow Required to Create Turbulent Flow on Both Tube and Shell Side of HPCS JW Heat Exchanger	
E22-029	SVI-E22-T2001–HPCS Pump Performance Test Acceptance Criteria	Revision 6
E22-037	Design Basis Heat Load & Required ESW Flow for HPCS DGJW HX	Revision 2
E22-039	Evaluate the Test Data Recorded During the Performance of PTI-E22-P0007	Revision 1
E22-041	Div 3 DG JW HX Thermal Performance Test Results August 27, 2003	Revision 0
E22-042	Div. 3 EDG Jacket Water Heat Exc. Performance Test Evaluation August 25, 2004	Revision 0

Calculations

Number	Title	Revision
E51-C03	RCIC-CST Low Level Transfer Trip 1E51-N635A(E)	Revision 6
E51-11	NPSHa For RCIC Pump When Pumping from S.P.	Revision 1, DCC-01
ECA-068	PSA-RCIC Room Heatup During Station Blackout	Revision 1
EPG-21	ECCS/RCIC NPSH Requirements vs. Temperatures of Suppression Pool	Revision 0
EPG04P45-1	RHR Loop B Containment Flood	Revision 3
EPGSAG-WS10	RPV Variables	Revision 4
EPGSAG-WS13	Vortex Limits	Revision 0
MOVC-0029	DC MOV Torque Capability & Stroke Time Calculation using Limitorque Method and also by using BWR Owners Group DC Motor Performance Methodology.	Revision 5, Add. 2
MOVC-0043	Required Thrust Calculation for Gate Motor Operated Valves (MOVs)	Revision 4
MOVC-0044	Globe Valve Required Thrust Calculation	Revision 4
MOVC-0068	EPRI PPM Evaluation of Borg Warner 6" ANSI Class 1500 Flex Wedge Gate Valves - 1G33F0001, 1G33F0004, and 1E51F0013	Revision 0, DCC 1
MOVC-0073	AC MOV Actuator Degraded Voltage Torque/Thrust Capability using Commonwealth Edison (Com Ed) Method	Revision 7
M39-014	HPCS Pump Room Cooler Air Flow Rate and Performance Evaluations at Design Basis Conditions	Revision 1
M39-015	HPCS Pump Room Cooler Performance Test	Revision 0
P11-012	P11 Level Setpoints in CST for E22 and E51 Instruments	Revision 1
P45-056	ESW Pump Performance Test Acceptance Criteria	Revision 0, PIN 1-2, DCC 1-5
P45-057	ESW System Thermal Hydraulic Model	Revision 2, PIN 1-3
P45-059	ESW Flow to Division 3 Components	Revision 1, PIN 1-6

Calculations

Number	Title	Revision
P45-075	Minimum Branch Flow Rates for P45 Surveillance Acceptance Criteria	Revision 0, PIN 1-2
P45-081	Eval. Of NPSH and Submergence Requirements for the ESW System Pumps	Revision 0
P52-003	Overpressurization of Isolated Piping and Components	Revision 1
P57-R13	Required Air Volume for ADS and MSIV Accumulators	Revision 2, PIN 1-2
PRDC-0012	Load Evaluation, Battery Sizing and End of Cycle Battery Voltage Determination for Division 1 & 2 Batteries During a Station Blackout (SBO)	Revision 1
PRMV-0014	(686-85-18) Division 3 HPCS Diesel Generator (1E22S001)/EH1301 Protective Relaying Setpoint,	Revision 3
PRMV-0016	High Pressure Core Spray Motor 1E22C001 Protective Relays Setpoints	Revision 3
PSA-002	Probabilistic Evaluation for Loss of Normal ESW Intake During Periods with Warm Lake Water Temperatures	Revision 1
PSTG-0014	Electrical Load Determination of Division 1, 2, & 3 Diesel Generators	Revision 1
PSTG-0017	Division 1, 2, & 3 Emergency Diesel Generators 7 Day Average kW Loading for Fuel Oil Consumption	Revision 7
R44-007	DG Air Start Check Valves	Revision 0
R44-008	DG Air Start Check Valves	Revision 0
R45-014	Assessment of Fuel Oil Transfer from Fuel Oil Storage Tank to Day Tank	Revision 0
R45-009	Determine Fuel Oil Volume for Standby and HPCS Diesel Generators for 7 day Supply	Revision 6
SQ-0014	Modification 1G33F0001/F0004; Justification F039/F040	Revision 3
SQ-0048	Valves 1E22-F0012 and F0059 Thrust Limits	Revision 3
SQ-0067	Seismic Qualification of Valves: 1E51-F0022, -F0045; 1G33-F0031, -F0102, -F0104, and -F0107	Revision 0

Condition/Issue Reports Generated Due to the Inspection

Number	Title	Date
06-00184	Calculation E22-029, R-6 Has an Apparent Error in its Methodology	1/13/06
06-00237	Frequency Variation In EDG Loading Calc. PSTG-0014, R/7	1/17/06
06-00296	Issue with Full Qualification of HPCS	1/19/06
06-00428	Potential Missed Surveillance (1P45)	1/27/06
06-00476	Editorial Changes to PRA Model Basic Events	1/31/06
06-00482	125VDC Short Circuit Analysis for Paralleling of Unit 1/2 Batteries	2/1/06
06-00493	Question Regarding HPCS Jacket Water Flow	2/1/06
06-00494	Tour Resulted In Questions	2/1/06
06-00519	Functional Location Reference Information for Calculation in Curator/filenet Incomplete	2/1/06
06-00527	Excessive Maintenance Notification Tags on the Div 1/2 DGs	2/2/06
06-00632	Review of ESW “C” Motor Oil Analysis	2/9/06
06-00696	Reactor Core Isolation Cooling System Pump Room Temperature	2/10/06
06-00701	FO Transfer Pump Seismic Qualification Did Not Consider Motor Starting Torque	2/13/06
06-00703	Commitment in IEB 88-04 Response Not Implemented	2/13/06
06-00704	Procedure Step Potentially Conflicts with Calculation Assumption	2/13/06
06-00722	PEI-SPI’s Appear to Be Open-ended	2/14/06
06-00734	EDG Load Margin Relative to Cable & Transformer Load Losses	2/14/06
06-00745	Incorrect Date Entered on Work Order Data Sheet	2/15/06
06-00746	Operator Actions From Memory Tracking Mechanism	2/15/06
06-00751	Lube Oil Results Not Evaluated In Accordance with TAI-2000-3	2/14/06
06-00754	HPCS Pump Curves in TAF and Vendor Manual Do Not Match	2/14/06
06-00761	Material Condition of ESWPH	2/15/06
06-00769	24 hour SBO Coping Analysis	2/16/06
06-00776	Lube Oil Sample Analysis Result Parameter Out-Of-Spec	2/16/06
06-00779	HPCS Division 3 EDG Operating Procedure Incorrectly Displayed the Operating Mode of the Ground Fault Protection	2/16/06

Condition/Issue Reports Generated Due to the Inspection

Number	Title	Date
06-00813	Question on Basis of Min Flow Adequacy	2/16/06
06-00815	Maximum Drywell Temperature Affects ADS Accumulator Pressure	2/16/06
06-00817	Affects Drywell Backpressure Not Addressed in Calc P57-13	2/19/06
06-00827	Gauge Reading Low	2/18/06
06-00866	Error found In Calculation P52-003 Revision 1	2/18/06
06-00872	PEI-SPI Procedural Compliance	2/22/06
06-00873	Procedural Enhancement Potential	2/22/06
06-00876	Question on Use of Pump Runout Flow in Calc E22-011	2/24/06
06-00898	HPCS Pump Motor Protective Calculations Had Incorrectly Portrayed Protection and Coordination	2/23/06
06-00970	Calculation MOV-1E22-1 Reference List Contains Errors	2/28/06
06-00973	Regarding Potential Noncompliance with ASME Code	2/28/06
06-00974	Bending stress Not Considered in Calc P52-003 R/1	2/28/06
06-00983	ADS Solenoid Valve Overcurrent Protection	3/01/06
06-00984	Calculation PRMV-0016, R/3 for HPCS Motor Protection	3/1/06
06-00985	Diesel Generator Neutral Grounding Resistors	3/1/06
06-00993	Drawing 302-0351 / 302-0355 Discrepancies	2/28/06
06-00997	Bending Stress Not Considered in Calc P52-003 R/1 for Tanks	3/1/06
06-01015	DG Neutral Grounding Resistors, Capacitive Current Evaluation	3/2/06
06-01017	Calculation PRMV-0001 (DG Ground Fault Protection)	3/2/06
06-01044	Code Questions For ADS Accumulators	3/3/04
06-01054	Testing Frequency of Battery D-1-B	3/3/04

Condition/Issue Reports Reviewed During the Inspection

Number	Title	Date
99-02677	RCIC Minflow Orifice May Be Too Small	11/8/99
00-03380	Sluice Gate Opening Without Prealignment of ESW to the Swayle	11/1/00
01-02848	LPCI Injection Outside the Shroud May Not Have Adequate Flow	7/24/01

Condition/Issue Reports Reviewed During the Inspection

Number	Title	Date
02-03972	HPCS Pump Failed to Start	10/23/02
02-04355	Div 2 DG Experienced Load Instability During the First Main. Run.	11/14/02
02-04772	LOOP and Generator Trip Due to Underfrequency	8/14/03
03-00547	Div 2 DG Wire to Brushes Losing It's Insulator	2/3/03
03-04373	Determine if the Cause of Diesel 3 DG Inoperability Existed in Operable Div 1 and Div 3 DGs	
03-04764	RHR-A/LPCS Water-Leg Pump Not Supplying Adequate Pressure	8/14/03
03-05065	ESW Pump A Failed	9/1/03
03-05580	Re-Review of SOER 97-1 Due to Air-Bound Waterleg Pump	10/3/03
04-03680	RCIC Test Return to CST Valve Did Not Pass PMT	7/15/04
04-04917	OE8987 Review Indicates Possible Air Void in Abandoned RHR to RCIC Suction Pipe	9/22/04
04-06462	Review of CR-03-04764 TS 3.5.1.1 ECCS Venting Concern	12/9/04
05-00230	Some ABB Breaker 10 Year Overhaul Had Not Been Performed	1/11/05
05-01899	Chemistry Program Improvement for MIC Monitoring and Control	3/7/05
06-00236	Calc PRDC-0012 Requires Revision to Clarify SRV Coil Resistance	1/16/06
06-00052	125vdc Control Circuit Coordination Calculation (PRDC-0004)	2/10/06

Drawings

Number	Title	Revision
022-0027	Environmental Conditions for Diesel Generator Areas	Revision JJ
206-0010	Main One Line Diagram 13.8kV and 4.16kV	Revision Z
206-0017	One Line Diagram Class 1E 4.16kV Bus EH11 & EH12	Revision EE
206-0020	Main One Line Diagram 480V	Revision DD
206-0025	One Line Diagram Class 1E 480V Bus EF1C	Revision XXX
206-0027	One Line Diagram Class 1E 480V Bus EF1D	Revision VVV
206-0029	One Line Diagram Class 1E 480V Bus EF1E	Revision KK
256-0050	Electrical One Line Diagram Class 1E DC system Division 3	Revision N

Drawings

Number	Title	Revision
256-0051	Electrical One Line Diagram Class 1E DC System	Revision CC
256-0052	Electrical One Line Diagram Non–class 1E DC System Bus D2A and D2B	Revision Y
302-0271	Safety Related Instrument Air	Revision M
302-0348	Standby Diesel Engine Mounted Piping 1R43-C001B Div 2	Revision F
302-0349	Standby Diesel - Engine Control Panel 1H51-P054B Division 2	Revision H
302-0351	Diesel Generator Starting Air	Revision AA
302-0357	Div 1 & Div 2 Diesel Air Dryer Diagrams 1R44-D001A & B and 1R44-D002A & B	Revision G
302-0358	Div 3 Diesel Starting Air / Air Dryer Diagram 1E22-S001	Revision E
302-0360	Division 3 Diesel Jacket Water Cooling System	Revision D
302-0631	Reactor Core Isolation Cooling	Revision BB
302-0632	Reactor Core Isolation Cooling	Revision JJ
302-0701	High Pressure Core Spray	Revision EE
302-0791	Emergency Service Water System	Revision RR
B-208-066, Sheets 100-119	High Pressure Core Spray Power Supply System	Various
B-208-206, Sheet 53	Metal Clad Switchgear 4.16 kV Stand By Diesel Bkr. EH1102 Protective Relaying	Revision J
B-208-216, Sheets 1-47	Standby Diesel Generator	Various
D-206-018	One Line Diagram Class 1E 4.16kV Bus EH13	Revision Z
D-206-051	One Line Diagram Class 1E DC System	Revision ZZ
D-206-051	Metal Clad Switchgear 4.16 kV Stand By Diesel Bkr.	Revision Z
D-206-050	One Line Diagram Class 1E DC System, Div. 3	Revision Y
	SRV Accumulator Tank Assy. Details-Vendor: Bishopric	Revision 7

Engineering Changes/Modifications

Number	Title	Date
DCC-01	Div 1, 2, 3 D/G Voltage Controlled Overcurrent and Load Test Overload Protection	8/17/93
ECP 00-5013	Safety Related Upgrade of the ESW Sluice Gate Sealing System	10/1/01
ECP 02-0224	Setpoint Change to Adjust Low Flow Alarms for ESW Heat Exchangers	1/17/05
ECP 02-0283	RCIC High Pressure Alarm Setpoint 1E51N0852	1/17/05
ECP 03-0013	High Pressure Core Spray Pump Breaker Modification	11/19/03
ECP 04-0263	Emergency Service Water (ESW) “C” Pump Upgrade Modification	4/7/05

Miscellaneous Documents

Number	Title	Revision/Date
741-S-1414	Pump Curve 14, High Pressure Core Spray, MPL 1E22C0001, Byron Jackson Curve,	4/19/78
BWROG-8836/WAZ2	BWR Owners Group Response to GL 88-04, “Safety-Related Pump Loss”	6/7/88
1E22-B5002	Heat Exchanger Trend Data	
F-C4350-3	Test of Electrical Cables, The Franklin Institute Research Laboratories, Final Report	7/1/76
G1-22	GE Task Report: Containment System Response	Revision 0
G1-45	GE Task Report: Station Blackout	Revision 0
G289	File for Ideal Electric Generator Manual SM100	1/30/78
GAI Vendor Print 4549-94Q-76-1-0	Final Report on the ASME Section III Analysis of Class 3 Nuclear Safety Related Shop Fabricated Tanks for Perry Nuclear Power Plant”, SDRC Report number 7584	12/16/77
MPL E22-S001	Purchase Specification Data Sheet Engine Generator for High Pressure Core Spray System	Revision 1
NEDC-30865	GE EQ Report-Perry Main Steam SRV Actuator Cylinder Air Valve, MPL Numbers B21-F041, F047, and F051	Revision 0
PY-CEI/NRR-1734-L	Implementation of Generic Letter 89-13, “Service Water System Problems Affecting Safety Related Equipment”	4/8/94
PY-CEI/NRR-1118-L	RAI-NRC Bulletin 88-04, “Safety Related Pump Loss”	1/9/90

Miscellaneous Documents

Number	Title	Revision/Date
PY-CEI/NRR-1081-L	RAI-NRC Bulletin 88-04, “Safety Related Pump Loss”	12/13/89
PY-CEI/NRR-0879-L	RAI-NRC Bulletin 88-04, “Safety Related Pump Loss”	7/11/88
QR-5804	Qualification Tests for Rockbestos Firewall III Chemically Cross-Linked Polyethylene Construction for Class 1E Service in Nuclear Generating Stations	Revision 3
SP-301-SO1-00	ECCS Motors, General Electric Company Environmental Qualification Report NEDC-30197, Book No. S01,	7/18/83
SP-504-4549-00	Conformed Specification for the Tanks	Revision 7
SP-559-4549-000	5.15 kV Power Cable Specification	Revision I
SP-560-4549-00	Specification Class 1E Small Power and Control Cables	Revision XX
SP-562-4549-00	Class 1E Diesel Generator Units	Revision 1
TAF 81834	Pump Curves	
600270036	Develop MIC Program Documents For Fleet	12/21/05
200068834	Inspection Report for the ECC “A” Heat Exchanger	3/2/05
200094880	Inspection Report for the RHR “C” Heat Exhanger	3/17/05
	Perry Project Design Criteria Volumes 1 and 2, by Gilbert Associates, Inc.	Revision 5
	EQ for Class 1E Safety Related Service, Small Emergency Service Water Pump Motors	Revision 2
	Deviation Analysis Report, Isolated Drywell Systems are Unprotected by Relief Valves	9/7/89
	Perry Evaluation of SOER 97-1	8/23/98
	Breaker Overhaul Tasks listed by Voltage Class/Due Date	3/1/06

Operability Determinations

Number	Title	Date
CR 06-00296	HPCS Pump Performance Due to DG Droop Adjustment	1/20/06
CR 06-00974	ADS Air Accumulators Design Stress	3/4/06
CR 06-00997	MSIV Air Accumulators Design Stress	3/4/06

Procedures

Number	Title	Revision
ARI-E22-P001	HPCS Diesel Generator Control Panel	Revision 4
ARI-H13-P601-0018-D1	RHR A Pump Room Sump Level High	Revision 11
ARI-H13-P601-0018-D2	RHR B Pump Room Sump Level High	Revision 11
ARI-H13-P601-0018-D3	RHR C Pump Room Sump Level High	Revision 11
ARI-H13-P601-0018-E1	HPCS Pump Room Sump Level High	Revision 11
ARI-H13-P601-0018-E2	LPCS Pump Room Sump Level High	Revision 11
ARI-H13-P601-0018-E3	RCIC Pump Room Sump Level High	Revision 11
ISTP	Pump and Valve Inservice Testing Program Plan	Revision 9
ONI-SPI C-4	EH13 System Operation	Revision 0
ONI-SPI D-1	Maintaining System Availability	Revision 0
ONI-SPI D-3	Cross-Tying Unit 1 and 2 Batteries	Revision 0
PDB-C0010	ICS Valve Stroke Time Correction Times	Revision 3
PEI-B13	RPV Control (Non-ATWS)	Revision L
PEI-B13	RPV Control (ATWS)	Revision J
PEI-B13	RPV Flooding	Revision J
PEI-B13	Emergency Depressurization	Revision H
PEI-T23	Containment Control	Revision G
PEI-SPI 4.2	RHR Loop B Flood Alternate Injection	Revision 1
PEI-SPI 4.6	Fast Fire Water Alternate Injection	Revision 1
PEI-SPI 5.1	HPCS Injection Prevention	Revision 0
PEI-SPI 5.2	LPCS and LPCI Injection Prevention	Revision 0
PEI-SPI 5.3	Feedwater Injection Prevention	Revision 1
PEI-SPI 6.1	LPCI A Outside the Shroud Injection	Revision 1
PEI-SPI 6.2	LPCI B Outside the Shroud Injection	Revision 1
PEI-SPI 6.3	LPCS Runout Injection	Revision 0
PEI-SPI 6.4	HPCS Runout Injection	Revision 1

Procedures

Number	Title	Revision
PEI-SPI 6.5	RHR C Runout Injection	Revision 1
SOI-P45/49	Emergency Service Water and Screen Wash Systems	Revision 11
SOI-R42	Div 1 DC Distribution, Buses ED-1-A and ED-2-A: Batteries, Chargers, and Switchgear	Revision 10
SOI-R42 Div 2	Div 2 DC Distribution, Buses ED-1-B and ED-2-B: Batteries, Chargers, and Switchgear	Revision 6
SOI-R43	Division 1 and 2 Diesel Generator System	Revision 25
SOI-E22A	HPCS System	Revision 15
SOI-E22B	Division 3 Diesel Generator	Revision 16
SOI-E51	RCIC System	Revision 20
SVI-E51-T1298	RCIC Actuation Logic System Functional Test	Revision 6
SVI-G33-T2003	RWCU Cold Shutdown Isolation Valves Operability Test	Revision 2
SVI-P57-T2201	Safety Related Air 1P57-F555B & F556B Leak Rate Test	Revision 2
SVI-P57-T2004	Safety Related Air 1P57-F572B & F574B Leak Rate	Revision 1
SVI-R43-T1317	Diesel Generator Start and Load Division 1	Revision 12
SVI-R43-T1318	Diesel Generator Start and Load Division 2	Revision 10
TAI-0501	Heat Exchanger Performance Monitoring	Revision 1

Surveillances (completed)

Number	Title	Date performed
	1E22F0012 Diagnostic Test Results	4/6/04
	1E51F0022 Diagnostic Test Results	6/14/05
	1G33F0001 Diagnostic Test Results	3/15/05
PTI-P57-P00001	ADS Air Leak Test	
SVI-633-T9131	Type C Local Leak Rate Test of 1G33 Penetration P131	3/26/05
SVI-E22-T1319	Diesel Generator Start and Load Division 3	12/14/05

Surveillances (completed)

Number	Title	Date performed
SVI-E22-T2001	HPCS Pump and Valve Operability Test	6/4/05, 8/27/05, 11/15/05, 1/15/05
SVI-E51-T2001	RCIC Pump and Valve Operability Test	8/10/05, 12/1/05, 1/20/06
SVI-P45T2003	HPCS ESW Pump and Valve Operability Test	8/26/05, 11/18/05
SVI-R42-T5202	Surveillance Instruction, Weekly 125V Battery Voltage and Category A Limits Check (Unit 1)	2/13/06
SVI-R42-T5203	Surveillance Instruction, Weekly 125V Battery Voltage and Category A Limits Check (Unit 2)	2/13/06
SVI-R42-T5211	Service Test of Battery Capacity, Division 1 (Unit 1)	2/13/06

Work Orders

Number	Title	Date
200035272	Relay IFC66K 50/51A(C) HPCS Pump 1E22C001	9/22/04
200035273	Relay IFC66K 50/51B HPCS Pump 1E22C001	9/22/04
200109234	Replace Entire125 volt Battery 1B and Rack (all cells)	2/25/05
200143564	BKR L2207 Overhaul to Remove Breaker Interference Plate	8/1/05
200149124	Replace Cells 1 and 53 in 125 volt Battery 1B	7/29/05

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CDBI	Component Design Bases Inspection
CR	Condition Report
CFR	Code of Federal Regulations
DC	Direct Current
DG	Diesel Generator
DRS	Division of Reactor Safety
ECP	Engineering Change Package
ESW	Essential Service Water
EF	Degree Fahrenheit
GE	General Electric
GL	Generic Letter
gpm	gallons per minute
HPCS	High Pressure Core Spray
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IST	Inservice Testing
kV	Kilovolt
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OA	Other Activities
PARS	Publicly Available Records
PEI	Plant Emergency Instruction
PRA	Probabilistic Risk Assessment
psig	pounds per square inch gauge
psid	pounds per square inch differential
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RIS	Regulatory Information Summary
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRV	Safety Relief Valve
TS	Technical Specifications
USAR	Updated Safety Analysis Report
URI	Unresolved Item