UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

January 13, 2012

Mr. Vito Kaminskas
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-A290
Perry, OH 44081-0087

SUBJECT: PERRY NUCLEAR POWER PLANT COMPONENT DESIGN BASES INSPECTION 05000440/2011008

Dear Mr. Kaminskas:

On December 2, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection (CDBI) at your Perry Nuclear Power Plant. The enclosed report documents the results of this inspection, which were discussed on December 2, 2011, with Mr. John Grabnar and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. All three findings involved a violation of NRC requirements. In addition, a Traditional Enforcement Severity Level IV violation was identified. Because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest these NCVs, 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 addition, if you disagree with the cross-cutting aspect assignments in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Website 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 05000440/2011008;
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	05000440/2011008(DRS)
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	October 31, 2011 through December 2, 2011
Inspectors:	C. Tilton, Senior Engineering Inspector, Lead B. Jose, Senior Engineering Inspector, Electrical R. Langstaff, Senior Engineering Inspector, Operations M. Jones, Engineering Inspector, Mechanical S. Kobylarz, Electrical Contractor C. Baron, Mechanical Contractor
Observer:	I. Hafeez, Engineering Inspector, Observer
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000440/2011008, 10/31/2011 – 12/02/2011; Perry Nuclear Power Plant, Component Design Bases Inspection (CDBI).

The inspection consisted of three weeks onsite baseline inspection that focused on the design of components. The inspection was conducted by regional engineering inspectors and two consultants. Three (Green) findings and a Traditional Enforcement Severity Level IV violation were identified by the inspectors. All three findings and Severity Level IV violation were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be (Green) or 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 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure safety-related equipment would be adequately protected from internal flooding. Specifically, the licensee failed to adequately evaluate the volume of water originating from a postulated crack in service water (SW) piping within the control complex. This finding was entered into the licensee's corrective action program. The corrective actions included performing additional analyses, establishing compensatory measures, issuing procedure orders, and revising operating procedures.

  The performance deficiency was determined to be more than minor because it was associated with the Initiating Events cornerstone attribute of Equipment Performance and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Based on the Phase III Analysis, the inspectors determined the finding was of very low safety significance (Green). The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. (Section 1R21.3b.(1))

- Severity Level IV: The inspectors identified a Severity Level IV violation of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Reactors," for failure to report within eight hours an unanalyzed condition that significantly degrades plant safety. Specifically, the licensee failed to notify NRC upon discovery of a postulated internal flood in the control complex could result in loss of single failure capability of safety-related equipment. This violation was entered into the licensee's corrective action program.

  The performance deficiency was determined to involve a traditional enforcement violation because it potentially impeded or impacted the regulatory process. The traditional enforcement violation was determined to be more than minor because the information that was not provided through the event notification had a material impact on safety and licensed activities. The traditional enforcement violation was determined to

be a Severity Level IV violation because the failure to report within eight hours an unanalyzed condition did not result in an unacceptable change to the facility or procedures. An evaluation for cross-cutting aspect was not applicable because this was a traditional enforcement violation. (Section 1R21.3b.(4))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", for failure to adequately evaluate the capability of motor control starter contactors to operate during design basis degraded voltage conditions. Specifically, the licensee did not analyze all circuit elements of resistance and failed to incorporate the latest results of calculated plant bus voltages.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating System Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability. Specifically, the licensee performed an operability evaluation taking into account all resistances in the circuit, the latest load flow analysis and test data and concluded there was sufficient voltage available. This finding has a cross-cutting aspect in the area of Resources for failure to ensure complete, accurate, and up-to-date design documentation, procedures, work packages and correct labeling of components. [H.2(c)] (Section 1R21.3b.(2))

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to test safety-related motor starter contactors at design basis conditions. Specifically, the licensee failed to demonstrate the ability of ESW Pump 'A' discharge valve 1P45F0130A motor starter contactor to operate at minimum pickup voltage during design basis degraded voltage conditions. This finding was entered into the licensee's corrective action program.

  The performance deficiency was determined to be more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. The finding screened as of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability. Specifically, after further evaluation, the licensee's engineering staff concluded the issue did not impact current operability because periodic testing for other type of contactors provided validation the valve motor contactor would operate when required for the postulated degraded voltage conditions. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution Corrective Action Program for failure to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. [P.1(d)] (Section 1R21.3b.(3))

B. Licensee-Identified Violations

No violations of significance were identified.

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 inspectable 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 Probabilistic Risk Assessment (PRA) and the Perry Standardized Plant Analysis Risk (SPAR) Model. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 4.0 and/or a risk reduction worth of greater than 1.001. This selection criterion was consistent with the inspection guidance contained in Inspection Procedure 71111.21, Section 03.01. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios.

The inspectors also used additional component information such as a margin assessment in the selection process. This design margin assessment considered original design reductions caused by design modification, power uprates, or reductions due to degraded material condition. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as performance test results, significant corrective actions, repeated maintenance activities, Maintenance Rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem areas/equipment, and system health reports. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

The inspectors also identified procedures and modifications for review that were associated with the selected components. In addition, the inspectors selected operating experience issues associated with the selected components.

This inspection constituted 20 samples as defined in Inspection Procedure 71111.21.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report (FSAR), Technical Specifications (TS), design basis documents, drawings, calculations 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, Institute of Electrical and Electronics Engineers (IEEE) Standards and the National Electric Code, to evaluate acceptability of the systems' design. The NRC also evaluated licensee actions, if any, taken in response to NRC issued operating experience, such as Bulletins, Generic Letters (GLs), Regulatory Issue Summaries (RISs), and Information Notices (INs). The review was to verify that the selected components would function as designed when required and support proper operation of the associated systems. The attributes reviewed were those needed for a component to perform its required function and included process medium, energy sources, control systems, operator actions, and heat removal. The attributes reviewed to verify that the component condition and tested capability was consistent with the design bases and was appropriate included installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, preventive maintenance activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, and licensee corrective action program (CAP) documents. Field walkdowns were conducted for all accessible components to assess material condition and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 16 components were reviewed:

- Scram Discharge Drain Valves (1C11F181/F011): The inspectors reviewed air operated valve (AOV) calculations and analyses to ensure the valve remained capable of performing its safety function under worst case design basis conditions. The inspectors reviewed valve set up control documentation and verified those inputs correlated with design basis calculations. The inspectors reviewed thrust calculations, maximum expected differential pressure calculations, valve set-up, and weak-link analysis. The inspectors conducted a field walkdown of the valve to verify the installed configuration, accessibility, environmental conditions, and valve material conditions. The inspectors also reviewed valve surveillance results to ensure the valves continued to meet stroke

time requirements in order to meet their licensing basis. The inspectors examined condition reports and corrective maintenance documents to ensure valve operability was maintained when procedural steps were not satisfied. The team also verified procedures and annunciator response documentation to ensure the design basis values were appropriately translated.

- Residual Heat Removal (RHR) Pump Anet (1E12C0002A): The inspectors reviewed pertinent information to verify the ability of the RHR pump of meeting its design basis. The inspectors also reviewed the adequacy of the basis for the pump's performance test acceptance criteria, pump performance test procedure and results of the most recent test to verify current performance. The inspectors reviewed calculations associated with pump hydraulic performance, net positive suction head (NPSH), minimum flow, and runout flow to ensure the pump was capable of successfully performing under all conditions. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system and design engineers and performed a walkdown of the pump and associated equipment to verify the material condition and surrounding environment of the equipment was adequate.

  The inspectors reviewed motor sizing calculation and motor/pump performance curves to confirm the electrical load was correctly included in the emergency diesel generator (EDG) and bus loading calculations. The inspectors also reviewed motor feeder ampacity and short circuit capability. Additionally, the inspectors reviewed the protective relay setpoint calculations and relay settings to assess the adequacy of the circuit protection under normal and faulted conditions and verify that trip setpoints would not permit the feeder breaker to trip during pump motor highest loading conditions. Available motor voltage was also reviewed to confirm the availability and capability of the pump to perform its safety function under most limiting conditions. The inspectors verified the technical specification allowance of +/- 2 percent frequency for the emergency diesel generators did not adversely affect the performance of the RHR pump motor. The inspectors reviewed motor control logic and wiring diagrams to ascertain compliance with system operation requirements and confirmed adequacy of environmental qualification of motor under accident conditions. The electrical separation was also reviewed to ensure that the redundancy of safety divisions was not compromised. The inspectors reviewed motor testing and inspection procedures for on-line and off-line conditions to assure that the testing parameters were adequate and in accordance with industry standards. The review also included recent electrical maintenance and test activities to confirm the readiness of the component to perform its required functions during system demands.

- Residual Heat Removal (RHR) Heat Exchanger Bypass Valve (1E12F0048A): The inspectors reviewed required thrust calculation and the weak link analyses for the motor operated valve to verify its capability to operate under the most limiting conditions. The inspectors reviewed the most recent diagnostic testing results, and reviewed the in-service testing procedures and recent results to verify valve performance. The inspectors reviewed the calculated differential

pressure for the valve to verify it bounded safety analyses. The inspectors also verified the basis of the voltage used in valve analyses to ensure that bounding values were used. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system and valve engineers; and performed a walkdown of the valve and associated equipment to verify the material condition of the equipment was adequately maintained.

- Residual Heat Removal (RHR) Heat Exchanger A (1E12B0001A): The inspectors reviewed calculations of required thermal performance of the RHR heat exchanger to meet its design basis and the basis of the Generic Letter 89-13 thermal performance test acceptance criteria to ensure values used bounded safety analyses. The inspectors reviewed thermal test performance procedure, results and data analysis. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system and GL 89-13 engineers, and performed a walkdown of the heat exchanger and associated equipment to verify the material condition of the equipment was adequately maintained.

- Residual Heat Removal (RHR) Pump 'A' Minimum Flow Valve Motor and Starter Voltage (1E12F0064A): The inspectors reviewed the required thrust calculation and the weak link analyses for the motor operated valve to verify its capability to operate under the most limiting conditions. The inspectors reviewed the most recent diagnostic testing results, and reviewed the in-service testing procedures and recent results to verify valve performance. The inspectors reviewed the control logic associated with the valve, including the basis of the time delay in automatically opening the valve. The inspectors reviewed the calculated differential pressure for the valve to verify it bounded the safety analyses. The inspectors also verified the basis of the voltage used in the valve analyses to ensure that bounding values were used. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system and valve engineers, and performed a walkdown of the valve and associated equipment to verify the material condition of the equipment was adequately maintained.

  The inspectors reviewed RHR Pump 'A' minimum flow valve motor voltage to verify that it was capable of meeting its design basis requirements at degraded voltage conditions at the motor starter and the motor terminals. The inspectors reviewed motor elementary wiring diagrams to ascertain compliance with system operation requirements and to verify the adequacy of the circuit model used to calculate the voltage available at the motor and motor starter contactor. The inspectors also confirmed the adequacy of environmental conditions of motor starter under accident conditions. The inspectors verified the calculated minimum voltage at the motor terminals and motor starter contactor was based on the motor control center voltage available at degraded grid conditions. The review also included recent electrical maintenance and motor starter test activities to confirm the readiness of the component to perform its required functions during system demands. Finally, corrective action documents and system health reports were reviewed to verify deficiencies were appropriately identified and resolved, and that the motor starter was properly maintained.

- High Pressure Core Spray (HPCS) Injection Isolation Valve (1E22-F004): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. The inspectors also reviewed the thrust/torque calculation, differential pressure calculation, valve set-up calculation, and weak link analyses. The inspectors reviewed surveillance test results to determine whether testing, inspection, and maintenance were being performed in accordance with program requirements. Diagnostic testing and in-service testing (IST) surveillance results, including stroke time testing, were reviewed to verify acceptance criteria were met and performance degradation could be identified.

- HPCS Room Cooler (1M39B0003): The inspectors reviewed calculations of required thermal performance of the RHR heat exchanger to meet its design basis and the basis of the Generic Letter 89-13 thermal performance test acceptance criteria to ensure values used bounded safety analyses. The inspectors reviewed thermal test performance procedure, results, and data analysis. The inspectors reviewed selected condition reports, work orders, and conducted interviews with the system and GL 89-13 engineers.

- Emergency Closed Cooling Water Heat Exchanger (ECC) (P42B0001A): The inspectors reviewed various calculations associated with the thermal performance of the heat exchangers under design basis accident and transient conditions, including conditions that impart maximum expected system heat load, maximum service water system supply temperature and minimum service water flow to the heat exchanger. The inspectors reviewed results and analysis from the licensee's thermal performance-testing program to verify the analyzed performance would be bounded by the as-found conditions. The inspectors reviewed condition reports, open and inspect surveillance documents associated with the heat exchanger and system health report documentation. The inspectors verified that tube plugging and blockage were being controlled based on thermal performance and that no such blockage was impacting heat exchanger thermal performance.

- Emergency Service Water (ESW) Pump (1P45C0001A): The inspectors reviewed SW pump calculations, maintenance history, operations history, and design requirements to verify that the pump was maintained such that it remains capable of operating in accordance with its design basis. The inspectors reviewed IST results and service water (SW) system performance testing to ensure that design basis requirements were correctly translated into test acceptance criteria and that the tests demonstrated the pumps capability to perform its design basis functions. The inspectors reviewed intake and forebay silting data and chemical treatments to ensure pump fouling mechanisms are being monitored and controlled in accordance with procedures that ensure system function is maintained.

  The inspectors also reviewed the ESW pump motor to determine whether it could fulfill its design basis function of providing adequate horsepower for the pump to deliver the required cooling water flow to safety-related loads. The inspectors performed a walk-down of the ESW pump, the pump motor, and the pump house

to independently assess the licensee's configuration control, and the pump motor operating environment and material condition. The inspectors confirmed the adequacy of environmental conditions of the motor under accident conditions. The inspectors reviewed the ESW pump performance curve and design basis flow requirement to evaluate the required capacity for the break horsepower required by the pump during design basis conditions. The inspectors reviewed the 4160 Vac system load flow calculation and motor nameplate data to confirm that adequate voltage would be available at the motor terminals for design basis conditions. The inspectors also reviewed the motor overcurrent relay setting calculation, relay settings and recent overcurrent relay calibration tests to evaluate whether the protective relays would provide for reliable motor operation at design basis minimum voltage conditions. Finally, corrective action documents and system health reports were reviewed to verify deficiencies were appropriately identified and resolved, and that the motor was properly maintained.

- Emergency Service Water Pump Discharge Strainer (1P45D0002A): The inspectors reviewed procedures, surveillances results, trend data, differential pressure, and debris loading calculations to ensure the strainers have remained capable of performing their intended functions while subject to limiting design conditions. The inspectors reviewed strainer design requirements to ensure debris-loading assumptions were consistent with industry guidance. The inspectors also reviewed the monitoring program, along with maintenance and operations procedures, to verify that historical differential pressure remained below the licensee established acceptance criteria for service water system operability. The inspectors verified procedures and operator actions to ensure design basis flow isn't impacted by debris blockage on the pump discharge strainers.

- Safety Related Instrument Air Storage Tank A (1P57A0003A): The inspectors reviewed calculations related to the storage tank capacity, including the minimum allowable pressure and the allowable leakage limits to verify its capacity under the most limiting conditions. The inspectors reviewed the basis for the pressure alarm setpoints and the leak test criteria, and reviewed the inservice testing acceptance criteria for the associated instrument air system boundary check valves. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system engineer, and performed a walkdown of the valve and associated equipment to verify the material condition of the equipment was adequately maintained.

- Containment Isolation Valve (1P57F0015A): The inspectors reviewed the required thrust calculation and the weak link analyses for the motor operated valve to verify its capability to operate under the most limiting conditions. The inspectors reviewed the most recent diagnostic testing results, and reviewed the in-service testing procedures and recent results to verify valve performance. The inspectors reviewed the calculated differential pressure for the valve to verify it was bounding for all operating conditions. The inspectors reviewed selected condition reports, work orders, conducted interviews with the system and valve

engineers; and performed a walkdown of the valve and associated equipment to verify the material condition of the equipment.

The inspectors reviewed motor operated valve (MOV) calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, 480 Vac power and control voltage drop, and breaker/fuse coordination. As a result of this review, the inspectors verified adequate voltage will be available at the motor starter and motor terminals during loss of coolant accident (LOCA) and degraded voltage conditions. The inspectors also reviewed condition report history and verified there are no issues affecting reliability of this MOV.

- Unit 1 Division 1 Battery (1R42S0002): The inspectors reviewed various electrical calculation associated with the safety-related DC battery to verify the battery was designed to perform its function and pick up the required loads during a LOCA and station blackout (SBO). These calculations included battery sizing, voltage drop, minimum voltage, and short circuit. The calculation review verified methodology, design inputs, assumptions, and results. The inspectors also reviewed TS surveillance requirements and completed surveillances to confirm that sufficient capacity existed for the battery to perform its safety function. The battery's performance history including cell voltage, charging, specific gravity, electrolyte level, and temperature correction were also reviewed to ensure acceptance criteria were met and performance degradation would be identified.

- EFD-1-A Battery Charger (1R42S0006): The inspectors reviewed 125 Vdc battery charger sizing calculations, TS surveillance requirements, and completed surveillances to confirm that sufficient capacity existed for the battery charger to perform its safety function and was being adequately maintained. The inspectors reviewed the vendor manual recommendations for charger maintenance, compared them with station maintenance procedures, and verified the charger was appropriately maintained including periodic replacement of components such as electrolytic capacitors. The inspectors reviewed battery charger feeder cable data and breaker/fuse coordination data and verified a fault on the load side of the charger will be cleared in time to preclude damage to the charger and its feeder cable. The inspectors also reviewed industry operating experience related to locking up of charger output and verified the Perry station battery chargers were not susceptible to such lock ups.

- Emergency Diesel Generator (EDG) 1A Starting Components and Loading (1R43S0001A): The inspectors inspected information related to the EDG 1A to verify that it was capable of meeting its design basis requirements for starting components such as starting air solenoids, generator field flash, the generator breaker close coil and the design basis load requirements. EDG 1A is designed to provide standby power to 4160 Vac safety-related emergency bus EH11 when the preferred power supply is not available. The inspectors reviewed the one-line diagrams for the EDG and the vendor nameplate rating data for the diesel generator. The inspectors reviewed the equipment specifications and vendor manual to determine the EDG rated output capability and the ratings for selected

starting components. The inspectors reviewed the adequacy of voltage available for selected starting components and their surveillance testing. The inspectors also reviewed the EDG loading study for the worse case design basis loading conditions to verify that the break horsepower basis for selected pump motors was considered in the loading study at worse case motor load conditions. The inspectors walked down EDG 1A to determine the material condition and the operating environment for indications of degradation of equipment. Finally, corrective action documents and system health reports were reviewed to verify deficiencies were appropriately identified and resolved, and that the generator and selected starting components were properly maintained.

- Startup Transformer (PY1 SUT): The inspectors reviewed the failed startup transformer's oil analysis reports, relay protection scheme operation at the moment of failure and alarms records to determine whether the transformer had been properly maintained and operated. The inspectors also reviewed the modification package to replace Unit 1 startup transformer, the vector diagrams, new protection scheme, and drawings for alarms and controls, vendor manuals, test procedures and short circuit analysis including relay coordination curves. The inspectors also performed a visual inspection of the failed and newly installed startup transformers to assess material condition and the presence of vulnerabilities.

b. Findings

(1) Failure to Adequately Protect Safety-Related Equipment from Internal Flooding

Introduction: The inspectors identified a finding of very low safety significance (Green) involving a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." for failure to ensure that safety-related equipment would be adequately protected from internal flooding. Specifically, the licensee failed to adequately evaluate the volume of water originating from a postulated crack in service water (SW) piping within the control complex.

Description: On November 10, 2011, as a result of following up on licensee's actions related to NRC Information Notice (IN) 2005-30, the inspectors reviewed portions of design basis calculation JL-083, Revision 2, "Flooding Analysis of CCB, IB, and FHB - Floor Elevation 574'-10"." This calculation was performed to address the flooding effects of postulated pipe cracks for moderate energy piping in the Intermediate, Fuel Handling, and Control Complex Buildings. In the case of postulated floods originating in the control complex building, this calculation assumed the SW pumps would be stopped within 30 minutes, but did not assume any additional operator actions. The calculation concluded, in part, that pipe cracks in the control complex would be identified by sump instrumentation. It also concluded that with the installation of a six inch flood barrier in the IB 103 doorway, (modification ECP 00-8075), water from a postulated pipe crack would be contained within the control complex without affecting any safety-related equipment.

The inspectors performed a detailed review of the flooding analysis, specifically the postulated crack in the 42-inch diameter SW piping within the control complex building.

After further review of the flooding analysis, the inspectors noted the calculation did not take into account the entire volume of water that could drain into the building. The calculation only accounted for the water located in the return header of the SW pump. The calculation failed to account for the water located in the supply header of the SW pump, which by gravity would pour out of the crack and into the control complex building. In addition, the calculation failed to account for the siphon effect in the service water system as a result of the pipe break. The inspectors were concerned the licensee did not properly evaluate a service water pipe crack flooding the control complex and did not have the proper measures in place to mitigate the effects this event.

In response to the inspectors' questions, the licensee performed an informal analysis and determined calculation JL-083 failed to account for the total volume of the piping that could drain through the postulated crack and erroneously predicted a maximum level of less than six inches in the control complex building. After assessing the actual volume of water that would drain to the building, the licensee performed informal calculations during the inspection and determined the water level in the control complex could exceed 21 inches and damage redundant safety-related equipment. At the time of discovery, operating procedures did not include steps to vent the piping to prevent the siphon effect of additional water from the service water system.

The licensee initiated Condition Report CR-2011-05217 on November 10, 2011, to address this issue. As a result of additional evaluation, the licensee issued night orders to the operators, installed sandbags at the doorway between the control complex and intermediate building, installed scaffolding for access to service water vent valves, and revised operating procedures. These compensatory measures were taken to ensure the operability of safety-related equipment in the control complex and auxiliary buildings. This safety-related equipment includes the pumps on both trains of Emergency Closed Cooling (ECC) and Residual Heat Removal (RHR), the Low Pressure Core Spray (LPCS) pump, and Reactor Core Isolation Cooling (RCIC) pump.

The licensee performed subsequent analyses, issued a Prompt Functional Assessment on November 22, 2011, and a Reportability Review on November 28, 2011. Additional evaluation determined it would have taken a minimum of 8.6 hours to reach a flood depth of 21 inches in the control complex without operator actions to vent the SW piping and prevent the siphon effect. In fact, the operators would have to vent the SW piping within approximately 7.5 hours to ensure the final level remained below the 21-inches required to prevent the impact on safety related components. These analyses were based on the maximum postulated pipe crack size and assumed SW pumps were stopped after 30 minutes with no additional operator actions. Therefore, these analyses showed additional operator actions were necessary to maintain operability.

Analysis: The inspectors determined that failure to ensure that safety-related equipment would be adequately protected from internal flooding was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. The performance deficiency was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of Equipment Performance and affected the cornerstone objective of limiting the likelihood of events that upset plant

stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the licensee failed to correctly analyze the consequences of a postulated moderate energy pipe crack and to provide adequate operating procedures to mitigate the event.

The Region III Senior Reactor Analysts (SRAs) evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of Findings," Table 4a for the Initiating Events Cornerstone. The inspectors answered "Yes" to the "External Event Initiators" screening question "Does the finding increase the likelihood of a fire or internal/external flood"?... since the probability and consequences of a flooding event in the Control Complex were increased as a result of the performance deficiency. Therefore, a Phase III Significance Determination Process (SDP) evaluation was required.

The SRAs used Electric Power Research Institute (EPRI) 1013141, Revision 2, Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk-Assessments" to estimate the random pipe failure frequency of the SW piping in the control complex. In the EPRI Report, Table ES-2 gave a value of 1.90E-7/foot reactor year for the failure probability of a leak of a magnitude of at least 100 gpm for non-safety-related Boiling Water Reactor (BWR) SW lake piping of greater than ten inch diameter. The diameter of the SW pipe is greater than 24 inches and there are approximately 550 feet of SW piping in the control complex. This results in a flood initiating event probability of approximately 1.0E-4/yr.

The SRAs used SAPHIRE Version 8.0.7.17 and the Perry Standardized Plant Analysis Risk (SPAR) model (Version 8.15) to estimate a conditional core damage probability (CCDP) for a flood initiating event in the control complex. The SRAs performed an Initiating Event analysis using the SPAR model. Using Section 3.3.7 of the Individual Plant Examination (IPE) for Perry (dated July 1992), a loss of instrument air initiating event was assumed followed by a failure of the following equipment as a result of a flooding event in the Control Complex: the Emergency Closed Cooling (ECC) pumps, the Low Pressure Core Spray (LPCS) pump, the Residual Heat Removal (RHR) pumps, the Reactor Core Isolation Cooling (RCIC) pump and the SW pumps.

The result of the initiating event analysis using the SPAR model was a CCDP of 2.0E-3. Using this CCDP, the estimated risk significance due to flooding was calculated to be 2.0E-7/yr. The dominant core damage sequence involved a loss of instrument air initiating event with failures of RCIC, suppression pool cooling, containment spray, containment venting, and late injection.

Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, IMC 0609, Appendix A, Attachment 3, "User Guidance for Screening of External Events Risk-Contribution," was used to screen external event contributions. Fire and external flooding were not contributors to the risk significance of the finding.

The seismic contribution to the risk of internal plant flooding was evaluated using guidance from the Risk-Assessment Standardization Project (RASP) handbook. Only the "Bin 2" seismic events were assumed to represent a delta CDF. "Bin 2" is defined in the RASP handbook as seismic events with intensities greater then 0.3g but less than 0.5g. Earthquakes of lesser severity are unlikely to result in large pipe failures and

earthquakes of a larger magnitude could result in major structural damage throughout the plant and thus would not increase baseline risk. The frequency of an earthquake in "Bin 2" was estimated to be 1.3E-5/yr. Using the CCDP of 2.0E-3 determined above, the estimated risk significance due to seismic failure of the SW piping was thus calculated to be 2.6E-8/yr.

The total risk significance is the sum of the risk significance due to internal flooding from random pipe breaks (2.0E-7/yr) plus the risk due to seismically-induced pipe breaks (2.6E-8/yr) or 2.3E-7/yr.

Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, the potential risk-contribution for this finding from large early release frequency (LERF) was screened using the guidance of IMC 0609 Appendix H, "Containment Integrity Significance Determination Process." Perry is a BWR with a Mark III containment. Sequences important to LERF for a Mark III containment include inter-system loss of coolant accidents (ISLOCAs), transients and small break LOCAs (i.e., high reactor coolant system (RCS) pressure sequences), and station blackout (SBO) sequences. For the Phase III analysis, high RCS pressure sequences were the dominant core damage sequences for this finding. As stated in Table 5.2, "Phase II, Assessment Factors – Type A Findings at Full Power," of IMC 0609 Appendix H, sequences involving high RCS pressure have a LERF Factor of 0.2. Using this LERF Factor, the risk significance due to LERF was estimated to be 4.6E-8/yr.

Based on the Phase III analysis, the inspectors determined that the finding was of very low safety significance (Green).

The inspectors determined the cause of this finding did not represent current licensee performance and no cross-cutting aspect was assigned. Specifically, calculation JL-083 was performed in 2004, therefore, did not reflect current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, 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.

Contrary to the above, as of November 10, 2011, the licensee failed to ensure that safety-related equipment was adequately designed and protected from internal flooding. Specifically, the licensee failed to evaluate the adequacy of their design by not accounting for all the volume of water originating from a postulated crack in service water (SW) piping within the control complex. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR-2011-05217, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000440/2011008-01, Failure to Adequately Protect Safety Related Equipment from Internal Flooding).

(2) Inadequate Control Circuit Voltage Calculation for Safety-Related Motor Starter Contactors

Introduction: A finding of very low safety significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", was identified by the inspectors for failure to adequately evaluate the capability of motor control starter contactors to operate during design basis degraded voltage conditions. Specifically, the licensee did not analyze all circuit elements of resistance and failed to incorporate the latest results of calculated plant bus voltages.

Description: On November 30, 2011, while reviewing Calculation PSTG-0030, "Voltage Drop in Control Circuits of Safety Related Motor Control Center Starters, NEMA Sizes 1, 2, 3, and 4 Starters," and interviewing licensee's staff, the inspectors noted the circuit model used to calculate the voltage available at safety-related motor starters failed to include circuit elements that addressed the effects of fuse resistance and control circuit contact resistance when determining the voltage available at the motor starter contactor and contactor interposing relay.

The inspectors also observed the licensee had incorrectly evaluated the change in plant bus voltages when determining the impact on calculated voltage for safety-related motor starters in their review of the results of the most recent plant load flow study. In condition report (CR) CR-G202-2011-97622, which addressed the most recent plant load flow study, the licensee's immediate actions taken stated: "The anticipated impact of updating the output calculations would be a slight overall improvement in the computed degraded voltage. Updates to the calculations are required to maintain configuration control." The senior reactor operator (SRO) later evaluated this CR and commented stating: "This is a configuration control issue and does not involve plant equipment."

After further review, the inspectors found an instance in which the computed degraded voltage did not improve but was degraded. Specifically, the team observed the motor starter for the ESW Pump 'A' discharge valve, 1P45F0130A, had the most limiting (smallest) voltage margin available for contactor and interposing relay pick-up. The 480 Vac motor control center (MCC) Bus EF1A12 feeds 1P45F0130A. The voltage in the latest load flow study, PSTG-0001, Revision 5, for the 480 Vac MCC Bus EF1A12 decreased six volts from the value that was used to calculate the control voltage available in Calculation PSTG-0030. The inspectors were concerned the licensee incorrectly evaluated the effect of the decrease in voltage at the motor starter. Without sufficient voltage available, the valve might not open on demand and the ability of ESW Pump 'A' to provide essential cooling water during a design basis accident condition would be affected.

As a result of the inspectors questioning, the licensee performed a preliminary calculation that incorporated the effects of fuse and contact resistances and the most recent MCC bus voltages from PSTG-0001. The preliminary calculation determined the new calculated voltage available at the 1P45F0130A starter contactor was 94.52 volts, which compared to the last verified relay pick-up voltage of 94.50 volts presented a small margin in available voltage. The small margin in available voltage prompted the licensee

to perform an immediate operability assessment. Condition Report 2011-06107 documented the operability assessment and concluded that based on a recent performance test where the relay picked-up at 94.5 volts and the original relay tests showing the relay consistently picking-up at 84.5 volts (much less than the manufacturer's listed minimum voltage) there was reasonable assurance of operability for the valve at degraded voltage conditions.

Analysis: The inspectors determined that failure to adequately evaluate circuit elements of resistance and to incorporate results of calculated changes in plant bus voltages was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, failure to adequately determine the capability of motor control starter contactors affected the availability of the ESW 'A' discharge valve to operate and provide essential cooling water during a design basis accident.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the mitigating system cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in a loss of operability or functionality. Specifically, the licensee performed an operability evaluation taking into account all resistances in the circuit, the latest load flow analysis and test data and concluded there was sufficient voltage available.

The inspectors determined this finding had a cross-cutting aspect in the area of Resources for the failure to ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety. Specifically, the licensee failed to ensure calculations were complete, accurate and represented the most up-to-date information. [H.2(c)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that the design bases are correctly translated into specifications, drawings, procedures, and instructions. It also requires, 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.

Contrary to the above, as of November 30, 2011, the licensee failed to check the adequacy of their design by incorrectly evaluating the capability of motor control starter contactors to operate during design basis degraded voltage conditions. Specifically, the licensee did not analyze all circuit elements of resistance and failed to incorporate the latest results of calculated plant bus voltages. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR-2011-06107 and CR-2011-05495, this violation is being treated as an NCV, consistent with Section 2.3.2

of the NRC Enforcement Policy (NCV 05000440/2011008-02, Inadequate Control Circuit Voltage Calculation for Safety-Related Motor Starter Contactors).

(3) Failure to Test Safety-Related Contactors at Degraded Voltage Conditions

Introduction: A finding of very low safety significance and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", was identified by the inspectors for failure to test safety-related motor starter contactors at design basis conditions. Specifically, the licensee failed to demonstrate the ability of several motor starter contactors but more specifically the ESW Pump 'A' discharge valve 1P45F0130A motor starter contactor to operate at minimum pickup voltage during design basis degraded voltage conditions.

Description: On November 16, 2011, while reviewing corrective action document CR-G202-2008-38977, the inspectors noticed the licensee committed to perform initial baseline pickup/drop-out tests of 32 motor starter contactors within two years and subsequently at an eight year frequency. This action included testing of 1P45F0130A motor starter for the ESW Pump 'A' discharge valve. The licensee determined these baseline and periodic tests were required because in the past, the licensee had only tested a limited sample of spare motor starters and not the motor starters installed at the plant. These results lacked representation and acceptable statistical analysis to bind the actual plant configuration.

In addition, while reviewing the limited test results available for motor starter contactors, the inspectors found inconsistent data in the special engineering test analysis report (TAR) 95-089. The special engineering test was performed to provide a basis to accept calculated voltages at contractors that were found to be less than the minimum specified by the manufacturer. The inconsistent data resulted from "swapped" measured hot and cold size 2B1 contactor pickup values and provided non-conservative test results. The test results were later used as modified acceptance criteria for contactor minimum pickup voltage for the 1P45F0130A motor starter. The 1P45F0130A motor starter was identified as the limiting component based on the margin available in the calculated voltage at the starter contactor. The modified acceptance criteria, derived from the limited test of the size 2B1 contactors, provided the basis for accepting the voltage available at the 1P45F0130A motor starter until periodic testing was performed on the motor starter installed in the plant. The inspectors were concerned as the licensee was using non-conservative acceptance criteria to demonstrate the ability of the starter contactor to perform its function.

The inspectors noted the previous CDBI performed at Perry documented (IR 20005000440/2008006) documented a licensee identified NCV of Title 10 CFR Part 50, Appendix B, Criterion III, Design Control for failure to provide verification of the adequacy of design, such as by the performance of a suitable testing program. The report documented that between mid 1990s and June 2008, approximately 40, 480 Vac motor control center components, such as, starter coils and interposing relays, had not been periodically tested to verify the adequacy of available voltages for their proper function. This condition was identified in the licensee's corrective action program as Condition Report 08-38977. The finding was of very low safety significance because the failure to perform periodic testing did not impact current operability of these

components since the purpose of the periodic testing was to assure the minimum voltage remained acceptable as these components aged.

The inspectors requested to see the results of the corrective action referred to in the 2008 Inspection Report, CR 08-38977. Specifically, the inspectors requested the results of the baseline pickup/drop-out test for 1P45F0130A contactor due one year ago per the condition report. This specific contactor was selected for detailed review because it had a history of having low margin. The licensee found that the test had not been performed. After further investigation, the licensee determined the baseline test for 1P45F0130A contactor; originally due on January 29, 2011, was missed because the maintenance plan was not updated to require this test by the specified date. After discussions with licensee staff, the inspectors determined the work scheduling group was not properly notified of the initial two year preventive maintenance test requirement.

The inspectors were concerned that by not testing the installed contactors at the plant, the effects of aging on plant equipment could lead to degradation and the ability of starter contactors to operate at design basis degraded voltage conditions at less than the manufacturer's minimum voltage value could go unnoticed. The licensee performed an analysis during the inspection to demonstrate sufficient voltage was available in the 1P45F0130A pick-up voltage to address the "swapped" measured value which was more conservative. The licensee also initiated corrective action CR-2011-05507 and concluded the issue did not impact current operability because periodic testing provided on-going validation of the special test data in TAR 95-089 for other type of contactors. In addition, the licensee reinstated the testing schedule and plans to perform baseline and periodic surveillance on all the contactors to demonstrate their capability to operate at design basis degraded voltage conditions at less than the manufacturer's minimum voltage.

Analysis: The inspectors determined that failure to test safety-related contactors to verify the ability to operate at design basis degraded voltage conditions was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. The performance deficiency was determined to be more than minor because if left uncorrected it would have the potential to lead to a more significant safety concern. Specifically, failure to perform an initial baseline test and subsequent periodic test plan for several motor starter contactors but more specifically the ESW Pump 'A' discharge valve 1P45F0130A motor starter contactor could have led to degradation of the component to go unnoticed and as a result failure of the component to operate at design basis degraded voltage conditions at less than the manufacturers minimum voltage value.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the mitigating system cornerstone. The finding screened as of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in a loss of operability or functionality. Specifically, the licensee's evaluation on CR-2011-05507, determined that the issue did not impact current operability because periodic testing provided on-going validation of the special test data in TAR 95-089 for other type of contactors.

The inspectors determined this finding had a cross-cutting aspect in the area of Problem Identification and Resolution Corrective Action Program for failure to ensure that issues potentially impacting nuclear safety were promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance. Specifically, the licensee failed to timely ensure a baseline and periodic testing program was conducted to address previously proposed corrective actions. [P.1(d)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that the design bases are correctly translated into specifications, drawings, procedures, and instructions. It also requires, 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 November 16, 2011, the licensee failed to establish a suitable testing program to check safety-related contactors and verify their ability to operate at design basis degraded voltage conditions. Specifically, the licensee failed to perform an initial baseline test and subsequent periodic test plan to demonstrate the ability of several motor starter contactors but more specifically the ESW Pump 'A' discharge valve 1P45F0130A motor starter contactor to operate at minimum pickup voltage. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR-2011-05505, CR-2011-05507, and CR-2011-06107, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000440/2011008-03, Failure to Test Safety-Related Contactors at Degraded Voltage Conditions).

(4) Failure to Report Unanalyzed Condition Related to Internal Flooding

Introduction: A Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.72(b)(3)(ii)(B), "Immediate Notification Requirements for Operating Nuclear Reactors," was identified by the inspectors for failure to report within eight hours an unanalyzed condition that significantly degrades plant safety. Specifically, licensee failed to notify NRC upon discovery that a postulated internal flood in the control complex could result in loss of single failure capability of safety-related equipment.

Description: As described earlier in Section 1R21.3b.(1) of this inspection report, the licensee failed to correctly analyze the consequences of a postulated moderate energy pipe crack of a SW pipe in the control complex and to provide adequate operating procedures to mitigate the event.

In response to the inspector's concerns, the licensee initiated corrective actions (CR-2011-05217) which included performing additional analyses, establishing compensatory measures, issuing procedure orders, and revising operating procedures. These corrective actions were required to ensure the operability of redundant safety-related systems located in the control complex building. Prior to the implementation of these measures, a postulated piping failure could have resulted in the loss of redundant safety-related equipment due to flooding. Specifically, as further explained in Section 1R21.3b.(1) of this inspection report, the following safety-related systems would

have lost their single failure capability: Emergency Closed Cooling (ECC) and Residual Heat Removal (RHR).

The licensee initiated condition report CR-2011-05217, on November 10, 2011, to address the internal flooding issue. The condition report identified the time of discovery as 13:55 on November 10, 2011. Based on the information available at that time, the licensee determined that the condition was not reportable.

The licensee issued a Prompt Functionality Assessment which determined safety-related equipment would be protected based on updated operating procedures which had been issued after the condition was discovered. The Prompt Functionality Assessment did not specifically address past operability or reportability, but it demonstrated that operator actions would be required to protect redundant safety-related equipment. The shift manager accepted the Prompt Functionality Assessment at 19:57 on November 22, 2011.

On November 28, 2011, the licensee issued a Reportability Review to address reportability in accordance with 10 CFR 50.73(a)(2)(ii)(B). The Reportability Review concluded that the condition was not reportable. This conclusion was based, in part, on the ability of the operators to take mitigating actions that were not identified in procedures and were not discussed in the UFSAR prior to discovery of this condition by the inspection team.

The inspectors noted this condition met the criteria of 50.72(b)(3)(ii)(B), and should have been reported to the NRC within eight hours of discovery. Specifically, by determining a single failure impacted the operability of two safety-related systems, the plant encountered an unanalyzed condition that significantly degraded plant safety. Based on analysis discussed in the Prompt Functionality Assessment, this condition should have been reported within eight hours of its acceptance at 19:57 on November 22, 2011. The licensee entered this issue into their corrective action program (CR-2011-06227 and CR-2011-06530) and reported the unanalyzed condition at 18:29 (EST) on December 7, 2011, as required by 10 CFR 50.72(b)(3)(ii)(b) in Event Notification (EN) 47508, “Postulated Flooding Scenario Results in Unanalyzed Condition.”
Analysis: The inspectors determined the failure to provide an eight-hour report of an unanalyzed condition that significantly degrades plant safety was contrary to 10 CFR Part 50.72(b)(3)(ii)(B) and was a performance deficiency. The performance deficiency was evaluated using IMC 0612, “Power Reactor Inspection Reports,” and was determined to be of minor significance. However, it was also determined to involve a traditional enforcement violation because it potentially impede or impact the regulatory process. Specifically, failure to notify NRC of an unanalyzed condition challenges the regulatory process because it prevents NRC from evaluating the need to expand the scope of inspection to include the circumstances surrounding the condition. The traditional enforcement violation was determined to be more than minor in accordance with the NRC Enforcement Policy because the information that was not reported to NRC had a material impact on safety and licensed activities.

The traditional enforcement violation was determined to be a Severity Level IV violation in accordance with Section 6.9 of the NRC Enforcement Policy. Specifically, the failure to notify NRC within eight hours of discovery of an unanalyzed condition that significantly

degraded plant safety did not result in an unacceptable change to the facility or procedures.

The inspectors determined an evaluation for cross-cutting aspect was not applicable because this is a traditional enforcement violation.

Enforcement: Title 10 CFR Part 50, Section 50.72(b)(3)(ii)(B), "Immediate Notification Requirements for Operating Nuclear Reactors," requires, in part, the licensee to provide an eight-hour report from the occurrence of an instance where the nuclear power plant is in an unanalyzed condition that significantly degrades plant safety.

Contrary to the above, as of November 22, 2011, the licensee failed to notify NRC within eight hours of a condition where the nuclear plant was in an unanalyzed condition that significantly degraded plant safety. Specifically, licensee failed to notify the NRC when they discovered that a postulated internal flood in the control complex could result in loss of single failure capability of safety-related equipment. In accordance with Section 6.9 of the NRC Enforcement Policy, this violation is classified as a Severity Level IV violation for failure to make a report required by 10 CFR 50.72. Because this violation was entered into the licensee's corrective action program as CR-2011-06227 and CR-2011-06530, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000440/2011008-04, Failure to Report Unanalyzed Condition Related to Internal Flooding).

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed 4 operating experience issues to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- GL 95-007 "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves";
- IN 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design";
- IN 2007-09, "Equipment Operability During Degraded Voltage Conditions"; and
- IR 2011-010, "Prairie Island Safety-Related Battery Charger – White Finding."

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The inspectors reviewed one permanent plant modification related to selected risk significant components to verify that the design bases, licensing bases, and performance capability of the component had not been degraded through modification. The modification listed below was reviewed as part of this inspection effort:

- ECP 11-0626-001, Temporary Installation of Davis-Besse Transformer for Perry Unit 1 Startup, Revision 3.

b. Findings

No findings of significance were identified.

.6 Operating Procedure Accident Scenario Reviews

a. Inspection Scope

The inspectors performed a detailed review of the procedure and procedure sections listed below. In plant operator actions were walked down with non-licensed operators to evaluate whether there was sufficient information to perform the procedure, the steps could reasonably performed in the available time, and the necessary tools and equipment were available. In addition, the procedures were reviewed to ensure that the procedure steps would accomplish the desired result.

- Section 7.12, “ESW Pump Strainer Manual Backwash on Loss of Power,” of procedure SOI-P45/P49, “Emergency Service Water and Screen Wash Systems,” Revision 19;

- Section 7.14, “ESW Screen Wash Strainer Manual Backwash on Loss of Power,” of procedure SOI-P45/P49, “Emergency Service Water and Screen Wash Systems,” Revision 19;

- ARI-H13-P970-0001, “Common Long Response Benchboard,” Revision 12; and

- EOP-SPI 7.1, “Preparation For Containment Venting,” Revision 3.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program (CAP)

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the CAP. 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, CAP documents 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 Meeting(s)

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Grabnar and other members of the licensee staff on December 2, 2011. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. None of the information reviewed by the inspectors was considered proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Grabnar, Plant General Manager
J. Tufts, Operations Manager
J. Oelbracht, Chemistry Manager
C. Elberfeld, Regulatory Compliance Supervisor
J. Pelcic, Regulatory Compliance Engineer

Nuclear Regulatory Commission
M. Marshfield, Senior Resident Inspector
T. Hartman, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000440/2011008-01	NCV	Failure to Adequately Protect Safety Related Equipment from Internal Flooding
05000440/2011008-02	NCV	Inadequate Control Circuit Voltage Calculation for Safety-Related Motor Starter Contactors
05000440/2011008-03	NCV	Failure to Test Safety-Related Contactors at Degraded Voltage Conditions
05000440/2011008-04	NCV	Failure to Report Unanalyzed Condition Related to Internal Flooding

LIST OF DOCUMENTS REVIEWED

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

1R21 Component Design Bases Inspection

CALCULATIONS

Number	Description or Title	Revision
PSTG-0006	Analyze the Temporary Replacement Transformer with Respect to its Impact on Fault Duties	3
AOV-14	Determination of Valve Thrust, Actuator Capability, and Margin for AOVs: 1C11F0011 & 1C11F0181	0
CL-M39-002	DT Across M39 Cooler in RCIC & RHR A & B Pump Rooms	0
CL-MOV-1E12-06	MOV 1E-F048A/B Max. DP	2/DCC-4
CL-MOV-1E12-2	MOV 1E-F064A/B/C Max. DP	1/DCC-1
CLMOV1P57-1	1P57 F015A/B and F020A/B Max. DP	0
CLMOV-3	Generic Letter 89-10 Program	5
DI-155	Emergency Closed Cooling Water (P42) System Heat Exchanger Size, Operating Temperature and Outlet Temperature and P42 System Operating Temperatures	5
DI-221	CEI PNPP Unit 1, ECC Heat Exchanger Design Input	1
E12-078	RHR-A/B Pump Performance Acceptance Criteria	3/DCC-2
E12-088	Residual Heat Removal System Hydraulic Calculation	1
E12-089	Required ESW Flow for the RHR HXs	3
E12-094	RHR/A Heat Exchanger Test Results	3
E12-1	NPSH – RHR System	0/DCC-5
E12-106	RHR Heat Exchanger “A” Loop Performance Test Evaluation	2
E12-53	Temperature Rise in LPCI Pumps Due to Increased Recirc. Valve Stroke Time	0/DCC-1
E12-C09	LPCI Pump Discharge Flow – Minimum Flow Control Valve Interlock	1
ECA-018	Environmental Conditions Analysis (DG-1)	4
JL-083	Flooding Analysis of CCB, IB, and FHB – Floor Elevation 574’-10”	2
MISC-0018	The effect of low voltage and high frequency on motor currents and breaker settings	0
MOV1E22-01	1E22F001, F004, F010, F011, F012, F015, and F023 Max dP Addendum A-01	2

CALCULATIONS

Number	Description or Title	Revision
MOVC-0043	Required Thrust Calc for Gate Motor Operated Valves	4/A-4
MOVC-0043	Required Thrust Calculation for Gate Motor Operated Valves (MOVs)	4
MOVC-0044	Required Thrust Calc for Globe Motor Operated Valves	4/A-2
MOVC-0063	Required Torque/Thrust Calc for Edward Globe Valves	1
MOVC-0073	AC MOV actuator degraded voltage torque/thrust capability using ComEd method	7
MOVC-0073	AC MOV Actuator Degraded Voltage/Torque/Thrust Capability Using Commonwealth Edison (ComEd) Method	7
MOVC-73	AC MOV Actuator Degraded Voltage Torque/Thrust Capability using Commonwealth Edison (Com Ed) Method	7
P42-050	Emergency Closed Cooling Heat Exchanger “A” Loop Performance Test Evaluation	1
P42-34	Reduced ESW Flow to the ECC Heat Exchanger	1
P42-39	Design Basis Heat Load & Required ESW Flow for the ECC HXs	2
P45-030	Perry Unit #1 P45 System Operating Temperatures	8
P45-056	ESW Pump Performance Acceptance Criteria for SVI-P45-T2001, -T2002, -T2003	0
P45-057	ESW System Thermal Hydraulic Model	3
P45-081	Evaluation of Net Positive Suction Head (NPSH) and Submergence Requirements for the Emergency Service Water (ESW) System Pumps	0
P45-56	ESW Pump Performance Acceptance Criteria	0
P45-T06	ESW Pump Discharge Strainer Delta P	1
P54-208	Response Calculation for the Carbon Dioxide System Heat Detectors in the Emergency Diesel Generator Rooms	2
P54-208	Response Calculation for the Carbon Dioxide System Heat Detectors in the Emergency Diesel Generator Rooms	1
P57-13	Required Air Volume and Leakage Acceptance Criteria for the Division 1 and 2 Safety-Related Instrument Air System	4
P57-T03	ADS A(B) Air Storage Tank Pressure Alarm Setpoints	0/DCC-1
PRDC-0012	Evaluation of DC Loads For A 4 Hours Design Basis Station Blackout (SBO) and Also For A 24 Hours Beyond Design Basis SBO Event	2
PRDC-0014	Division 1, 125 VDC System Load Evaluation, Voltage Drop, Battery/Battery Charger Sizing Calculation	3
PRDC-002	Unit 1 Divisions 1, 2, 3 125 VDC System Coordination	4
PRLV-0004	Class 1E – MCC Breaker Settings	3
PRLV-0004	Class 1E – MCC Breaker Settings	3
PRMV-0003	Emergency Service Water (1P45)	3
PRMV-005	RHR Pump Motor Protection	2
PRMV-0053	Protective Relay Setpoints and Auxiliary System Protection	2

CALCULATIONS

Number	Description or Title	Revision
PSTG-0001	PNPP Class 1E Power Distribution System Voltage Study	5
PSTG-0001	PNPP Class 1E Power Distribution System Voltage Study	5
PSTG-0014	Electrical Load Determination of Division 1, 2, and 3 Diesel Generators	2
PSTG-0030	Voltage Drop in Control Circuits of Safety-Related Motor Control Center Starters, NEMA Sizes 1, 2, 3, and 4 Starters	1
SQ-0035	Motor Operated Valves 1E12F064A/B/C, IE51F031, IE51F010, 1E12F0609/610	1/DCC-2
SQ-0041	MOV 1E12-F021, F003A/B and F048A/B	4
SQ-0062	Seismic and Weak Link Analysis for Valves 1P57-F0015A/15B/20A/20B	0

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
CR-2011-04850	Poor Housekeeping in the RHR “A” Heat Exchanger Room - Aux 599’	11/3/11
CR-2011-04864	NRC CDBI/URI Fire Protection Calculation Errors	11/03/11
CR-2011-05217	Internal Flooding Calculation Did Not Specify the Actions that Needed to be Taken	11/10/11
CR-2011-05484	NRC CDBI Identified Procedure Enhancements May Be Needed	11/16/11
CR-2011-05495	NRC CDBI – PSTG-0030 Omission of Resistive Values	11/16/11
CR-2011-05505	NRC CDBI – Test/Analysis Report 95-089 Appears to Provide Inconsistent Data	11/16/11
CR-2011-05507	NRC Identified CDBI – Repetitive Maintenance Plan Not Revised as Required	11/16/11
CR-2011-05510	NRC CDBI SOI-P45/49 Incorrectly States ESW Strainer Backwash Starts at 2.75 PSID	11/16/11
CR-2011-05577	Typo in Operations Night Order	11/17/11
CR-2011-05644	Consider Staging tools for ARI-H13-P970-0001 Window F8	11/18/11
CR-2011-06101	Door Closer Needs Adjustments	11/30/11
CR-2011-06107	CDBI 2011 – ESW ‘A’ Discharge Valve (1P45F0130A) uses Modified Acceptance Criteria When Determining Adequate Starting Voltage	11/30/11
CR-2011-06189	NRC CDBI – Repeat Failure of SCV Vent and Drain Valve Differential Times	12/01/11
CR-2011-06227	Reportability Requirements for Internal Flooding Calculation JL-083 Issues NCV	12/02/11
CR-2011-06530	8-Hour ENS Report Needed for Control Complex Flooding Calculation Issues	12/07/11

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
CR-G202-2008-39390	Pre-CDBI, NPSH Calculation P45-081 Does Not Account for EDG Frequency Variations	04/29/08
CR-G202-2008-39419	Recurring Problem: Maintain Constant HPCS ESW Keep Fill Pressure	04/29/08
CR-G202-2008-45376	Lubrication Contamination Present in Motor Lower Bearing Reservoir for ESW "A"	08/27/08
CR-G202-2009-56356	Set Screws Galled in Seal Collar on RHR A	03/31/09
09-56432	Leak Rate Valve 1E12F053A Manually Seated Invalidates Tes	04/01/09
CR-G202-2009-56476	Replacement RHR Motor Set In Place Sooner Than Needed Causing Additional Dose	04/01/09
CR-G202-2009-56493	Temporary Shielding Installed, Which Did Not Produce Any Dose Rate Reduction	04/02/09
CR-G202-2009-57118	RHR Pump A Excessive Minimum Flow Run Time	04/11/09
CR-2009-59397	SDV Vent and Drain Valve Differential Times are Slow	05/19/09
09-65972	Maintenance Rule Evaluation Form	10/15/09
10-79624	NRC Identified Concern for Pre-Conditioning Valve During Surveillance Testing	7/14/10
CR-2011-02505	Design Change from GE Does Not Correct the Initiating Problem	09/28/11
CR-2011-04572	Degraded Wiring Found In Start-Up Transformer Condulets	10/29/11
CR-2011-05180	1C11F010 and 1C11F011 Failed Initial Stroke Time Closed During SVI-C11-T2004	11/10/11

DRAWINGS

Number	Description or Title	Revision
206-0017-00000	One line diagram Class 1E 4.16 KV Bus EH 11 and EH12	EE
206-0021-00000	Single line diagram Class 1E 480 V Bus EF1A	P
206-0034-00000	One-Line diagram, Non-Class 1E 480 V bus F1B	N
206-0052-00000	One-Line diagram, Non-Class 1E DC system bus D1A and D1B	D
208-0006-00001	Elementary diagram 15 KV switchgear internals	B
208-0055-00017	Elementary diagram RHR pump C002A	X
208-0176-00004	A Emergency Service Water Pump Discharge Valve F130A	CC
302-0212-00000	Service Water System	CCC
302-0271-00000	Safety Related Instrument Air System	R
302-0621	Emergency Closed Cooling System	SS
302-0641-00000	Residual Heat Removal System	CCC
302-0642-00000	Residual Heat Removal System	HH
302-0643-00000	Residual Heat Removal System	ZZ

DRAWINGS

Number	Description or Title	Revision
302-0791	Emergency Service Water System	SS
302-0792	Emergency Service Water System	LL
302-871	Control Rod Drive Hydraulic System	EE
B-022-0010-00000	Environmental Conditions for Control Building	H
B-208-0055-00007	Residual Heat Removal System Relay Logic Bus A	FF
B-208-0055-00014	Residual Heat Removal System Testability (A)	P
B-208-0055-00038	Residual Heat Removal System RHR Pump Minimum Flow MOV F064A	S
B-208-0176-00001	"A" Emergency Service Water Pump C001A	DD
B-208-0176-00004	"A" Emergency Service Water Pump Discharge Valve F130A	CC
B-208-0206-00024	4.16 KV Bus EH11 Diesel Breaker EH1102	BB
B-208-0206-00053	4.16 KV Standby Diesel Brk. EH1102 Protective Relaying	CC
B-208-075	Elementary diagram RCIC Gland Seal Air Compressor Pump C004	U
B-208-199, Sh. 1	Elementary diagram Containment isolation valve 1P57-F015A	N
D-206-0023-00000	One Line Diagram Class 1E Bus EF1B	UUU
D-206-051	One line diagram Class 1E DC system	ZZ
S-235-199	MOV Data Sheet, 1P57-F015A, Sh. 1A	C
S-235-199	MOV Data Sheet, 1P57-F015A, Sh. 1B	A

MISCELLANEOUS

Number	Description or Title	Date or Revision
	ESW Pump A Vibration Data	10/24/11
	Operations Night Order – Control Complex Flooding Evaluation	1
	Motor Operated Valve Program Plan	7
09-65972	Maintenance Rule Evaluation Form	10/15/09
11-04904	Compensatory Measure for PFA 2011-05217 (50.59)	0
11-05217	Reportability Review	11/28/11
11-05217	Prompt Functionality Assessment	11/22/11
14291-1	PCP Inc. Instruction Manual for Battery Charger Model 35D-130-400	01/31/79
1E12	RHR System Health Report	2011-2
1E12F0048A-007	MOV Test Report	0
1E12F0064A-008	MOV Test Report	0
1P57	Safety Related Instrument Air System Health Report	2011-2

MISCELLANEOUS

Number	Description or Title	Date or Revision
1P57F0015A-002	MOV Test Report	0
1P57N0025A	Hi-Lo Pressure Alarm Unit for ADS Air Storage Tank A	0
47508	50.72 Notification – Postulated Flooding Scenario Results in Unanalyzed Condition	12/07/11
DIE No. 107	Impact of revision 5 of calculation PSTG-0001 on MOVs	0
DIR DV9325	MCC Bus Voltages for MOV Degraded Voltage Analysis	6
DIR DV9325	MCC Bus Voltages for MOV Degraded Voltage Analysis	7
IB3.2.1.7-1B	ABB installation and maintenance instructions, Metal clad medium voltage power switchgear	0
L02197	Response to Generic Letter 95-07	10/16/95
P.O. 55101449	ESW Pump A Motor Refurbishment	02/16/04
SAP Notification No. 600251310	NRC Information Notice 2005-30	0
SP-101-4508-00	General specifications for 13.2 and 4 KV motors supplied by GE	1
TAR-95-089	Cutler-Hammer Starter/Contact Coils, NEMA Size 2B1, 3A1, and 4	03/28/95
Vendor Manual 0360	ECC Heat Exchanger Manual	13

MODIFICATIONS

Number	Description or Title	Date or Revision
00-8075	Install Flood Barrier Angle at Door IB-103	0
11-0755-001	Place Sandbags Against Control Complex Doors IB-103	0
ECP 11-0626-001	Temporary Installation of Davis-Besse Transformer for Perry Unit 1 Startup	3

PROCEDURES

Number	Description or Title	Revision
	Operations Night Order; Control Complex Flooding Evaluation	1
ARI-H13-P601-0018	Leak Detection	13
ARI-H13-P601-0019-H8	ADS A Air Strg Tank Pressure HI/LO	14
ARI-H13-P870-0003	Circulating Water	8

PROCEDURES

Number	Description or Title	Revision
ARI-H13-P970-0001	Common Long Response Benchboard	12
ARI-H13-P970-0001	Common Long Response Benchboard	12, 13, 15
ARI-H51-P031-13	Floor Drain Sumps	4
EMARP-0008	Clam/Mussel Monitoring	7
EOP-SPI 7.1	Preparation For Containment Venting	3
EPI-A1	Emergency Action Levels	23
GEI-0006	General Maintenance of Motor Control Circuits	17
GEI-0135	ABB Power Circuit Breakers 5KV Types 5HK250 and 5HK350 Maintenance	10/6/10
PTI-E12-P0002	RHR Heat Exchangers A and C Performance Testing	8
PTI-P57-P0001	Loss of Air Test for Safety-Related Instrument Air System	8
SOI-E12	Residual Heat Removal System	55
SOI-E51, Section 7.4	RPV Level And Pressure Control using RCIC during SBO	29
SOI-P45/P49	Emergency Service Water And Screen Wash Systems	19
SVI-E12-T2001	RHR A Pump and Valve Operability Test	28
SVI-P57-T2200	Safety-Related Air 1P57-F555A And 1P57-F556A Leak Rate Test	2

WORK DOCUMENTS

Number	Description or Title	Date
200477087	Startup transformer and sec WNDG cable TDR test	10/19/11
1E22F004-006	MOV Test Report	10/29/09
200068834	Emergency Closed Cooling Water Heat Exchanger	04/06/05
200074607	CC Buck Chk MCC 1R24s0021 Disc Se EF1B07-CC with shadow 200104126 Relay Replacement	10/06/06
200144120	Emergency Closed Cooling Water Heat Exchanger	03/28/05
200238880	Relay IFC66K 50/51A Em Serv Wtr Pmp 1P	05/13/08
200238881	Relay IFC66K 50/51A Em Serv Wtr Pmp 1P	06/04/10
200238882	Relay IFC66K 50/51A Em Serv Wtr Pmp 1P	06/07/10
200238883	Relay IFC66K 50/51A Em Serv Wtr Pmp 1P	06/03/10
200280587	Replace Elastomers And Test	03/19/09
200290590	DG1 Meggar Generator	01/27/10
200290947	Clean and Inspect Generator Panel	02/03/10

WORK DOCUMENTS

Number	Description or Title	Date
200291629	CC Miscellaneous Maintenance of Division 1 Diesel Control Cabinet	02/04/10
200320226	Emergency Closed Cooling A Heat Exchanger Performance Testing	06/19/08
200342671	“New PM” Verify/Adjust K1 Relay	02/03/10
200362436	Scram Discharge Volume Vent and Drain Valves Operability Test	07/04/11
200363629	ESW Pump A and Valve Operability Test	07/10/11
200364187	Replace Elastomers And Test	05/18/11
200365540	(24M) Division 1 Standby Diesel Generator Functional Test	05/20/11
200365839	SVI-R43-T5366, LPCS/LPCI A Initiation and Loss of EH11 Response Time Test	04/18/11
200393373	Exercise and Service Breaker EH1102	10/06/10
200403955	(R/T) Thermography Div 1 EDG High Voltage Cabinet	01/11/11
200418483	SVI-R43-T1317, (31D) Diesel Generator Start and Load Division 1	06/09/11
200418734	RHR A Pump and Valve Operability Test	10/28/11
200442914	Scram Discharge Volume Vent and Drain Valves Operability Test	05/19/11
200455668	Scram Discharge Volume Vent and Drain Valves Operability Test	08/14/11
200455670	Scram Discharge Volume Vent and Drain Valves Operability Test	11/12/11
200456021	(31D) Division 1 Diesel Generator Starting Air SCV and FCV Operability	10/06/11
200456072	SVI-R43-T1317, (31D) Diesel Generator Start and Load Division 1	07/04/11
200482050	Air to Scram Discharge Vol Dr	11/17/11
960005554	Heat Exchanger 1P42B0001A Needs Inspected	10/03/97
PY-SVI-R42T5202	Unit 1, Division 1 Weekly 125 V Battery Voltage And Category A Limits Check	10/31/11
PY-SVI-R42T5219	Unit 1, Division 1 125 V Battery Category B Limits, Terminal Corrosion And Electrolyte Temperature Check	04/04/11

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AOV	Air Operated Valve
ARI	Alarm Response Instruction
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CARD	Corrective Action Review Documents
CCDP	Conditional Core Damage Probability
CDBI	Component Design Basis Inspection
CFR	Code of Federal Regulations
CR	Condition Report
DC	Direct Current
ECC	Emergency Closed Cooling
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EST	Eastern Standard Time
ESW	Essential Service Water
FENOC	First Energy Nuclear Operating Company
FSAR	Final Safety Analysis Report
GL	Generic Letters
HPCS	High Pressure Core Spray
HPCS	High Pressure Core Spray
IB	Intermediate Building
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notices
IPE	Individual Plant Examination
IST	Inservice Test
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
MOV	Motor Operated Valve
NCV	Non Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
PRA	Probabilistic Risk-Assessment
RASP	Risk-Assessment Standardization Project
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RIS	Regulatory Issue Summaries
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SBO	Station Blackout
SDP	Significance Determination Process
SOI	Significant Operating Instruction
SPAR	Standardized Plant Analysis Risk
SRA	Senior Risk Analyst

SUT	Startup Transformer
SW	Service Water
TAR	Test Analysis Report
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VAC	Voltage Alternate Current
VDC	Volts Direct Current

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Sincerely,
/RA/

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

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