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
Director - Licensing and Regulatory Affairs
Manager Licensing - Clinton, Dresden and Quad Cities
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Assistant Attorney General
State Liaison Officer
Illinois Emergency Management Agency
Chairman, Illinois Commerce Commission
Illinois Emergency Management Agency

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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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REGION III

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| 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<br>December 19, 2007 |
| Inspectors: | Z. Falevits, Senior Engineering Inspector, Lead<br>M. Munir, Engineering Inspector<br>D. Tharp, Resident Inspector<br>G. O'Dwyer, Engineering Inspector<br>F. Baxter, Electrical Contractor<br>S. Spiegelman, Mechanical Contractor |
| Observer: | J. Bozga, Engineering Inspector |
| Approved by: | J.F. Lara, Chief<br>Engineering Branch 3<br>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 Standarized 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 form 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

piping. 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.

4OA6 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<br>05000461/2007008-01 | NCV | Continuously Submerged Cables Design Deficiency |
| 05000461/2007008-02<br>05000461/2007008-02 | NCV | Division 3 Emergency Diesel Generator Neutral Ground Resistor Design Inadequacy |
| 05000461/2007008-03<br>05000461/2007008-03 | NCV | Inadequate Design of Emergency Diesel Generator Exhaust |
| 05000461/2007008-04<br>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 of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

### CALCULATIONS

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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 1I |
| 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 |
| 01RI13 | 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 1DOO1PA,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

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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 Up[stream Relay Settings | Revision 0 |
| 19-AN 08 | 4160 V Switchgear Buses 1A1 and1B1 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

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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

| Number | Description or Title | Date or Revision |
|---|---|---|
| EC 368530 | Evaluation of November 29, 2007 Division I SX Pumps test data | December 5, 2007 |

## CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| ATI 562699-04 | Clinton CDBI FASA Report | August 24, 2007 |

**DRAWINGS**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| | Area | |
| E05-1029, Sheet 1 | 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

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

**PROCEDURES**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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**

| Number | Description or Title | Date or Revision |
|---|---|---|
| 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 | Auguust 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 |