

February 4, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville IL 60555

SUBJECT: CLINTON POWER STATION, NRC COMPONENT DESIGN BASES
INSPECTION REPORT (CDBI) 05000461/2007008(DRS)

Dear Mr. Pardee:

On December 19, 2007, the U. S. Nuclear Regulatory Commission (NRC) completed a biennial component design bases baseline inspection at your Clinton Power Station. The enclosed report documents the inspection findings, which were discussed during an initial exit meeting on November 16, 2007, and during the final exit telecon on December 19, 2007, with Mr. F. Kearney, and other members of your staff.

This 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 calculations design bases documents procedures and records, observed activities, and interviewed personnel. Specifically, this inspection focused on the design of components, that were risk significant and had low margin.

Based on the results of this inspection, five NRC-identified findings of very low safety significance were identified, all of which involved violations of NRC requirements. 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 (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest any finding or the subject or severity of any NCV in this report, 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 copies 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 NRC Resident Inspector Office at the Clinton Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and 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 (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at <http://www.nrc.gov/reading-rm/adams.html> (the Public Electronic Reading Room).

Sincerely,

/RA/

Julio Lara, Chief
Engineering Branch 3
Division of Reactor Safety

Docket No. 50-461
License No. NPF-62

Enclosure: Inspection Report 05000461/2007008(DRS)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
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Chairman, Illinois Commerce Commission
Illinois Emergency Management Agency

C. Pardee

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cc w/encl: Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
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Inspection Report to Mr. Charles G. Pardee from Mr. Julio F. Lara dated February 4, 2008.

SUBJECT: CLINTON POWER STATION, NRC COMPONENT DESIGN BASES
INSPECTION REPORT (CDBI) 05000461/2007008(DRS)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: 50-461
License No: NPF-62

Report No: 05000461/2007008(DRS)

Licensee: AmerGen Energy Company, LLC

Facility: Clinton Power Station

Location: Clinton IL, 61727

Dates: October 15 through November 16, 2007
December 19, 2007

Inspectors: Z. Falevits, Senior Engineering Inspector, Lead
M. Munir, Engineering Inspector
D. Tharp, Resident Inspector
G. O'Dwyer, Engineering Inspector
F. Baxter, Electrical Contractor
S. Spiegelman, Mechanical Contractor

Observer: J. Bozga, Engineering Inspector

Approved by: J.F. Lara, Chief
Engineering Branch 3
Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000461/2007008; 10/15/07 - 11/16/07; Clinton Power Station; Component Design Bases Inspection.

The inspection was a 3-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. Five findings of very low safety significance were identified, all with associated Non-Cited Violations (NCVs). 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 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. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving inadequate cable design. Specifically, the team identified that the licensee failed to incorporate appropriate licensing and design basis requirements reflecting worst case environmental conditions for power and control safety related cables. Incorporation of these requirements would have ensured that the cables were designed for the continuous submerged conditions that are experienced at Clinton. The issue was entered into the licensee's corrective action program to initiate a review of the current cable monitoring programs, and to initiate long-term corrective actions.

The issue was more than minor because if left uncorrected it could result in the loss of safety related and non-safety related power and control cables including cables providing offsite power to the safety related buses. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because of lack of evidence, so far, of cable degradation or of past failures of these and similar other energized cables. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because the licensee did not thoroughly evaluate problems such as the resolutions, address causes, and extent of condition (P.1 (c)). (Section 1R21.3.b.1)

- Green. The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving inadequate equipment design. Specifically, the Division 3, emergency diesel generator (EDG) neutral ground resistor was found to be in a non-ventilated enclosure contrary to the USAR, which called for a ventilated housing. The issue was entered into the licensee's corrective action program to address this non-conforming condition and develop a design change to enhance ventilation for the resistor.

The finding was more than minor because if left uncorrected the failure of the grounding resistor could adversely impact availability of the Division 3, high pressure core spray (HPCS) EDG. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because of the very low probability of the scenario involving a ground fault in conjunction with a loss of offsite power and a loss of coolant accident. The team determined that there was no cross-cutting aspect to this finding. (Section 1R21.3.b.2)

- Green. The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving inadequate design of the emergency diesel generator (EDG) exhaust sub-systems. Specifically, the licensee failed to properly account for severe weather in the design of the exhaust ducts for the EDGs. Consequently, during severe weather conditions, icing or glazing could potentially result in blockage of the exhaust ducts screens located at the duct outlet and in exceeding the backpressure requirements of the ducts. Once identified, the licensee initiated a prompt operability evaluation to verify system operability and an Issue Report which included appropriate compensatory actions.

The finding was more than minor because the licensee failed to properly account for external factors including severe weather conditions as discussed in the Clinton UFSAR. The issue was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because the licensee provided information to support the EDGs operating for short periods of time at elevated back pressures. The team determined that there was no cross-cutting aspect to this finding. (Section 1R21.3.b.3)

- Green. The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving a temporary installation that added lead shielding to the Unit 1 residual heat removal (RHR) piping. Specifically, the team identified numerous non-conservative technical errors and calculation omissions in seismic design basis analysis calculations that supported this temporary installation. Once identified, the licensee initiated a prompt operability evaluation to verify system operability and an Issue Report which included appropriate compensatory actions.

The finding was more than minor because the presence of these non-conservative calculational deficiencies resulted in the need to re-perform the seismic design basis analysis calculations and in the removal of temporary lead shielding to assure that pipe supports would function as required during the design basis seismic event. The finding screened as having very low significance because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone Column of the Phase 1 worksheet. After re-performing the calculations, the licensee was able to demonstrate that sufficient margin was available to support the loads that would be seen during the design basis seismic event. The cause of the finding is related to the cross-cutting element of Human Performance Resources, because the licensee did not provide complete, accurate and up-to-date design documentation to assure nuclear safety (H.2(c)). Specifically, the licensee had the temporary installation of lead shielding in

place since 2002 and did not formally update the associated pipe support calculations in a timely manner. (Section 1R21.3.b.4)

- Green. The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance, in that, the shutdown service water (SX) pump tests conducted did not appropriately demonstrate that the SX pumps met design basis accident requirements. Specifically, the pump test acceptance criteria allowed the pump performance to degrade below the performance assumed by the design analysis. Once identified, this finding was entered into the licensee's corrective action program and the licensee completed an evaluation and retesting that demonstrated the pumps' capacity to perform required safety functions.

This issue was more than minor because the test allowed the "A" SX pump to degrade below the analyzed design limit and required a prompt operability assessment to demonstrate that the pump performance was above the reanalyzed limits. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because on re-evaluation, the design safety function was found to have been maintained. The team determined that there was no cross-cutting aspect to this finding. (Section 1R21.3.b.5)

B. Licensee-Identified Violations

None

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 team selected risk significant components and operator actions for review using information contained in the licensee's PRA and the Clinton Standardized Plant Analysis Risk (SPAR) Model, Revision 3.31. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 2.0. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios.

The team 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 modification, or 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 action, 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.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), 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) 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 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 was consistent with the design bases and was appropriate may include 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, system health reports, operating experience-related information and licensee corrective action program 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 19 components (inspection samples) were reviewed:

1. Residual Heat Removal (RHR) Pump 1A (1E12 C002A): The team reviewed piping and instrumentation diagrams, pump line up, pump capacities, surveillance procedures, test data and test results for the RHR pumps. Design calculations related to pump head, minimum required flow, and net positive suction head (NPSH) were reviewed to ensure the pumps were capable of providing their accident mitigation function during all required conditions.

The team also reviewed associated electrical calculations to confirm that the design basis minimum voltage at the motor terminals would be adequate for starting and running the motor under design basis conditions. The protective device relay settings were reviewed to ensure that adequate margin existed. For the RHR pump motor, the team assessed the bases for brake horsepower values used as design inputs to the licensee's electrical calculations. A review of the cable's ampacity was performed and evaluated to determine if adequate margin was available for all motor operating conditions. The team reviewed the breaker closure and opening control logic diagrams and the 125Vdc voltage calculations to ensure adequate voltage would be available for the control circuit components. Design change history was also reviewed to assess potential component degradation and impact

on design margins. The team also reviewed pipe stress design basis analysis and pipe support design basis calculations for the addition of temporary lead shielding.

2. RHR Pump 1A Minimum Flow Valve (1E12 F064A): The team reviewed the one-line and schematic diagrams. The team reviewed associated electrical calculations to confirm that the design basis minimum voltage at the motor terminals would be adequate for starting and running the motor under design basis conditions. The protective device/thermal overload relay settings were reviewed to ensure that adequate margin existed. A review of the cable's ampacity was performed and evaluated to determine if adequate margin was available for all motor operating conditions. The team reviewed the breaker closure and opening control logic diagrams and the control circuit voltage calculations to ensure adequate voltage would be available for the control circuit components. The team also reviewed pipe stress design basis analysis and pipe support design basis calculations for the addition of temporary lead shielding.
3. RHR HX 1A Shell Side Bypass Valve (1E12 F048A): The team reviewed the one-line and schematic diagrams. The team reviewed associated electrical calculations to confirm that the design basis minimum voltage at the motor terminals would be adequate for starting and running the motor under design basis conditions. The protective device/thermal overload relay settings were reviewed to ensure that adequate margin existed. A review of the cable's ampacity was performed and evaluated to determine if adequate margin was available for all motor operating conditions. The team reviewed the breaker closure and opening control logic diagrams and the control circuit voltage calculations to ensure adequate voltage would be available for the control circuit components. The team also reviewed pipe stress design basis analysis and pipe support design basis calculations for the addition of temporary lead shielding.
4. 4160 VAC Switchgear 1C1 (1E22 S004): The team reviewed the one-line diagrams and the switchgear purchase specification to verify acceptable equipment ratings. The bus loading calculations were reviewed to assure that worst case loads had been applied and that the switchgear rating had not been exceeded, and that the bus duct ratings were equal to the switchgear continuous rating as required. The short circuit and voltage calculations were reviewed to verify that they were representative of the worst case line-up of power sources and bus configurations, and that sufficient margin existed. The 5 kV cables used for distribution of power to and from the switchgear were reviewed to assure they were designed for the worst case service conditions. Breaker coordination plots were reviewed to assure selective breaker tripping. System grounding calculations were reviewed to assure optimum functionality of the grounding system. The switchgear maintenance program was also reviewed to ensure conformance with industry and vendor recommendations. A walkdown of the switchgear and diesel generator was performed to assess material condition.
5. 480 VAC Switchgear and Motor Control Center (MCC), (1E22 S002 and S003): The team reviewed the one-line diagrams and the purchase specification for the switchgear and the MCC to verify acceptable equipment ratings.

The team verified bus loading limits, voltage adequacy, short circuit capability, breaker coordination, and satisfactory operation of connected loads. The review included verifying ac voltage calculations to assure satisfactory voltage to the bus under worst case conditions, verifying that bus loading did not exceed bus rating, and reviewing short circuit calculations to verify that a condition did not exist that could result in exceeding the switchgear and breaker ratings. The team also reviewed the 4160/600 V transformers for adequate sizing, ventilation, and correct tap position. The team reviewed the breaker test program, the maintenance program and maintenance history. The team reviewed modification packages to replace the entire MCC buckets with equivalent buckets, and to replace specific internal relays with equivalent relays. Molded case breaker sizing was verified to ensure adequacy to meet the requirements of Regulatory Guide 1.106.

6. 125 VDC Division 1 Bus, MCC 1A (1DC13E): The team reviewed short circuit calculation, electrical coordination calculation, voltage drop calculation, fuse sizing criteria and basis, fuse control program, maintenance procedure. The team also performed a walkdown of a few selected panels to verify the fuse types and ratings against the fuse specification and the drawings.
7. 125 VDC Division 1 Battery 1A (1DC01E): The team reviewed electrical calculations including battery sizing, dutycycle, voltage drop calculations, short-circuit fault current calculation, breaker interrupting ratings and electrical coordination, battery float and equalizing voltages. In addition, the voltage drop calculations for safety-related dc loads and dc control power to 4160V and 480V switchgear were evaluated to verify that adequate voltage was available at these loads during a design bases event with loss of offsite power and for a station blackout event. The team verified minimum and maximum battery room temperatures and hydrogen buildup calculations for consistency with design basis requirements. The team reviewed the 125Vdc ground detection system including the ground sensitivity and basis for alarms and action levels. The operating procedures for normal, abnormal, and emergency conditions were reviewed. The team also reviewed the overall battery capacity, latest modified performance discharge test and service test, and quarterly battery surveillance tests required by technical specifications. The team performed a visual non-intrusive inspection of observable portions of the batteries to assess the installation configuration, material condition, and potential vulnerability to hazards.
8. 125 VDC Division 1 Battery Charger 1A (1DC06E): The team reviewed electrical calculations for the 125Vdc battery charger 1DC06E, including sizing calculation, contribution to short-circuit fault current, and breaker sizing. The operating procedures for normal, abnormal, and emergency conditions were reviewed. In addition, the test procedures were reviewed to determine if maintenance and testing activities for the battery chargers were in accordance with UFSAR requirements and vendor recommendations. The team performed a visual non-intrusive inspection of the battery chargers to assess the installation configuration, material condition, and potential vulnerability to hazards.

9. 4160 VAC Switchgear 1A1 (1AP07E); 480V Unit Sub Station A (0AP05E); 480V MCC 1C (1AP31E) and 1C1 (1AP78E): The team reviewed the one-line diagrams and the switchgear purchase specification to verify acceptable equipment ratings. The bus loading calculations were reviewed to assure that worst case loads had been applied and that the switchgear rating had not been exceeded, and that the bus duct ratings were equal to the switchgear continuous rating as required. The short circuit and voltage calculations were reviewed to verify that they were representative of the worst case line-up of power sources and bus configurations, and that sufficient margin existed. The 5 kV cables used for distribution of power to and from the switchgear were reviewed to assure they were designed for the worst case service conditions. Breaker coordination plots were reviewed to assure selective breaker tripping. System grounding calculations were reviewed to assure optimum functionality of the grounding system. The switchgear maintenance program was also reviewed to ensure conformance with industry and vendor recommendations.

The team also reviewed the 4160/600 V transformer for adequate sizing, ventilation, and correct tap position. The team reviewed the breaker test program, and the maintenance program and history. A walkdown was also performed to determine the material condition of the switchgear. Modification packages to replace the entire MCC buckets with equivalent buckets, and to replace specific internal relays with equivalent relays were reviewed. MCC molded case breaker sizing was verified to ensure adequacy to satisfy the requirements of Regulatory Guide 1.106.

10. Emergency Diesel Generator (EDG) 1A (1DG01KA): The team evaluated the EDG to determine if it can meet its design basis functions. Included in this inspection was the evaluation of the lube oil, inlet air, air starting, and exhaust air sub-systems. The fuel oil pump and fuel supply subsystem were reviewed. The team performed a walkdown of the EDG, observed a portion of a 24 hour surveillance run of the diesel generator and performed a walkdown of the exhaust silencer and ducting. Included in the review were structural calculations of the lube oil tubing, air supply receiver volume, pressure drop for the air exhaust system for normal and abnormal conditions, and jacket water/heat exchanger heat transfer calculations. An interview was conducted with the systems engineer regarding EDG performance and the a corrective action related to EDG governor speed control. Maintenance records and records of EDG In-service Testing (IST) for the past 3 years were reviewed.

The team also reviewed the EDG purchase specification. The EDG loading calculation was reviewed to assure that worst case loading had been considered, and that process controlled loads, load increase due to over frequency conditions had been considered. The EDG's response to a degraded grid voltage signal on the bus was reviewed. Periodic surveillance tests were reviewed to assure that test results were in compliance with TS requirements. The alarm response procedure used for EDG ground fault annunciation was reviewed to ensure adequate operator response in the event of a ground fault. Diesel fuel oil consumption and storage calculations were reviewed to verify that they accounted for the use of ultra low sulfur fuel. A walkdown of the diesel generator was performed to assess the material condition.

11. EDG 1A, Fuel Oil Transfer Pump (1DO01PA): The team reviewed the fuel oil transfer pump to verify that it would provide oil to the emergency diesel generators during EDG operation. The team reviewed the oil supply tank and piping to assure that an adequate supply of oil was available and that the Pump NPSH requirements were satisfied. The discharge piping to the day tank and the day tank were also reviewed. Associated hydraulic calculations and test results were reviewed and a walkdown of the equipment was performed during pump and EDG operation. Maintenance records were reviewed for the EDG oil supply and Corrective actions were reviewed. The oil volume calculation was compared to the oil consumption in two 24 hour diesel runs to assure that the calculated oil supply would meet design requirements. The systems engineer was interviewed to discuss recent system performance and discuss results from past EDG tests.
12. Reserve Auxiliary Transformer (RAT) and Switcher 4538. The team reviewed the three-winding RAT transformer nameplate, sizing, current carrying capability, and output voltage under worst-case conditions. The transformer tap position was also verified to ensure consistency with the tap position assumed in the voltage drop and short circuit calculations. The bus ducts emanating from the transformer were reviewed to ensure that the bus duct ratings were equal to the switchgear continuous rating as required. A walkdown of the RAT transformer was performed.
13. 345KV Circuit Switcher 4522: The team reviewed the history of newly installed switchyard breaker 4522, which connects the 345 kV North, bus to line 4522. The team noted that earlier failures of this new breaker had raised questions concerning its reliability, but the team determined that subsequent modifications assured present day reliability commensurate with the other four 345 kV circuit breakers. A walkdown of the switchyard and relay house was performed.
14. Nuclear System Protection System (NSPS) Invertors 1A thru 1D (1C71S001A-D): The team reviewed surveillance testing; short circuit calculations for sizing inverter bus; maintenance records and vendor requirements.
15. Shutdown Service Water (SX) Pump 1A (1SX01PA): The team reviewed analyses, operating procedures, test procedures, and test results associated with operation of the SX pump 1A. The evaluation considered both test and accident conditions. The analyses included hydraulic performance, NPSH, and minimum flow to the pumps. The team reviewed completed tests to ensure that design basis requirements were correctly translated into test acceptance criteria and that the tests demonstrated the pump's capacity to perform its design basis required functions. Design change history and IST results were reviewed to assess potential component degradation and impact on performance margins. The team also reviewed periodic inspection program to ensure strainers were maintained in a clean condition. The team reviewed operating and maintenance procedures associated with the service water strainers. In addition, the team reviewed the IRs issued to document SX pump problems.

The team reviewed the bus and diesel generator loading calculations, the breaker coordination plots, and the voltage and short circuit calculations to assure that the SX pump would perform its function as required. The team reviewed anchorage calculations for the pump for the plant design basis seismic loads. The team assessed water hammer analysis for Generic Letter 89-10 commitments and action taken to satisfy these commitments. The team reviewed flow balance data with resized piping for affects on pipe analysis and pipe support loads.

16. SX Strainer 1A (1SX01FA): The team reviewed anchorage calculations for the strainer for the plant design basis seismic loads. The team reviewed the preventive maintenance tasks, corrective maintenance history, problem history, and operating history of SX strainers to ensure that they were capable of performing their required functions under required condition. The team also reviewed procedures, surveillances, corrective actions, surveillance results, trending data and differential pressure and debris loading calculations to ensure the strainers were capable of performing their required functions under required conditions. The team reviewed strainer design requirements to ensure debris loading assumptions were consistent with industry guidance.
17. RHR Hx Inlet Valve (1E12 F014A): The team reviewed the one-line and schematic diagrams and the associated electrical calculations to confirm that the design basis minimum voltage at the motor terminals would be adequate for starting and running the motor under design basis conditions. The protective device/thermal overload relay settings were reviewed to ensure that adequate margin existed. A review of the cable's ampacity was performed and evaluated to determine if adequate margin was available for all motor operating conditions. The team reviewed the breaker closure and opening control logic diagrams and the control circuit voltage calculations to ensure adequate voltage would be available for the control circuit components.
18. Reactor Core Isolation Cooling (RCIC) Pump (1E51 C001): The RCIC pump and turbine were reviewed to assure that it could meet its safety function of supply of water to the reactor. Hydraulic calculations were reviewed to assure that the flow requirements were met and that sufficient NPSH was available from both the RCIC tank and the suppression pool. The water supply was further examined to assure that a reliable water supply was available and that transfer from the RCIC tank to the suppression pool could be accomplished without pump damage and within acceptable transfer times. An interview was conducted with the systems engineer to discuss recent testing, parts replacement and maintenance history. The requirements from both the UFSAR and TSs were reviewed to assure that the design conformed to the licensing commitments.
19. RCIC Steam Supply Turbine Governing Valve (1E51 F610): The team reviewed the MCC molded case breaker utilized for the valve motor to verify adequate sizing to satisfy the requirements of Regulatory Guide 1.106.

b. Findings

1. Continuously Submerged Cables Design Deficiency

Introduction: The team identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green), involving inadequate cable design. Specifically, the team identified that the licensee failed to incorporate appropriate licensing and design basis requirements reflecting worst case environmental conditions for power and control safety related cables. Incorporation of these requirements would ensure that the cables were designed for the continuous submerged conditions that are experienced.

Description: Safety related and non-safety related power and control cables are submerged in water on a continuous basis, often with no physical means of evacuating the water. The affected cables included cables from the Emergency Reserve Auxiliary Transformer (ERAT) transformer carrying offsite power to the plant Class 1E buses, cables to the SX pumps, and cables to the water system pumps. The team reviewed the specifications used to purchase these cables and noted that both safety-related and non-safety related cables were specified and purchased to the same specification. The normal environmental conditions in the specification were listed as 90 percent maximum, 50 percent average, and 5 percent minimum relative humidity. A review of the licensee's underground cable duct drawings showed that some of the cables would be continuously submerged in water because there were no procedures in place to routinely evacuate the water from manholes, and some cable duct banks sloped downwards and away from the manholes thus ensuring that cables would be continuously submerged. The licensee failed to ensure that the cables were designed for the anticipated environmental conditions by not thoroughly evaluating the submerged cables conditions that were identified in IR 00114481 and IR 00633239, issued July 5, 2002 and May 24, 2007, respectively.

USAR Sections 8.1.6.1.24, 8.1.6.2.4, and 8.3.1.4.4.2 stated that the cables would be qualified for the worst case design basis event. The physical configuration of the cable duct banks and manholes determined that the worst case service conditions for the cables would be submergence on a continuous basis; however, these USAR design requirements were not applied when specifying and procuring cable. Additionally, IEEE Standard 323-1971 "IEEE Standard for Qualifying Class 1E Equipment," requires that the service conditions for Class 1E equipment include environmental loading expected as a result of normal and abnormal operating environments throughout the installed life of equipment. These IEEE requirements were also not met for the above cables even though USAR Section 8.1.6.2.4 indicated that the design of the Clinton Power Station was in accordance with the requirements of IEEE 323.

Analysis: The team determined that failure to ensure that the installed cables were designed for the worst case anticipated environmental conditions was a performance deficiency warranting a significance evaluation. The team determined the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability,

and capability of safety-related power systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure of cables could adversely impact availability of power to redundant equipment as well as affect the availability of offsite power to safety related equipment in all three safety divisions.

The team evaluated this finding using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," SDP Phase 1 screening. The inspectors answered "NO" to all screening questions in the Mitigating Systems Cornerstone column because the licensee's evaluation in IR 00692997, dated November 1, 2007, stated that although the cables were not designed for continuous submergence duty, based on lack of evidence of cable degradation or of past failures, including that of similar energized cables, the condition was acceptable. Therefore, the finding screened as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because the licensee did not thoroughly evaluate problems such as the resolutions, address causes, and extent of condition (P.1 (c)). Specifically, the licensee failed to ensure that the cables were designed for the anticipated environmental conditions by not thoroughly evaluating the submerged cables conditions that were identified in IR 00114481 and IR 00633239, issued July 5, 2002 and May 24, 2007, respectively. In the former IR, the Plant Health Committee was presented with a recommendation to not implement corrective actions regarding the submerged cable issue. In addition, in the response to GL 2007-1, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," dated February 7, 2007 the licensee made a commitment to monitor inaccessible underground power cables using Exelon procedure ER-AA-3003; "Cable Condition Monitoring Program," Revision 0; however, the licensee informed the team that there were no cables at Clinton Power Station that would be subjected to the monitoring program requirements of ER-AA-3003, and furthermore, there were no maintenance or surveillance requirements in place to specifically inspect the cables, or to evacuate water from accessible submerged areas. In addition, the licensee did not effectively incorporate pertinent industry experience from NRC IN 2002-12, "Submerged Safety Related Electric Cables" into their evaluations and decisions, and erroneously concluded that the 4 kV cables were environmentally qualified to meet IEEE 323, when in fact the cables were neither specified nor qualified for their abnormal worst case operating environments, that is, submergence duty.

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 performance of design reviews, by the use of alternate simplified calculational methods, or by performance of a suitable testing program.

Contrary to the above, as of November 16, 2007, the licensee's design control measures failed to ensure the adequacy of the power and control cables that are submerged on a continuous basis. Specifically, the as-built design failed to meet USAR Sections 8.1.6.1.24, 8.1.6.2.4, 8.3.1.4.4.2, and those of IEEE Standard 323. The licensee entered the finding into their corrective action program as IR 00692997 to initiate a review of the current cable monitoring programs given their recognized service conditions, and to initiate actions over the long term. Because this finding was of very low safety significance, and it was entered into the licensee's corrective action program, this

violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2007008-01).

2. Division 3 - Emergency Diesel Generator (EDG) Neutral Ground Resistor Design Inadequacy

Introduction: The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green), involving inadequate equipment design. Specifically, the Division 3 EDG neutral ground resistor was found to be in a non-ventilated enclosure contrary to the USAR, which called for a ventilated housing.

Description: The Division 3 EDG incorporates a high resistance grounding system in accordance with USAR Section 8.3.1.1.2. The high resistance grounding permits continuous operation of the diesel generator in the presence of a ground fault on the system. On experiencing a ground fault, a current of approximately 30 Amperes flows through the resistor producing approximately 4 Kilowatts of heat. In accordance with GE NEDO-10905 (reference USAR 8.3.1.1.2.1) the grounding apparatus was specified to be separately mounted and enclosed in a ventilated steel housing. However, the team identified that the grounding transformer and resistor were neither separate nor in a ventilated housing, but in a totally enclosed panel within the non-ventilated Division 3 EDG control panel. The lack of ventilation or other means of heat dissipation from the transformer and resistor could result in an unacceptable increase of temperature within the transformer and resistor panel as well as the EDG 3 Control Panel. Any consequential equipment failures, or smoke from the overheated painted panel, could result in the failure of the Division 3 EDG.

To address this non-conforming condition and design deficiency, the licensee issued IR 00697048, IR 00702527, and Operability Evaluation 00697048-02, and also proposed a long term corrective action to develop a design change to enhance ventilation for the resistor.

Analysis: The team determined that failure to ensure that the installed grounding transformer and resistor enclosures were in accordance with the design basis was a performance deficiency warranting a significance evaluation. The team determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety-related power systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure of the grounding resistor could adversely impact availability of the Division 3 EDG.

The team evaluated this finding using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," SDP Phase 1 screening. The inspectors answered "NO" to all screening questions in the Mitigating Systems Cornerstone column because the failure to ensure that the Division 3 EDG neutral ground resistor was in a ventilated enclosure did not impact current operability of the EDG. Therefore, the finding screened as having very low safety significance (Green).

The team determined that there was no cross-cutting aspect to this finding.

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 performance of design reviews, by the use of alternate simplified calculational methods, or by performance of a suitable testing program.

Contrary to the above, prior to November 16, 2007, the licensee's design control measures failed to ensure the adequacy of the EDG 3 neutral grounding resistor housing. Specifically, the as-built design failed to meet USAR Section 8.3.1.1.2.1. However, because this violation was of very low safety significance, and it was entered into the licensee's corrective action program (IR 00697048), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2007008-02).

3. Inadequate Design of Emergency Diesel Generator Exhaust Sub-systems

Introduction: The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the EDG exhaust sub-systems. Specifically, the design of the system did not account for severe weather conditions that could result in blockage of the EDG exhaust ducts by icing the screen that are located at the outlet of the duct. The blockage could result in a backpressure that exceeds the design allowable value potentially impacting operability of the EDGs.

Description: The team reviewed the EDG exhaust system calculation and identified that the maximum back pressure of five inches of water would be exceeded when complete blockage of the primary exhaust outlet of all three EDGs occurs during severe weather conditions due to icing or glazing, and when partial blockage of the exhaust screens occurs during external events conditions (tornados). The EDG ducting has two outlets; a smaller outlet located upstream of an exhaust silencer and a larger outlet located downstream of the silencer. The function of the smaller opening is to limit the backpressure to five inches of water in the event that partial blockage occurs from external events. The analysis calculated the backpressure to be 45 inches of water if complete blockage of the silencer or larger outlet occurs.

During a walkdown of the EDGs, on November 1, 2007, the team noted that the end of the exhaust ducts were oriented upward and were potentially subject to icing and snow blockage. The ducting had been modified during original plant construction to include a screen to prevent debris from entering the duct. The screen was specified to be non-safety grade although it's obstruction could impact the operation of a safety grade system. Also, there was no documented evidence that icing of the screen had been evaluated by the licensee.

The team noted that the exhaust ducts of the three EDGs had similar designs and were therefore subject to a potential common mode failure. During the inspection, documented evidence was not available for review to confirm that the EDGs could start or operate at rated power with the elevated back pressure.

In response to this finding, the licensee performed an operability evaluation and determined that the EDGs were operable due to the fair weather conditions at the time

of this inspection. The licensee also implemented compensatory actions to remove the screen in the event of severe weather that could result in icing or glazing. In addition, a design change was initiated to modify the exhaust ducts design to prevent future blockage. The team determined that severe weather conditions that can lead to icing or blockage of the screens can occur every 2-3 years as discussed in Chapter 2 of the UFSAR. The team reviewed weather conditions for the past 20 years and noted that the weather conditions in the area of the Clinton plant were consistent with the conditions documented in the UFSAR.

Clinton also identified an EDG of similar design, located at another nuclear facility that was started with a back pressure higher than 45 inches of water. A test report for this test was provided to the team for review. The test was performed to demonstrate the function of a relief disc, as such was only at the higher pressure for a brief period. Clinton also provided the team with results of another test, performed by a vendor which demonstrated that the EDG was capable of running for 5-6 minutes at pressures greater than 45 inches of water. The diesel generator operation time at the higher back pressure was not identified and the report was therefore not conclusive regarding the ability for the EDG to operate for a sufficient time to melt the ice on the exhaust screen. Nonetheless, the team did not have further questions regarding the EDG operability.

To address this non-conforming condition and design deficiency, the licensee issued IR 00695303 to document and correct this design deficiency.

Analysis: The team determined that the failure to properly account for external factors including severe weather conditions was a performance deficiency warranting a significance evaluation. The team determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety-related power systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to account for severe weather conditions described in Chapter 2 of the UFSAR in the design of the EDG exhaust system. This condition, if left uncorrected, could affect the EDG operability.

The team evaluated this finding using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," SDP Phase 1 screening. The inspectors answered "NO" to all screening questions in the Mitigating Systems Cornerstone column because re-evaluation of this issue confirmed the operability of the system. Therefore, the finding screened as having very low safety significance (Green). The team determined that there was no cross-cutting aspect to this finding.

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 performance of design reviews, by the use of alternate simplified calculational methods, or by performance of a suitable testing program.

Contrary to the above, prior to November 2007, the licensee failed to establish effective measures to ensure that the design basis for the EDG exhaust system considered

severe weather conditions in the design of the exhaust ducting. The failure to ensure that the design basis considered severe weather conditions could have affected operability of all three EDGs when needed for an event. However, because this violation was of very low safety significance, and the issue was entered into the licensee's corrective action program (IR 00695303), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2007008-03).

4. Residual Heat Removal (RHR) Pipe Support Calculation Deficiencies

Introduction: The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the pipe support analysis performed for the RHR piping system. Specifically, the team identified numerous design basis calculational omissions and non-conservative technical errors associated with calculation IP-M-0704, Revision 0, which analyzed the RHR subsystem 1RH09 piping for design basis loading conditions with the installation of lead shielding.

Description: The team reviewed calculations IP-M-0704, "Evaluation subsystem 1RH-09 due to installation of temporary shielding per TSP 2001-110, 2001-120 and 2001-121" Revision 0, and associated RHR pipe support calculations. Calculation IP-M-0704 analyzed the RHR subsystem 1RH09 piping for the addition of temporary lead shielding to ensure that piping and supports would be able to withstand a design basis earthquake. This calculation provided the basis for the installation of the temporary lead shielding. The acceptance criteria of this temporary installation stated "All penetration, anchors and support load shall remain within their design load or shall be qualified for their acceptance." The team identified that the temporary installation, as implemented, failed to demonstrate Code compliance for the RHR pipe supports analyzed for design basis seismic loads and all other design basis loads.

During the review of the calculations associated with this temporary installation, the team identified numerous errors and omissions in the design calculations. These errors included two codes of record for a single base plate evaluation, non-conservative load increase factor used for anchor bolt design, non-conservative computation error of design forces and moments, calculational omission of proper reduction in stress cone for anchor bolt evaluation and unsubstantiated support acceptability for an overstress condition. The team also found an instance where a pipe hanger embedment plate was not analyzed for the induced bending moment due to pipe hanger angularity in combination with the applied straight pull out force. The pipe hanger embedment plate was qualified using a standard embedment plate design criteria which is based only on a straight pull out force. As a result of the pipe hanger embedment plate stresses exceeding their design basis and operability allowables, the RHR B LPCI, Containment Spray, feedwater leakage control, suppression pool cooling and shutdown cooling subsystems were declared inoperable during the inspection. Subsequently, Temporary Shielding Package 2001-121 was removed and the operability was restored. In response to these issues, the licensee initiated IRs 695282, 695925, 695250, 698813, and 696851.

Subsequent engineering justification and calculation performed in these IRs provided reasonable assurance that these errors did not result in an operability concern.

However, even though the licensee was able to provide this reasonable assurance for operability, the issues provided multiple examples of inadequate design basis calculations supporting this temporary installation.

Analysis: The team determined that the numerous design basis calculational omissions and non-conservative technical errors associated with calculation IP-M-0704 were a performance deficiency warranting a significance evaluation. The team determined that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because it was associated with the Mitigating systems attribute of design control, which affected the Mitigating System Cornerstone objective of ensuring the availability, reliability, and capability of the RHR system during a Seismic Class I design basis event. Specifically, numerous non-conservative technical errors and calculation omissions resulted in the need to re-perform seismic design basis analysis calculations and removal of the temporary lead shielding to assure that the piping supports for the aforementioned RHR piping would function as required during a design basis seismic event.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the team answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after re-performing the calculations and removing temporary lead shielding for the supports that were called into question by the inspection team, the licensee was able to show that enough design margin was still available to support the loads that would be seen during a design basis seismic event.

The cause of the finding is related to the cross-cutting element of Human Performance, Resources, because the licensee did not maintain complete, accurate and up-to-date design documentation to assure nuclear safety (H.2(c)). Specifically, the licensee had the temporary installation of lead shielding in place since 2002 and did not formally update the associated pipe support calculations in a timely manner. Further, a preliminary calculation was performed for the pipe support calculations in 2002. The preliminary calculation failed to identify numerous non-conservative technical errors and calculational omissions.

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

Contrary to the above, prior to November 13, 2007, the licensee design control measures failed to verify adequacy of design of RHR pipe supports, in that the methodology and design input did not account for as-built conditions (pipe hanger angularity and stress cone reduction for anchor bolt evaluation) and did not apply an appropriate load increase factor to an anchor bolt evaluation, which affected RHR pipe support design basis calculations. Specifically, calculation IP-M-0704 which analyzed RHR subsystem 1RH09 piping for design basis loading conditions with the addition of lead shielding contained numerous non-conservative calculational assumptions and omissions affecting the design basis for seismic analysis. However, because this

violation was of very low safety significance and because the issue was entered into the licensee's corrective action program (IR 695250 and IR 698813), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000461/2007008-04).

5. Inappropriate Shutdown Service Water (SX) Pump Test Acceptance Criteria

Introduction: The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance (Green), in that, the A and B SX pumps tests did not have appropriate acceptance criteria. Specifically, the tests failed to have appropriate minimum pump hydraulic operability limits to ensure the pumps' minimum design basis requirements were met.

Description: On November 13, 2007, the team requested the design calculation that determined the acceptance criteria for the "A" SX pump operability test. No calculation could be found for determining that the test acceptance criteria for all three SX pumps would ensure that the pumps' allowed test performance would be greater than the minimum design basis accident analysis requirements assumed in design calculation IP-M-0486, "Shutdown Service Water (SX) System Hydraulic Network Analysis Model and Flow Balance Acceptance Criteria," Revision 6, dated January 1, 2003.

Based on the team's questions, the licensee identified that: (1) the "A" and "B" SX pumps tests' acceptance criteria had been changed in 2003 without verifying that the new minimum required pump performance allowed by the test acceptance criteria was more than the minimum pump performance required by the design limits; (2) the result of the October 3, 2007, performance (WO 01044766) of the A SX pump test was more than the test acceptance criteria but less than the minimum pump performance assumed by the design calculation; (3) the test acceptance criteria had only been based on the section XI IST limits for the pumps and did not consider the minimum required performance assumed in the design calculation; (4) the A and B pump re-baselining was performed in accordance with Procedure ER-AA-321, Administrative Requirements for In-service Testing and this procedure and process did not direct personnel to search for impacted calculations. The personnel re-baselining the pump were not aware that IP-M-0486 was impacted and therefore did not request an update to the calculation prior to revising the test procedure. The licensee initiated IR 700713 to document and address these deficiencies. The licensee performed a prompt operability assessment (OE 700713-02, Revision 0) which determined that the SX pumps were able to perform their safety function because the lake was at 51 degrees F vice the design basis maximum of 95 F. The OE used an equation which had been derived in design calculation IP-M-0486, Revision 6.

On November 28, 2007, the team determined that the methodology used in design calculation IP-M-0486, Revision 6, to derive the equation used in the OE was incorrect. The equation derived in design calculation IP-M-0486, Revision 6, was then used to incorrectly model potential pump degradation and calculate the resulting IST limits for the A and B SX pumps. The preparer of design calculation IP-M-0486, Revision 6 had incorrectly inserted (without a correct technical basis) a term (denoted "L" to represent internal leakage in a pump) into a curve-fitting equation. The preparer then incorrectly used the curve-fitting equation as a theoretical hydraulic performance equation and

solved the equation for "L". The leakage values predicted by the incorrect equation were not realistic pump degradation estimates, e.g., (1) at shutoff head the equation incorrectly predicted that the leakage in the pump would be zero and (2) the equation incorrectly predicted that the higher the pump differential pressure, the lower the internal leakage would be. The licensee issued IR 706104 to initiate actions to correct the methodology used in the OE and the design calculation, and retested the "A" SX pump using corrected acceptance criteria. The licensee revised the OE to use corrected methodology and test acceptance criteria and determined that the SX pumps were operable under all design conditions. The team reviewed the evaluation and did not identify any deficiencies with the licensee's conclusion.

Analysis: The team determined that failure to establish appropriate test acceptance criteria was a performance deficiency warranting a significance evaluation. The team determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the procedure quality attribute of the Mitigating System cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of the SX system to respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not appropriately ensure that test acceptance criteria for the SX pump demonstrated that the SX pumps would be capable of providing the required design basis flow during accident conditions.

The team evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening. The team answered "No" to all the screening questions because the SX pumps remained operable based on re-evaluation and re-testing. Therefore, the finding screened as having very low safety significance (Green). The team concluded this finding did not have a cross-cutting aspect.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service and be performed in accordance with written test procedures that incorporate the requirements and acceptance limits contained in applicable design documents. The results shall be documented and evaluated to assure that test requirements have been satisfied.

Contrary to the above, from October 2003 to November 16, 2007, the licensee failed to incorporate appropriate SX system flow and pump differential pressure design requirements into the A and B SX pumps test acceptance criteria. Specifically, on October 3, 2007, the A SX pump passed the acceptance criteria but did not meet the performance assumed in the design calculation. Also, the A and B SX pumps test acceptance criteria from October 2003 to November 16, 2007, allowed the pumps to pass the test but not meet the performance assumed in the design calculation. However, because this issue was of very low safety significance, and it was entered into the licensee's corrective action program (IR 700713 and IR 706104), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2007008-05).

.4 Operating Experience

a. Inspection Scope

The team reviewed six operating experience issues (6 samples) 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:

- IN 2005 -04, Bus Single Failure – Crystal River;
- IN 2006-18, Significant Loss of Safety Related Electric Power at Forsmark Unit 1 in Sweden;
- AR 00456192 (IN 06-02), Galvanized Supports/Trays with Jacketed Cable;
- AR 00296788, Bus Single Failure - Crystal River - IN-2005-04;
- ATI 521373, OE 23020 -- Potential for RCIC Water Hammer; and
- ATI 179683, OPEX SME review of Airbound Containment Spray Pumps.

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The team reviewed seven permanent plant modifications related to selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modifications listed below were reviewed as part of this inspection effort:

- ECN 31580, Installation of Interposing Relay into Voltage Regulator Circuit;
- EC-330624, Class 1E MCC Molded Case Circuit Breaker Control Unit (Bucket) Replacement;
- ECN 32363, Install a Synchrocheck Rly 225-DG1KA to Provide DG Div 1 Breaker Closing Permissive When Paralleling;
- EC-350432, Replacement of Obsolete ITE/Gould J10 Control Relays Used in Safety and Non-Safety 5600 Series 480 V MCC Applications;
- EC 0000349235, Replace Division 1 Diesel Generator Governor Actuators;
- EC 359252, HPCS and RCIC Pump Suction Line Submersion; and
- ECN 30873, Modify Valve 1E12F064A to Eliminate Pressure Lock.

b. Findings

No findings of significance were identified.

.6 Risk Significant Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of five risk significant, time critical operator actions (five samples). These actions were selected from the licensee's PRA rankings of human action importance based on risk achievement worth 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 team performed a detailed review and walk through of associated procedures, including observing the performance of some actions in the station's simulator and in the plant for other actions, with an appropriate plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required.

The following operator actions were reviewed:

- Operator Fails to Initiate Automatic Depressurization System for Depressurization;
- Operator Fails to Shed 125V DC Non-Essential Loads;
- Operator Fails to Manually Initiate Suppression Pool Cooling and Manipulate Valves;
- Operator Fails to Vent the Containment; and
- Operator Fails to Line up Main Condenser Vacuum Pumps.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors routinely reviewed issues during inspection activities to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed.

Issues

During the review of the RCIC pump and turbine design (component sample 18), the team reviewed the RCIC tank vortex calculation and the empirical method that was used to calculate the required water level to assure that a vortex will not draw air into the RCIC System (suppression height). The method that was used to calculate the submergence height (Lubin and Springer) was not consistent with the redesigned outlet

pipings. The Lubin and Springer method was developed based on downward flow through a gravity fed drain whereas the Clinton design withdraws water by pumping it upward through an elbow leading to a horizontal pipe. The team questioned whether the use of the Lubin and Springer method was appropriate and could not validate the submergence height calculation for either the RCIC or the HPCS systems. This was of potential concern because recent changes were made to reduce the margin of the submergence height when accounting for the flow of both the RCIC system and the HPCS system. The team reviewed pertinent documents, including licensee corrective action documents and extent of condition reviews, since the RCIC tank also is a suction source for the HPCS system. The NRC previously performed a 95001 Supplemental Inspection relating to vortexing concerns (IR 05000461/2007009).

The Clinton calculation for the HPCS system, using the Lubin and Springer formula determined that required submergence height 12 inches results in a margin of 5.4 inches prior to vortex formation. The Denny method for a downward facing pipe determines that 2 to 2.5 times the exit diameter is needed when the exit flow is in the range of 10 feet per second. This results in a required submergence height of 30 – 45 inches. This would result in a negative margin in the range of 17.6 to 23.6 inches. In the case of the RCIC system Clinton calculated a required submergence height of 6 inches for a margin of 3.9 inches. The Denny method determined that the submergence height should be approximately 11 to 13.5 inches resulting in a negative margin of approximately 7 to 10 inches. The team noted that, the geometry of the Clinton outlet pipe is different than the Denny experiments and the Denny method may over-predict the required submergence height.

The team did not identify a concern regarding immediate operability relating to the RCIC tank and associated pumps. Nonetheless, the NRC continues to review pump vortexing concerns, and lack of confirmatory testing to evaluate whether this is a potential generic issue.

40A6 Meeting(s)

Exit Meeting

The inspectors presented the inspection results to Mr. F. Kearney and other members of licensee management at the conclusion of the inspection on November 16, 2007. A second telephone exit was conducted on December 19, 2007, to inform the licensee of changes to the findings discussed during the exit meeting on November 16, 2007. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

F. Kearney, Plant Manager
R. Peak, Site Engineering Director
R. Schenck, Work Management Director
J. Domitrovich, Maintenance Director
D. Schavey, Operations Director
C. VanDenburgh, Nuclear Oversight Manager
R. Vickers, Radiation Protection Manager
R. Weber, Sr. Manager Design Engineering
J. Miller, Design Engineering Supervisor

Nuclear Regulatory Commission

J. F. Lara, Chief, Engineering Branch 3
B. Dixon, Senior Resident Inspector

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000461/2007008-01 05000461/2007008-01	NCV	Continuously Submerged Cables Design Deficiency
05000461/2007008-02 05000461/2007008-02	NCV	Division 3 Emergency Diesel Generator Neutral Ground Resistor Design Inadequacy
05000461/2007008-03 05000461/2007008-03	NCV	Inadequate Design of Emergency Diesel Generator Exhaust
05000461/2007008-04 05000461/2007008-04	NCV	Residual Heat Removal Pipe Support Calculation Deficiencies
05000461/2007008-05	NCV	Inappropriate SX Pump Test Acceptance Criteria

Discussed

None

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 or 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.

CALCULATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
19-D-23	Estimating Load for 125 VDC System – MCC 1A	Revision 7
19-D-42	Station Blackout Analysis – 4 Hour Battery Capacity	Revision 4
19-D-14	Molded Case Circuit Breaker Settings for 125 VDC MCC's 1A, 1B, 1D	Revision 11
19-D-28	Review of Division 1 DC System 1A	Revision 14-B
19-AJ-70	Class 1E MCC Control Circuit Voltage and Fuse Adequacy Analysis	Revision 2
19-D-19	Sizing Battery Charger 1A for Div 1	Revision 0
19-AK-13	Analysis of Load Flow, Short Circuit and Motor Starting Using ETAP Power Station	Revision 2
19-D-22	Hydrogen Generation By Batteries During Charging	Revision 0
19-AJ-65	Device Pickup Currents	Revision 1
01D006	Diesel Fuel Oil Storage Requirements	Revision 7
01DO04	DO System Design Pressure	Revision 0
01R16	RCIC Pump Curves Comparison with the System Resistance for Operating Modes A-F	Revision 0A
01R113	RCIC NPSH	Revision 2
08.09.01-3	(Silencer Anchor Bolts)	NA
1D006	Diesel Fuel Oil Requirements	Revision 2
1DG-01	As-built Addendum to Piping Stress Report, Diesel Generator System	Revision 1
1DG-02A	Addendum E to Piping Stress Report, Subsystem 1DG-02A	Revision 2
1DGO07	N-5 Close-Out Addendum for 1DO-07	Revision 6
1EDO07	Summary of Reaction Loads for 1DO01PA (p 19)	Revision 2
3C10-0284-003	Revision to SBO Analysis for Power Uprate	Revision 2
CQD-012917	Stress in Motor Hold-down Bolts 1DO01PA,B,C	Revision 1
CQD012918	Frequency of Diesel Oil Transfer Pump	Revision 0
IP-C-0006	Diesel Fuel Oil Storage Tanks Div I, II, III, Tank volume Calculations	Revision 0
IP-C-0061	Setpoint Calculation for RCIC Storage Tank Low Level Transmitters ...	Revision 1
IP-C-0088	Setpoint Calculation for Suppression Pool High RCIC 1E51N036A&E	Revision 1
IP-M-0133	Vortexing in Fuel Oil Storage Tanks	Revision 1
IP-M-0384	Evaluation of Vortex in the RCIC Storage Tank (Historical)	Revision 2
IP-M-0502	DG Air Start Pilot Operated Solenoid Valves ...	Revision 0

CALCULATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
IP-M-0587	RCIC EOP Operation Max Suppression Pool Temp	Revision 0
IP-M-0761	Evaluation of Vortex in RCIC Storage Tank for HPSC and RCIC Suction Lines	Revision 0
IP-O-0122	Tech Spec Loop Indicator Uncertainty Evaluation, air receiver pressure	Revision 0
IP-Q-0028	Evaluation of Nozzle Reactions for Equipment 1DO01 PA/PB/PC, Diesel Oil Transfer Pumps	Revision 0
IP-Q-0160	Calculation of Torque and Valve Tolerances for Bolts associated with Div I,II,III Lube Oil Pumps and Motors	Revision 0
IP-Q-0373	SC of Lube Oil Tubing, Fittings, Strainers, and Check Valves for Div I/II and III DG Engines, (Includes volumes A,B,C,D)	Revision 2
IP-Q-0442	Seismic Evaluation of Div II DG circulating oil pump/motor assemblies	Revision 0
M/PMED 01R101	RCIC System Piping	Revision 7
MAD 85- 479/686	Diesel Exhaust Rupture Disc	Revision 0
8.9.1-3	Tornado Missile Impact Study	Revision 0
19-AN-04	480 V ESF Switchgear Breakers and Associated Upstream Relay Settings	Revision 0
19-AN 08	4160 V Switchgear Buses 1A1 and 1B1 Motor relay Settings	Revision 3
19-AN-09	4160 V Division 3 ESF Bus 1C1 Motor Relay Settings	Revision 1
19-AN-30	ERAT Protective Relay Settings	Revision 0
19-AN-31	Reserve Auxiliary Transformer (RT1) Protective Relay Settings	Revision 0
19-T-2	Diesel Generator 1A and 1B Neutral Grounding Resistance	Revision 1
EAD-DG-1	Starting KVA During Loss of Offsite Power (LOOP) Coincident with Loss of Coolant Accident (LOCA) for Diesel Generators 1A and 1B	Revision 2
19-AJ-05	Motor Operated Valves Molded Case Breakers	Revision S
SDQ10- 94DG05- 1RH07103X	Mechanical Component Support – Aux. Steel 1RH07103X	Revision B-04
SDQ10- 94DG05- 1SX03019G	Qualification of pipe support 1SX03019G for structural components and mechanical components	Revision 0A
IP-M-0724	Evaluation of subsystem 1RH-08 due to installation of temporary shielding per TSP 2001-145	Revision 0
IP-M-0723	Evaluation of subsystem 1RH-07B due to installation of temporary shielding per TSP 2001-144	Revision 0
1SX20	Pressure Setpoints for Automatic Initiation of SSW pumps 1SX01PA, B and C	Revision 0
IP-M-0470	Evaluation of SX Flow to SX Pump Motor Bearing Cooler	Revision 01

CALCULATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
CQD-018470	Anchor Bolts of Switchgear 4160V	Revision 0
SDQ10-94DG05-1SX48010G	Qualification of pipe support 1SX48010G for structural components and mechanical components	Revision 0A
SDQ10-94DG05-1RH09030X	Pipe Support M-1RH09030X Structural Qualification	Revision 0
IP-M-0704	Evaluation subsystem 1RH-09 due to installation of temporary shielding per TSP 2001-110, 2001-120 and 2001-121	Revision 0
IP-M-0704	Evaluation Subsystem 1RH-09 due to removal of TSP 2001-121	Revision 0-A
IP-M-0704	Evaluation Subsystem 1RH-09 due to installation of temporary shielding package per TSP 2001-110, TSP 2001-120 and TSP 2001-121	Revision 0-B
SDQ10-94DG05-1RH09035X	Pipe Support M-1RH09035X Structural Qualification	Revision F 04
SDQ10-94DG05-1RH09085X	Mechanical Component Support – Aux. Steel 1RH09085X	Revision 2
SDQ10-94DG05-1RH09070S	Mechanical Component Support – Aux. Steel 1RH09070S	Revision 2
SDQ10-94DG05-1RH09041X	Pipe Support M-1RH09041X Structural Qualification	Revision B 03
SDQ10-94DG05-1RH09075S	Pipe Support M-1RH09075S Structural Qualification	Revision 0
EMD 014709	Foundation Loads for Shutdown Service Water Pump (Equipment Number 1SX01PA, B)	Revision 0
EMD 011325	Foundation Loads for Shutdown Service Water Strainer (Eq. No. 1SX01FA, B)	Revision 0
IP-M-0527	G.L. 95-07. Thermally Induced Pressure Locking	Revision 0
IP-M-0005	Water Hammer Calculation for RH System GL 89-10 Valves	Revision 1C
01RH19	Technical Specification Surveillance Requirement For LPCI Pump Differential Pressure At Rated Flow. EC 336808	Major Revision 5
01RH21	RHR pump NPSHA from Suppression Pool	July 21, 2006
01RH29	RHR pump curves for FWLC with RH A-1, A-2, B-1, and B-2	August 3, 1999
None	1E12F014A AC MOV gate valve MIDACALC results	Revision 1
None	1E12F048A AC MOV globe valve MIDACALC results	Revision 1
None	1E12F064A AC MOV gate valve MIDACALC results	Revision 2

CALCULATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
EC 368530	Evaluation of November 29, 2007 Division I SX Pumps test data	December 5, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
00683386	USAR Table 8.3-10 was deleted in update	October 11,2007
00685579	Spring In Check Valve Broken	October 29,2007
00686445	USAR wording 9.5.7.2 should be improved for clarity	October 18,2007
00686539	Walkdown: NRC questioned telephone sitting on (NSR) junction box	October 18,2007
00686546	FSAR 8.3.1.1.2 description of RAT transformer tap setting	October 18,2007
00686557	Plant lighting: Burned out lights in EDG bays.	October 18,2007
00688125	Battery Pilot Cell check procedure error	October 23,2007
00688201	EDG Exhaust Pressure Calculation should be "Historical"	October 23,2007
00688981	Cable temperature compensation for 1A RR pump is non-conservative	October 24,2007
00691005	DURING CDBI NRC FINDS MINOR TYPO IN 3404.01.	October 29,2007
00691631	CDBI response to NRC questions not documented	October 30,2007
00691648	Calc result overload not tracked for closure	October 30,2007
00691813	Improved USAR wording for RCIC water source	October 30,2007
00691894	IP-M-470 has discrepancies for SX Pumps nozzle qualification	October 30,2007
00691903	TS Bases wording error in B 3.8.3.1	October 30,2007
00692530	Thermocouple Wires Near Exhaust Manifold	October 31,2007
00692532	Drawing conflict on resistor size and rating	October 31,2007
00692540	Thermocouple wires near exhaust manifold (Div II EDG)	October 31,2007
00692541	Thermocouple wires near exhaust manifold (Div III EDG)	October 31,2007
00692807	Wrong revision of calculation provided to NRC	October 31,2007
00692997	Submerged Cable Long Term asset management strategy	November 1, 2007
00693180	CDBI: incorrect MOV test data review sheet in work order	November 1, 2007
00693250	Errors found in calc 01DO06, Rev. 7	November 1, 2007
00693264	Old resistance limit used in 8433.07	November 1, 2007
00693451	Bolt allowable stress increased by factor of 1.33	November 1, 2007
00693493	Disclosed document discrepancies with 19-AK-13	November 2, 2007
00693536	Thermocouple wires near exhaust manifold (Div I EDG)	October 31, 2007
00693537	USAR Discrepancy Identified During Calc Review	November 2, 2007
00693540	Clinton Has No calculation for EDG Surveillance 9080.2	November 2, 2007
00693609	RCIC tank Vortex issue	November 2, 2007
00693668	Document clarification - RCIC water temperature values	November 2, 2007
00694902	EDG Exhaust Pressure Calculation Inadequacies	November 5, 2007
00695072	EO5 Drawing error--cites incorrect sheet No. in Title	November 6, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
00695250	Structural calculations not updated	November 6, 2007
00695282	Calculation error on RH pipe support	November 6, 2007
00695303	Common vulnerability on EDG exhaust for snow/ice blockage	November 6, 2007
00695925	Error in the Preliminary Evaluation per IP-M-0704	November 7, 2007
00696851	Detailed structural evaluation requires completion	November 9, 2007
00697048	Div 3 DG grounding resistor housing not ventilated	November 9, 2007
00698734	NRC Identified xfmr with PCB fill	November 13, 2007
00698813	RHR-B Pipe Hangar Over Stress	November 13, 2007
00699334	Evaluate EDG silencers failure modes	November 14, 2007
00700197	Question on DG High Coolant Temp Response During LOOP	November 16, 2007
00700198	No Calc Exists for DG 1C Neutral Grd. Res. Size	November 16, 2007
00700713	October 3, 2007 performance (WO 01044766) of A SX Pump test less than the performance assumed by the design calculation.	November 11, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
00296297	OPEX – Braidwood SME Review of NRC Power Reactor Event 41362	February 2, 2005
00543762	NER NC-06-009 Yellow – Air Voids in S/R Sys Procedure Enhancement	October 13, 2006
00506752	NER NC-06-009 Yellow – Air Voids in Safety Related Systems	July 6, 2006
00664240	Add CDBI Ops Training Review with SBO Scenario	August 24, 2007
00664259	Ops Training Review Critical Objectives to PRA	August 24, 2007
1-99-03-203	Potential 10CFR21 on Possible Misapplication of Agastat E7000 Series Timing Relays	Revision 0
00473470	1DG01KA; Div 1 EDG Neutral Grounding Resistor Cover Spacing	March 3, 2006
00633239	Monitoring Plan for Buried 6.9 kV Cable	May 24, 2007
00114481	Water in MPT to UAT Manholes Submerging Power Cables	July 5, 2007
00244968	Spectrum Replacement AC Bucket Concern	August 17, 2004
00456192	NRC IN 06-02 Galvanized supports/trays with jacketed cable	February 20, 2006
00630226	Some MOV local position indications are obscured	May 16, 2007
00655833	1E21R501: Gauge over-ranged during 9052.01	July 31, 2007
00655836	1E12F084A, wlp check to RHR A, failed to close	July 31, 2007
00655838	1E12F085A, wlp to RHR A, failed to close	July 31, 2007
00658234	NRC questions on op. eval 655836-02 on RHR A	August 7, 2007
00666407	NRC question on RHR A/LPCS water leg pump and piping	August 30, 2007
00666793	NRC question on RHR operability evaluation	August 31, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
ATI 562699-04	Clinton CDBI FASA Report	August 24, 2007

DRAWINGS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
E02-1DC01	Key Diagram 125 V DC MCC 1A (1DC13E)	Revision AF
E02-1DC06	125 V DC & Uninterruptible Power Supply Systems	Revision X
E02-1AP44	Key Diagram 4160 V Bus 1A1	Revision F
E02-1DG99	Diesel Generator 1A Control Part 1	Revision AA
E02-1DG99	Schematic Diagram Diesel Generator 1A Control Part 2	Revision M
E02-1RP99, Sheet 101	Schematic Diagram Reactor Protection System NSPS Power Distribution	Revision P
E02-1RH99, Sheet 509	Residual Heat Removal Sys HX 1A Bypass Valve 1E12-F048A	Revision M
E02-1RH99, Sheet 511	Residual Heat Removal Sys Pump 1A Min Flow Valve 1E12-064A	Revision N
E02-1RH99, Sheet 529	Schematic Diagram Residual Heat Removal Pump 1A	Revision G
E02-1DG99, Sheet 16	Schematic Diagram Diesel Generator 1A Excitation	Revision M
79011	Type N3DBS-137, Delaval IMO Pump	Revision 2
D-77-267	Outline, Type:N3DBS-137 Delaval Pump	Revision 3
EC 347204	Piping Isometrics - RCIC Piping to RV Head	Revision 0
EC-35952	RCIC Elbows for Vortex Mod	Revision 1
E-CLT6-51.0-2	Clinton Power Station Switchyard	Revision 1
E02-1AP03	Electrical Loading Diagram	Revision AA
E02-1AP40	Key Diagram, 6900 V Bus 1A	Revision J
E02-1AP44	Key Diagram, 4160 V Bus 1A1	Revision F
E02-1AP45	Key Diagram, 4160 V Bus 1B1	Revision F
E02-1AP15	Key Diagram, 480 V Unit Substation 1A	Revision K
E02-1HP01	Key Diagram, ESF Div. 3 HPCS MCC 1C	Revision S
E02-1AP01	Single Line Diagram, Part 4	Revision H
E02-0AP01	Key Diagram, 480 V Unit Substations A and B	Revision H
E02-0APO2	Key Diagram, 480 V Unit Substations C	Revision E
E02-1AP16	Key Diagram, 480 V Unit Substation 1B	Revision L
GE 762E298AC	One Line Diagram High Pressure Core Spray System	Revision 10
GE 762E298AC	One Line Diagram High Pressure Core Spray System, Bus 1C1	Revision 12
E05-1013	Grounding and Conduits in Curbstructure Unit and Reserve Auxiliary Transformer Plan	Revision V
E05-1017	Grounding and Conduits Substructure, Screenhouse	Revision S

DRAWINGS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
E05-1029, Sheet 1	Area Grounding and Conduits in Substructure ERAT and Pumphouse Duct Runs – Sections and Details,	Revision G
E05-1029, Sheet 2	Grounding and Conduits in Substructure ERAT and Pumphouse Duct Runs – Sections and Details,	Revision J
E05-1029, Sheet 3	Grounding and Conduits in Substructure ERAT and Pumphouse Duct Runs – Sections and Details,	Revision F
E05-1029, Sheet 4	Grounding and Conduits in Substructure ERAT and Pumphouse Duct Runs – Sections and Details,	Revision D
E02-1AP04, Sheet 20	Transformer Tap Settings	Revision C

MODIFICATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
ECN 31580	Installation of Interposing Relay into Voltage Regulator Circuit	April 4, 1999
EC 0000349235	Replace Division 1 Diesel Generator Governor Actuators	Revision 1
EC 359252	HPCS and RCIC Pump Suction Line Submersion	Revision 1
EC-330624	Class 1E MCC Molded Case Circuit Breaker Control Unit (Bucket) Replacement	Revision 0
EC-350432	Replacement of Obsolete ITE/Gould J10 Control Relays Used in Safety and Non-Safety 5600 Series 480 V MCC Applications	Revision 1
ECN 31302	Piping Changes for 1SX01PA Spare Motor Change- Out	Revision 0
ECN 30873	Modify Valve 1E12F064A To Eliminate Pressure Lock.	September 5, 1998

OPERABILITY EVALUATIONS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
655836-02	E12F84A and E12F85A leaking	August 2, 2007
700713-02	SX Pumps Operability Evaluation	November 21, 2007
700713-02	SX Pumps Operability Evaluation	December 6, 2007

OPERATING EXPERIENCE

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
AR 00296788	OPEX – Bus Single Failure – Crystal River – NRC IN 2005 -04	February 2, 2005

PROCEDURES

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
CPS 4401.01	RPV Control	Revision 28
CPS 4404.01	ATWS RPV Control	Revision 28
CPS 4403.01	RPV Flooding	Revision 28
CPS 4407.01	Emergency RPV Depressurization	Revision 28
CPS 4402.01	Primary Containment Control	Revision 28
CPS 4406.01	Secondary Containment Control	Revision 28
CPS 4701.01	Primary Containment Flooding	Revision 4
CPS 4702.01	RPV, Containment, and Radioactivity Release Control	Revision 4
CPS 3101.01	Main Steam	Revision 20
CPS 3112.01	Condenser Vacuum	Revision 19c
CPS 3112.01P001	Emergency/Post Scram Startup of a Vacuum Pump	Revision 0c
CPS 3204.01	Turbine Building Closed Cooling Water	Revision 11
CPS 3211.01	Shutdown Service Water	Revision 24e
CPS 3312.01	Residual Heat Removal	Revision 37e
CPS 3319.01	Standby Gas Treatment	Revision 16
CPS 3404.01	Fuel Building HVAC	Revision 11
CPS 4004.02	Loss of Vacuum	Revision 1g
CPS 4100.01	Reactor Scram	Revision 19a
CPS 4200.01	Loss of AC Power	Revision 16e
CPS 4200.01C002	DC Load Shedding During a SBO	Revision 4a
CPS 4200.01C003	Monitoring CNMT Temperatures During a SBO	Revision 1
CPS 4200.01C004	Manual CNMT Isolation During a SBO	Revision 3
CPS 4410.00C010	Defeating CNMT Vent Interlocks	Revision 4
CPS 4411.06	Emergency Containment Venting, Purging, and Vacuum Relief	Revision 4b
TQ-AA-106-0114	Simulator Demonstration Examination Crew Competency Evaluation Form	Revision 3
TQ-AA-106-0113	Simulator Demonstration Examination Individual Competency Evaluation Form	Revision 4
TQ-AA-108-0101	Simulator STA/IA Competency Evaluation Form	Revision 1
3506.1C001	Diesel Generator 1A Pre-Start Checklist	Revision 14
OP-AA-108-111-1001	Severe Weather and Natural Disaster Guidelines	Revision 2
CPS 3310.01	Reactor Core Isolation Cooling	Revision 26d
3506.01	Diesel Generator and Support Systems	Revision 32g

PROCEDURES

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
9054.01	RCIC System Operability Check	Revision 42e
9054.01D004	Combined RCIC (1E51-c001) High Pressure Operability Check and RCIC Cold Quick Restart Checklist	Revision 1d
003.01C002	RSP-RCIC Operation	Revision 2b
9054.01C001	RCIC Water Leg pump (1E51-C003) Operability test 1851-F040 Closure Test and 1SX37/timing	Revision 6
9054.01C002	RCIC(1E51-C001) High Pressure Operability Checks	Revision 2c
9054.01C003	RCIC (1E51-C001) Low Pressure Operability Test	Revision 2d
9054.01D002	RCIC (1E51-C001) High Pressure Operability Check list	Revision 23B
CC-AA-103	Configuration Change Control for Permanent Physical Plant Changes	Revision 14
SM-AA-300	Procurement Engineering Support Activities	Revision 1
ER-AA-3003	Cable Condition Monitoring Program	Revision 0
CC-AA-112	Temporary Configuration Changes	Revision 12
CC-AA-401	Maintenance Specification: Installation and Control of Temporary Shielding and Shielding Components	Revision 8
CPS 5064.01	Auto Start of Shutdown Service Water Pump 1A, page 5 of 7	Revision 31b
CPS 5065.01	Auto Start of Shutdown Service Water Pump 1B, page 7 of 7	Revision 29a
CPS 5064.01	Auto Start of Shutdown Service Water Pump 1C, page 3 of 7	Revision 31b
CPS 9052.01	LPCS /RHR A PUMPS and LPCS/RHR A WATER LEG PUMP OPERABILITY completed July 31, 2007	Revision 43d
CPS 9052.01D001	LPCS/RHR A PUMP and LPCS/RHR A water leg pump operability data sheet completed July 31, 2007	Revision 41b
CPS 9053.04	RHR A/B/C valve operability checks	Revision 44c
CPS 9053.04C001	RHR loop A valve operability completed April 30, 2007	Revision 1c
CPS 9053.04D001	RHR loop A valve operability data sheet completed April 30, 2007	Revision 43b
9069.01	SX System Operability Test	Revision 45b

VENDOR DOCUMENTS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
NEDO-10905	High Pressure Core Spray System Power Supply Unit	May 1973
K-2983	Specification for Control Cable	Revision Amd. 8
K-2982	Specification for Power Cable	Revision Amd. 8

WORK DOCUMENTS

<u>Number</u>	<u>Description or Title</u>	<u>Date or Revision</u>
WO 01064514	125 V DC Pilot Cell Check (Div I)	September 28, 2007
WO 01044399	Quarterly ICV and Charger Checks for Div I Battery	October 4, 2007
WO 00019737	125 V DC Battery Service Test	April 22, 2002
WO 00501098	Div 1 Battery Modified Performance Test	October 15, 2003
WO 00666741	125 V DC Battery Service Test (Div I)	May 16, 2005
2007-08-0080A	Create JPM for Emergency Containment Venting	August 15, 2007
2007-08-0194A	Training Request for LORT CRC – Room Cooling Recovery	August 28, 2007
2007-08-0196A	Training Request for LORT CRC – Important Operator Actions PRA	August 28, 2007
WO 00019109	Perform 1PSDG005A Calibration (8801.01)	July 18, 2002
WO00573115	Pressure Switch Stops Compressor Too Late	August 14, 2003
WO0019110	Perform 1PSDG005B Calibration (8801.01) and Check Relay K12	July 17, 2002
WO1079294-01	Operability Evaluation - Degraded emergency diesel generator exhaust system	Revision 0
WO 00005345	Perform 1PSDG005B Calibration	Revision 1
WO 00005341	Perform 1PSDG005A Calibration	Revision 1
WO01038306	DG 1A Monthly Test Results	Revision 0
WO00863080	DG 1B 24 Hour Run, 10/13/07	Revision 0
WO00693078	DG 1B 24 Hour Run, 10/18/05	Revision 0
WO00673376	Manhole Inspection	June 23, 2005
WO 01025783	RHR A loop A valve operability completed July 31, 2007 per 9053.04C001/D001	July 31, 2007
WO 00999341	9052.01A21 OP RHR A pump oper	May 1, 2007
WO 00999701	RHR LOOP a valve operability completed April 30, 2007 per 9053.0404C001/D001	May 1, 2007
WO 01026130	RHR A pump oper completed July 31, 2007 per 9053.04	July 31, 2007
WO 00322054	Clean/Inspect 1SX01FA strainer motor	September 11, 2002
WO 00362080	Clean/Inspect 1SX01FA strainer	September 11, 2002
WO 00485343	1E12F064A Thrust Verification by VOTES	October 5, 2003
WO 00002627	1E12F014A Thrust Verification by VOTES	August 14, 2001
WO 00002629	1E12F048A Thrust Verification by VOTES	February 6, 2002
MWR D79967	Valve 1E12F064A modified eliminating Pressure Lock.	September 28, 1998
WO 01044766	9069.01A20 SX Pump A Operability Test completed October 4, 2007	October 4, 2007

LIST OF ACRONYMS USED

°F	degrees Fahrenheit
AC	Alternating Current
ACB	Air Circuit Breaker
ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
C&I	Control and Instrumentation
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CW	Circulating Water
DC	Direct Current
EDG	Emergency Diesel Generator
DGCW	Diesel Generator Cooling Water
DRS	Division of Reactor Safety
GL	Generic Letter
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilation and Air Conditioning
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IR	Inspection Report or Issue Report
IST	In-service Testing
M&TE	Measurement and Test Equipment
MCC	Motor Control Center
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U. S. Nuclear Regulatory Commission
P&ID	Piping and Instrumentation Diagram
PARS	Publicly Available Records System
PRA	Probabilistic Risk Assessment
RHR	Residual Heat Removal
SDP	Significance Determination Process
T/C	Thermocouple
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
V	Volts
WO	Work Order