November 28, 2006

Mr.Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
Quad Cities Nuclear Power Station
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC COMPONENT DESIGN BASES INSPECTION (CDBI)
INSPECTION REPORT 05000254/2006003(DRS), 05000265/2006003(DRS)

Dear Mr. Crane:

On September 15, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Quad Cities Nuclear Power Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on September 15, 2006, with Mr. R. Gideon and other members of your staff. A second exit was conducted on November 3, 2006, with Mr. T. Tulon and other members of your staff to discuss changes with the initial inspection results.

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

Based on the results of this inspection, 12 NRC-identified findings of very low safety significance were identified, 11 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 (NCV) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U. S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Quad Cities Nuclear Power Station, Units 1 and 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (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/

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

Docket Nos. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosure: Inspection Report 05000254/2006003(DRS); 05000265/2006003(DRS)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Quad Cities Nuclear Power Station
Plant Manager - Quad Cities Nuclear Power Station
Regulatory Assurance Manager - Quad Cities Nuclear Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional
Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
Vice President - Law and Regulatory Affairs
Mid American Energy Company
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
D. Tubbs, Manager of Nuclear
MidAmerican Energy Company

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Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
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D. Tubbs, Manager of Nuclear MidAmerican Energy Company

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos.:	50-254; 50-265
License Nos.:	DPR-29; DPR-30
Report No.:	05000254/265/2006003(DRS)
Licensee:	Exelon Nuclear
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	Cordova, Illinois
Dates:	August 14 through September 15, 2006; November 3, 2006
Inspectors:	Z. Falevits, Senior Engineering Inspector, Lead Inspector C. Brown, Engineering Inspector M. Munir, Engineering Inspector G. O'Dwyer, Engineering Inspector B. Sherbin, Mechanical Contractor S. Kobylarz, Electrical Contractor S. Burgess, Senior Risk Analyst
Approved by:	A. M. Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000254/2006003(DRS), 05000265/2006003(DRS); 08/14/2006 - 09/15/2006; Quad Cities Nuclear Power Station, Units 1 and 2; Component Design Bases Inspection (CDBI).

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. Twelve findings of very low safety significance were identified with eleven associated Non-Cited Violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process (SDP).” Findings for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3; dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The team identified a Non-Cited Violation (NCV) of Technical Specification (TS) Surveillance Requirements (SR) 3.8.4.2, Amendment 199/195, having very low safety significance for failure to meet the TS SR when visible corrosion on Units 1 and 2, 125 Vdc safety-related battery inter-cell and terminal connections was identified. Upon discovery, the licensee’s corrective actions included: initially cleaning of all 125 Vdc terminals and connectors; taking connection resistance measurements; and initiating a root cause analysis to identify the cause(s) of this adverse to quality condition.

  The finding was more than minor because failure to ensure that Units 1 and 2, 125 Vdc safety-related batteries are being maintained in accordance with vendor specified requirements, applicable procedures and TS SRs could result in unacceptable battery terminal connection resistances and decreased battery capacity, rendering the DC system incapable of performing its intended safety function. Based on the results of the licensee’s analysis, the finding was determined to be of very low safety significance using the SDP Phase 1 screening worksheet. The cause of the finding related to the cross-cutting aspect of human performance. (Section 1R21.3.b.1)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance involving the failure to verify and ensure that the 125 Vdc safety-related batteries would remain operable if all the inter-cell and terminal connections were at the resistance value (< 150 micro-ohms) allowed by TS SR 3.8.4.2 and SR 3.8.4.5.

  The finding was more than minor because if left uncorrected, the finding could become a more significant safety concern. Specifically, the 125 Vdc safety-related batteries would become incapable of meeting their design basis function if the inter-cell and connection resistance were allowed to increase to the TS allowed value. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.2)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving inadequate design review of the loading calculation for the emergency diesel generators (EDG's). Specifically, the licensee's engineers failed to adequately identify design input data and perform an adequate design review of the design data for the EDGs that was used in the auxiliary power analysis and the EDG loading calculations. The licensee subsequently determined that the EDGs were operable and that the load margin was not adversely affected based on a revised loading calculation.

  The finding was more than minor because failing to correctly identify and input the correct equipment design data into the auxiliary power analysis program would result in the load conditions on the EDG's or other areas of the electrical power analysis not being accurately evaluated, resulting in inaccurate determination of EDG loading. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet.
  (Section 1R21.3.b.3)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving licensee's failure to select an appropriate method for calculating the onset of vortexing at the intake of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps' suction lines from the contaminated condensate water storage tank (CCST) water storage tank. Additionally, the licensee failed to fully account for the impact of instrument uncertainty in the tank level switch setpoint which determines the point where suction for the pumps is switched from the CCST to the torus. Once identified, the licensee issued IR 00524923 which contained an evaluation of a more appropriate method for determining the onset of vortexing in the tank.

  The finding was more than minor because the failure to prevent the formation of vortexing at the intake of the HPCI and RCIC suction lines would result in air entrainment causing pulsating pump flow and/or reduction in pump performance. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet.
  (Section 1R21.3.b.4)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving the sizing calculation for the Target Rock ADS/SRV air accumulator tank. Specifically, the team identified that the licensee failed to correctly specify the minimum differential air pressure required to actuate the ADS/SRV valves, failed to include the volume of the piping from the solenoid to the ADS/SRV actuator, and had the wrong assumption for leakage rate used as acceptance criteria in air drop testing. Once identified, the licensee determined that the calculation required revision to correct the problems that were identified by the team.

The finding was more than minor because the failure to have adequate pneumatic pressure and volume in the accumulator tank would result in over-predicting the accumulator capacity. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.5)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance concerning the failure to use proper and most current design input for the control circuit voltage drop calculation for safety related motor operated valves in motor control center 28-1B. Subsequently, on September 1, 2006, the licensee determined, based on review of other electrical design calculations, that the affected circuits will have adequate voltage to ensure proper function of the valves components

  The finding was more than minor because the licensee failed to update the control circuit voltage drop calculation for the MOVs to reflect the more conservative MCC design input voltage and ensure the correct voltage for the motor contactor pick up was available. This finding has been screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.6)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance for failing to maintain an adequate procedure for establishing an accurate load tabulation to ensure that the bus feeder breakers to Bus 24-1 were not overloaded during bus cross-tie operation. Specifically, the procedure did not require entering the expected load data from Bus 14-1 during a bus cross-tie operation into the load tabulation. Once identified, the licensee entered the finding into their corrective action program as IR 00521012 and planned to revise the procedure.

  The finding was more than minor because, if left uncorrected, it could result in an overloaded bus feeder breaker, since Bus 14-1 cross-tie load could not be accounted for in the tabulation of the Bus 24-1 loading. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.7)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance for the operations 125 Vdc safety-related battery procedure being discrepant from vendor specified instructions and other plant battery procedures. Specifically, the procedure stated that, "if electrolyte is spilled on batteries, then use only demineralized water for cleaning." This differed from the vendor's specific instructions and other maintenance procedures which stated that electrolyte spill on batteries shall be neutralized with baking soda water solution. The licensee entered the finding into their corrective action program as IR 00525113.

  The finding was more than minor because demineralized water will not neutralize the electrolyte spill on the batteries and could lead to undesirable consequences such as corrosion and potentially affect the battery's design function. This finding has been screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.8)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance for failure to follow the 125 Vdc station battery preventive maintenance procedure requirement and vendor recommendation not to re-torque corroded battery cell connections. Additionally, the licensee failed to document the as left re-torque values, after re-torquing was performed. Subsequently, the licensee evaluated the as-found conditions and determined the batteries remained operable.

  The finding was more than minor because frequent re-torquing of connections will result in distortion of cell posts and connectors, thus degrading rather than improving the connections and may result in affecting the capability of the battery in performing its safety function. This finding has been screened as Green using the SDP Phase 1 screening worksheet. The cause of the finding related to the cross-cutting aspect of human performance. (Section 1R21.3.b.9).

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance for failure to ensure that the HPCI pump hydraulic performance tests had acceptance criteria that incorporated the acceptance limits from applicable design documents. If the HPCI pump had degraded to the lower limit of the acceptance band, as listed in the test acceptance criteria, the pump would not have been able to meet the design basis discharge pressure and flow requirements. Following the identification of the issue the licensee entered the issue into the corrective action program as IR 00525592 and verified the operability of the pump based on actual test results.

  The finding was more than minor because inadequate pump testing could result in HPCI pump not capable of providing the required design basis flow during accident conditions. The finding was of very low safety significance and screened as Green because subsequent analysis determined that the pumps were currently capable of meeting the design basis discharge pressures and flows. (Section 1R21.3.b.10)

- Green. The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance involving the air drop testing for the Target Rock ADS/SRV air accumulator tank. Specifically, the team identified that the licensee failed to correctly specify the minimum accumulator pneumatic pressure required to test the Target Rock ADS/SRV valves. Once identified, the licensee entered the finding into their corrective action program as IR 0052383 to revise the test procedure. An Operability Evaluation for Unit 1 was performed by the licensee to ensure system operability was not affected.

  The finding was more than minor because the failure to test the pneumatic accumulator tank at its design basis minimum pressure would result in over-predicting the accumulator capacity. This condition could effect reliable operation of the Target Rock ADS/SRV valves. The finding was of very low safety significance because licensee determined the issue was a test deficiency confirmed not to result in loss of operability per “Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment. (Section 1R21.3.b.11)

- Green. The team identified a finding of very low safety significance involving shift management failing to adequately document the basis for prompt operability calls on condition reports (CRs). Specifically, the team identified that the licensee failed to adequately identify and document the basis and logic for continued operability of the Unit 1 and Unit 2 125 Vdc battery on CRs that identified corrosion on the battery connections.

  The finding was more than minor because it could reasonably be viewed as a precursor to a significant event and if left uncorrected, the finding could become a more significant safety concern. Specifically, failing to maintain adequate rigor in ascertaining and verifying the basis for operability calls could lead to an incorrect conclusion which could result in a component not fulfilling its design basis in an event. Based on the results of the licensee’s analysis, the finding screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.12)

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

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.

In addition, the team reviewed several licensee audits and self-assessments to assess how effective licensee personnel were at self-identifying problems. The assessment was accomplished by comparing licensee-identified problems with problems that the team identified during this inspection. The sample included a self-assessment in preparation for the CDBI and selected assessments of the Engineering Design Control program.

.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 Quad Cities Standardized Plant Analysis Risk (SPAR) Model, Revision 3.21. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 2.0 and/or a risk reduction worth of greater than 1.005. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios.

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 failed performance test results, significant corrective action, repeated maintenance activities, maintenance rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem 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 team reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), component/system design basis documents, drawings, and other available design basis information, to determine the performance requirements of the selected components. The team used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code and the Institute of Electrical and Electronics Engineers (IEEE) Standards, to evaluate acceptability of the systems' design. The review was to verify that the selected components would function as 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 team reviewed the maintenance history, system health reports, OPEX related information and condition reports. Field walkdowns were conducted for all accessible components to assess material condition and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 16 components were reviewed (16 inspection samples):

1. **Turbine Building 125 Vdc Main Bus 1A (ESS. Div. 1):** The team reviewed short circuit calculation for the distribution panel, breaker interrupting ratings and electrical coordination, and electrical schematics for selected Appendix R circuits to ensure that coordination existed between the downstream and the upstream fuses.

2. **125 Vdc Battery No. 1:** The team reviewed various electrical documents including battery load and margin calculations, battery float and equalizing voltages, overall battery capacity, battery single cell charge procedure, performance discharge test (initial acceptance test), weekly battery surveillance tests, quarterly battery surveillance tests, electrical maintenance procedures, short circuit calculation for distribution panel, breaker interrupting ratings and electrical coordination. The team also reviewed electrical schematics for selected Appendix R circuits to ensure that coordination existed between the downstream and the upstream fuses.

3. **125 Vdc Battery Charger No. 1:** The team reviewed battery charger sizing calculation, testing data, environmental qualifications, preventative maintenance documents and OPEX related information.

4. **ADS Valves 2-0203-3a (Air Operated Valve):** The team reviewed calculations used for sizing of the air storage tank, and structural support capability of the individual safety relief valve (SRV) accumulators to ensure the Target Rock ADS valves were capable of functioning under design conditions. The team also reviewed the air leak rate testing procedure, and recently completed leak rate testing performed for the air system connected to the accumulators to verify that the acceptance criteria were appropriate and data was within the defined criteria. The team also reviewed licensing and design basis requirements for the Target Rock ADS/SRV valves related to commitments made as a result of Extended Power Uprate (EPU).

   The team also reviewed control logic schematic diagrams, system description, and flow control diagrams to verify the adequacy of valve control logic design and to ensure that the valve was capable of functioning under design conditions. In addition, the team reviewed testing of the air system, accumulators, design calculations.

5. **RHR Train 2A Pump (2-1002-A):** The team reviewed design calculations to ensure that the pump's design requirements were properly determined, (e.g., pump pressures and flows), required pump suction submergence and net positive suction head (NPSH). The team ensured that design basis requirements were correctly translated into test acceptance criteria. The team reviewed completed tests to ensure the tests demonstrated the pump's capability to perform its design basis required functions. The team reviewed the system normal and abnormal operating procedures to ensure component operation and alignments were consistent with the design bases. Design change history and IST results were reviewed to assess potential component degradation and impact on design margins. The team also ensured that completed tests were accomplished appropriately at an appropriate frequency.

   The team also reviewed the electrical diagrams, electrical system and voltage analysis, motor protective relay coordination and settings, recent preventative maintenance for the motor feeder breaker and protective relays.

6. **RHR Train 2A HX (2-1003A-2A):** The team reviewed design documents to ensure the design requirements for preventing excessive flow-induced vibrations in the heat exchanger. The team ensured that design basis requirements were properly translated into operating procedures. The licensee informed the team that there had been no indications of vibration problems with this heat exchanger within the last 2 years.

7. **4160 Vac ESS Switchgear Bus 24-1:** The team reviewed electrical diagrams, the UFSAR, surveillance testing and the electrical distribution system calculations to assess the status and maintenance condition of the equipment

and to verify the adequacy of bus and circuit breaker load capacity, short circuit ratings, bus voltage, electrical protection and coordination, and interrupting ratings of the circuit breakers, to ensure that the bus bracing and the circuit breakers can withstand the short circuit current.

The team reviewed recent preventive maintenance and performed a walkdown of the switchgear to verify the as-built condition and to verify the protective relay settings and the second level undervoltage (degraded) relay setpoint calculation for the 4160 Vac bus. The team also reviewed the calibration procedure and the latest calibration data sheets to ensure that the relays were set in accordance with the calculation.

8. **Division II 4160 Vac Crosstie to Unit 2 Bus 24-1 (Breaker 1421):** The team reviewed electrical diagrams, the UFSAR, bus loading and crosstie operating procedures, circuit breaker vendor documentation, recent preventive maintenance, surveillance testing and the electrical distribution system calculations to assess the status and maintenance condition of the equipment and to verify the adequacy of bus and circuit breaker load capacity, short circuit ratings, bus voltage and electrical protection. The team interviewed plant engineers concerning the electrical distribution system calculations and power system analysis. The team interviewed plant operators on bus crosstie procedures. A walkdown of the switchgear was conducted to observe general material condition of the selected components and to verify the protective relay settings.

9. **Division 1/2 EDG (0-6601):** The team reviewed the diesel generator loading calculation including the loading sequence during loss of offsite power (LOOP) and loss of coolant accident (LOCA). The team reviewed electrical diagrams, UFSAR, system operating and test procedures, protective relay settings, and the electrical distribution system calculations to verify the adequacy of protective relaying scheme and to verify that operator actions were consistent with the UFSAR and Technical Specifications. The team reviewed the diesel generator the latest protective relaying completed surveillance procedure to verify that the relays operated properly.

   The team also reviewed design calculations to ensure that the fuel oil, air starting and room ventilation system design requirements were properly determined, e.g., required flow rates and tank capacities. The team ensured that design basis requirements were correctly translated into test acceptance criteria. The team reviewed completed tests to ensure the tests demonstrated the systems' capability to perform their design basis required functions. To ensure the quality of the fuel oil and starting air, the team ensured that appropriate chemical control programs were in place, e.g., appropriate moisture and impurity controls. The team reviewed the systems' normal and abnormal operating procedures to ensure component operation and alignments were consistent with the design bases.

The team reviewed the hydraulic analysis, and system flow tests that verified the DG jacket water heat exchanger would receive minimum design cooling water flow from the Division 1/2 Emergency Diesel Generator Cooling Water Pump, in accordance with the flow value specified in the thermal evaluation of the heat exchanger. The team reviewed heat exchanger inspections that were performed to verify heat transfer capability. The team interviewed engineering personnel to ascertain whether the condition of the heat exchanger and attached piping was meeting the guidance of GL 89-13. The team reviewed the tube plugging count, and limits determined for the heat exchanger.

10. **Unit 1/2 EDG CW Pump:** The team reviewed the electrical diagrams, including the schematic diagram, electrical system and voltage analysis, and recent preventative maintenance for the pump motor feeder breaker. The team also reviewed calculations related to pump flow, head, and NPSH requirements to ensure the pump was capable of functioning as required. Design change history and pump IST results were reviewed to assess potential component degradation and impact on design margins. Recent surveillance tests related to flow distribution were reviewed to ensure adequate flow to the diesel generator, and room coolers that are supplied by the pump. The potential for pump room internal flooding was reviewed.

11. **Service Water (SW) System Strainers 2-3902 (and 0-3902, 1-3902):** The team reviewed the preventive maintenance tasks, corrective maintenance history, problem history, and operating history to ensure the valves were capable of performing their required functions under required conditions. Test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Walkdowns were performed to ensure that the installed configuration was consistent with design calculations. Walkdowns also ensured appropriate physical condition of the valves.

12. **Unit 2 High Pressure Coolant Injection (HPCI) Pump (2-2302):** The team reviewed instrument set point calculations, calculations related to pump's net positive suction head (NPSH) to ensure the pump was capable of functioning as required. Hydraulic calculations were reviewed to ensure design requirements for flow and pressure were translated as acceptance criteria for pump in-service testing (IST). Design change history and IST results were reviewed to assess potential component degradation and impact on design margins. Technical Specification surveillance procedures for the HPCI pump were reviewed to ensure surveillance requirements were met. Maintenance and calibration procedures were reviewed to ensure instrument setpoints were consistent with design basis assumptions. In addition, the licensee responses and actions to Bulletin 88-04, "Potential Safety-Related Pump Loss" were reviewed to ensure pump minimum flow requirements were addressed. The team reviewed vortexing calculations for HPCI pump suction alignment to the suppression pool and CCST.

13. **RHR and RHRSW Outlet MOVs (2001-5A and 2001-5B):** The team reviewed the motor-operated valve (MOV) calculations, including required thrust, and

maximum differential pressure, to ensure the valve was capable of functioning under design conditions. Diagnostic and IST results were reviewed to verify acceptance criteria were met and performance degradation would be identified. The team reviewed the degraded voltage calculation for both the power and the control circuit of the MOVs to ensure that the valves were capable of performing their function under design conditions. The team also reviewed the control logic schematic diagrams, the system description, and flow control diagrams to verify the adequacy of valve control logic design and to ensure that the valve was capable of functioning under design conditions.

14. **Containment Damper (1-1601-24):** The team reviewed the licensee responses and actions to Generic Letter 89-16, "Installation of Hardened Wetwell Vent" to ensure the vent was installed in accordance with the requirements of the letter. The team also reviewed the BWR Owners Group Design Criteria that Quad Cities committed to use in designing the hardened vent system. The team discussed the timeline assumed in the PRA for failing to open the vent due to a loss of air supply to ensure that the assumptions on the time to open the vent were consistent with the time to restore the instrument air equipment. The team reviewed design calculations to ensure the vent valve would open under design basis conditions. Surveillance test results were reviewed to demonstrate and trend the valves ability to open.

15. **RPS Trip System Relays:** The team reviewed RPS electrical logic, schematic and wiring diagrams, the system description, the UFSAR, preventive maintenance activities, functional surveillance testing and corrective action taken and proposed for RPS related relay issues, Several interviews were conducted with the RPS system engineer regarding ongoing system issues, surveillance testing, condition monitoring tests, data trending and preventive maintenance.

16. **Unit 1 ECCS Suction Strainers (1-1600-4, 8, 12, 16):** The team reviewed strainer design requirements to ensure debris loading assumptions were consistent with industry guidance. The team reviewed NPSH calculations for ECCS pumps to ensure that the hydraulic pressure drop through the strainers were considered for design basis strainer debris loading. The team reviewed periodic inspection program to ensure strainers were maintained in a clean condition.

b. Findings

The team identified 12 findings of very low safety significance of which 11 were associated with Non-Cited Violation (NCV).

.1 Failure to Comply with TS SR 3.8.4.2 for 125 Vdc Battery Terminal Connection Corrosion and Resistance Measurements

Introduction: The team identified a NCV of Technical Specification (TS) Surveillance Requirements (SR) 3.8.4.2, Amendment 199/195, having very low safety significance (Green) for licensee's failure to comply with the quarterly TS SR to verify adequacy of

Units 1 and 2, 125 Vdc safety-related battery inter-cell and cable to plate electrical connections. The team identified several time periods (between November 2004 and August 2006) greater than the TS specified limit where the licensee failed to consistently identify, document or take battery connections resistance measurements on all battery cell terminations containing visual corrosion.

Description: On August 15, 2006, during design review of selected DC system components, the team, with the DC system engineer, performed a field walkdown and visual inspection of the Units 1 and 2, 125 Vdc, safety-related batteries. The team identified that corrosion existed on multiple battery inter-cell electrical connections. Unit 1 deficiency tag No. 181328 was noted to be hanging on the Unit 1 battery. The tag was initiated by operations during the weekly surveillance performed on January 10, 2006, to document that eight battery inter-cell connections (9-, 20+, 21-, 27-, 34-, 48-, 50-, and 52-) were identified as having corrosion and to request that the corrosion be cleaned. No other deficiency tags were noted hanging in the Unit 1 battery room to identify and record the corrosion visually observed by the team on numerous other battery inter-cell and terminal to plate electrical connections.

Maintenance procedure QCEPM 0100-01, Revision 22, Page 6, defined corrosion as, "a growth that is nodular, powdery, thick, cauliflower-like or otherwise three dimensional. It is usually green, although it may be white or dark." The team noted that numerous battery inter-cell connections contained corrosion that very closely matched the description defined in the procedure. In response to the team's concerns, on August 16, 2006, the licensee Fix It Now (FIN) team was dispatched to clean the 12 corroded cell terminals that were identified in the scope of work in Unit 1 WO 00883795 01 (which also included the 8 cells identified on January 10, 2006, on deficiency tag No. 181328). The team reviewed the completed WO package and noted the following work control related deficiencies:

- The WO documented that three deficiency tags had been removed after cleaning the corroded terminals while on August 15, 2006, the team observed that only tag No.181328 was hanging on the Unit 1, 125 Vdc battery. The licensee later determined that the second deficiency tag, No.197381, was issued for Unit 2 125 Vdc battery corroded cells (Nos. 1+, 23+, and 29+) and should not have been included in the Unit 1 WO package. In addition, the licensee could not identify source of the third deficiency tag No. 151704 (which documented corrosion on cell No. 45) that was also apparently removed during performance of the Unit 1 WO 00883795 01.

- Cell No. 45+ was not listed as having been cleaned even though it was associated with the third deficiency tag (No. 151704). The team noted that only 11 of the 12 corroded cells had been documented as being cleaned. Cell No. 45+ was identified as corroded on June 27, 2006, and was merged into the Unit 1 WO via IR 00504285. The cell was subsequently cleaned on August 29, 2006, using WO 00883795 01 after the team questioned why only 11 corroded cell terminals had been cleaned on August 16, 2006, when numerous other cells were observed by the team to be corroded.

- The team also determined that on June 20, 2006, Unit 1 surveillance WO 00930774-01 identified cell No. 24+ as being corroded and needing cleaning. IR 00440765 was initiated as required by procedure; however, the cell was not added to any WO to be cleaned. Cell No. 24+ was subsequently cleaned on August 29, 2006, using WO 00883795-01.

Because of the observed corrosion and deficiencies identified with the maintenance activities, the team reviewed past surveillances and work orders. The team noted that operations and electrical maintenance personnel failed to follow battery related procedure requirements concerning identification of visual corrosion, use of deficiency tags and required resistance measurements during performance of the weekly and the quarterly battery procedures. Also, location of corrosion on battery cells and recording of required inter-cell resistance measurements of affected cells was not accomplished each time corrosion was identified on every corroded cell connections. Specifically:

- Procedure QCOS 6900-01,"Station Battery Weekly Surveillance," Revision 20, Step H.17.C.(1).(a) required that if any visual corrosion was identified, its location needed to be recorded on Attachment E. Step H.17.C.(1).(b) of the Procedure required that inter-cell resistance of affected cells be measured and recorded using Procedure QCEPM 0100-01,"Station Battery Systems Preventive Maintenance," Revision 22 and that the applicable resistance checklist(s) be provided to the Unit Supervisor for operations review. In addition, Step H.17.C.(1).(c) of the procedure required that if corrosion was found, then an Issue Report (IR) indicating the corroded cell number be initiated and the IR number be recorded on Attachment E. In addition, the team noted that Procedure WC-AA-106, "Work Screening and Processing," Revision 5, Attachment 5, "Use of Deficiency Tags," required that the IR originator fill out a deficiency tag and hang it on the equipment requiring maintenance activity.

  The team noted that on numerous occasions, between January 10 and June 6, 2006, the Unit 1 weekly surveillances were signed off as "satisfactory" and the corrosion columns in the procedure were signed off as "not applicable" even though previously identified corrosion had not been removed. Discussions with operations staff who performed the weekly surveillance procedure revealed that the operators were routinely only noted new corrosion; specifically, if a corroded cell or terminal was already listed on a CR or a deficiency tag, the operators had marked the corrosion columns as N/A on subsequent weekly surveillance procedures. For example, on January 10, 2006, operators identified corrosion on several cells and wrote a WO on Unit 1. However, on subsequent weekly surveillances, the operators did not note this corrosion even though it had not been corrected yet. The inspectors noted similar observations for the Unit 2 battery surveillance procedures for the period between March 3, 2006 to June 6, 2006.

- The licensee included the newly identified corroded cells into other existing, low priority, WOs for follow-up and action to remove the existing corrosion from the terminal connections. However immediate actions to address the newly identified corrosion were not always completed. For example, WOs 881144-01; 926556 01; 932102-01 and 930774-01 which documented previously identified

Unit 1 corroded battery cells were merged into WO 00883795 01, dated July 20, 2006. The team determined that WO 00883795 01 was assigned to the FIN team on January 10, 2006, for follow up and prompt corrective action, but corrective action to perform the corrosion cleaning activity and address this adverse to safety condition was not initiated until after it was identified by the team on August 15, 2006.

- Similarly, during review of completed Unit 2 WOs, the team determined that Surveillance WOs 00902418-01 (March 21, 2006); 00891313-01 (May 12, 2006); 00932102 (June 29, 2006) and 00934403-01 (July 4, 2006) were initiated to document identified Unit 2 corroded battery cell connections needing cleaning and were all merged into WO 00905552-01 (May 24, 2006). Also, at the time of the inspection, the team observed three deficiency tags hanging on Unit 2 125 Vdc batteries. Deficiency tags No. 151625 (March 21, 2006); 151870 (May 12, 2006) and 1973819 (July 4, 2006) were initiated to document corroded cell terminals. However, as of August 15, 2006 no action was initiated to clean these corroded cells. Subsequently, between August 15 and 29, 2006, WO 905552-01(Unit 2) and WO 748123-01 (Unit 2) were revised to require cleaning of all battery connections on the Unit 1 and Unit 2 battery cells and to record a full set of resistance measurements.

The team determined that the licensee's failure to follow the weekly and quarterly surveillance procedures and maintenance procedures resulted in the failure to take appropriate actions as required by Technical Specification Surveillance Requirement 3.8.4.2.

Upon discovery, the licensee's corrective actions included: initially cleaning some of, and later all battery 125 Vdc cell connections, taking connection resistance measurements, and initiating a root cause analysis (RCA) to identify the cause(s) of this adverse to quality condition. The licensee entered the finding into their corrective action program as IR 00530544, IR 00520627, and IR 00521252. The licensee's preliminary review of this issue concluded that since the cleaning of the corroded cells was completed within 24 hours of discovery, the issue was not reportable under 10 CFR 50.72 and the batteries were operable. The licensee also performed an audit of a sample of 250 deficiency tags to determine whether work requests (WRs)/(WOs) were issued to correct the deficiencies identified. The audit identified 54 deficiency tags of which the WR/WO were either complete or canceled but the tags were still hanging in the field. The audit also identified 8 deficiency tags with no associated WR/WO.

A week after the exit on September 22, 2006, during performance of Unit 1 125 Vdc station safety-related battery quarterly surveillance QCOS-6900-02, the licensee initiated IR 00534947 to document corrosion identified on battery cells 35-, 37-, 39-, 41-, 43-, 45-, and 52-. On September 23, 2006, during performance of Unit 2, 125 Vdc station safety-related battery quarterly surveillance QCOS-6900-02, IR 00534947 was initiated to document corrosion identified on battery cells 8+, 16+, 17-, 19+, 22-, 50+, 57-, 58+ and 48+. (Note that the Unit 1 and Unit 2 battery cell terminal connections had been previously cleaned in August 2006 to remove all visible corrosion.)

Analysis: The team determined that the failure to follow battery procedure requirements and TS SR and verify that no visible corrosion exists at the safety-related 125 Vdc terminal connections or verify that battery connection resistance of corroded terminals is within acceptable limits was a performance deficiency and a finding warranting a significance evaluation. Corroded inter-cell connections and post connectors can fail when exposed to the discharge current. As the battery ages (17 years old at Quad Cities), terminal post corrosion is a common problem that must be corrected by periodic checking and cleaning.

This finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Disposition Screening,” in that the finding was associated with the attribute of procedure quality and equipment performance and affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the DC power system to respond to initiating events to prevent undesirable consequences. Specifically, visual corrosion on terminal connections and failure to verify battery inter-cell and terminal connections resistance values, could potentially result in unacceptable battery terminal connection resistance and decreased battery capacity, rendering the DC system incapable of performing its required safety function.

The team evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per GL 91-18, did not represent an actual loss of a system’s safety function, did not affect external event mitigation. The team determined that the licensee failed to follow procedure requirements and consequently had exceeded the TS allowed surveillance periodicity for verifying corrosion or measuring resistance on 125 Vdc battery cell connections. However, this did not result in battery inoperability based on subsequent resistance measurement results. The team determined that the DC system would have performed its design function as determined by the licensee’s condition evaluation.

The cause of the finding was related to the cross-cutting aspect of human performance because operations, engineering and maintenance personnel failed to follow procedure requirements during performance of weekly and quarterly surveillances, maintenance procedure activities, and system engineering system walkdowns due to a mind set that corrosion did not impact battery operability.

Enforcement: Technical Specifications Surveillance Requirements 3.8.4.2, Amendment 199/195, required that, every 92 days, licensee verify that no visible corrosion exists at the battery terminals and connectors OR verify that battery connection resistance is $\leq$1.5E-4 ohm for inter-cell and terminal connections.

Contrary to the above, on September 11, 2006, the team identified that the licensee exceeded the 115 days (92 + 25 percent grace period) time period allowed for performing TS SR 3.8.4.2 on Units 1 and 2. Specifically, the during the quarterly and weekly surveillances the licensee failed to follow procedure and TS surveillance requirements to identify, document and remove existing visible corrosion on battery cell terminations OR measure the connection resistance of the corroded terminations, as required by the TS SR for the following periods: on Unit 1, 125 Vdc battery between January 10 and June 6, 2006 (a period of 147 days), and between November 16, 2004 and April 29, 2005 (a period of 164 days); and on the Unit 2, 125 Vdc battery between February 4 and December 5, 2005 (a period of 304 days). However, because the violation was of very low safety significance, and the licensee entered the finding into their corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-01(DRS)). The licensee entered the finding into their corrective action program as IR 00530544, IR 00520627, and IR 00521252.

.2 **Battery Connection Resistance Value Specified in TS SRs Insufficient to Ensure Operability**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving the failure to verify and ensure that the 125 Vdc safety-related batteries would remain operable if all the inter-cell and terminal connections were at the resistance value (≤ 150 micro-ohms) allowed by TS SR 3.8.4.2. and TS SR 3.8.4.5.

Description: During the investigation of the corrosion on the Unit 1 and Unit 2, 125 Vdc batteries, the team noted that Procedure QCEPM 100-01, “Station Battery Systems Preventive Maintenance,” Revision 22, had different standards for the safety and non-safety-related batteries allowed connection resistance. For the non-safety-related batteries, the procedure required following the manufacturers specified requirements of ensuring that no connection could exceed the baseline average value plus 20 percent (about 33 micro-ohms) without evaluation for being degraded; whereas, the safety-related battery connections were only required to be less than or equal to 150 micro-ohms. Since the baseline resistance value for the non-safety-related 125 Vdc batteries was about 5 times less than the TS allowed value, the team questioned the basis for the less conservative acceptance value for the safety-related batteries. The licensee informed the team that the 150 micro-ohm value had been adopted in the mid 1990s when the Quad Cities Technical Specifications were upgraded to the standard improved technical specifications. The Quad Cities submission for the proposed technical specifications placed the discussion in Section M.3 of “TECHNICAL CHANGES - MORE RESTRICTIVE,” which stated:

> “The CTS 4.9.C.2.b and 4.9.C.3.c provisions which allow the battery terminal and connector resistance to be ≤ 20 percent above the baseline connection resistance is not being retained in ITS 3.8.4. This allowance is an alternative to demonstrating that the measured battery terminal and connector resistance is ≤ 150 X 10-6 ohms and is not needed to ensure battery OPERABILITY. The ≤ 150 X 10-6 ohm limit is based on the battery manufacturer’s recommendations. This change deletes the alternative to meeting the 150 X 10-6 ohm battery terminal and connection resistance limit and establishes requirements consistent with IEEE-450 recommendations and BWR ISTS, NUREG-1433, Revision 1. As such, this change is considered more restrictive.”

The team considered the change to a larger acceptance value to be less restrictive and requested the licensee to quantify the available design margin related to increase inter-terminal resistance. Specifically, the team requested the effects of resistance increases to 150 micro-ohms as well as increases to 20 percent above baseline average resistance. The licensee’s response of August 31, 2006, stated:

"Per UFSAR Section 8.3.2, the battery terminal voltages are not allowed to drop below 105 Vdc. During the latest service test for the Unit 2 125 Vdc battery, the lowest voltage recorded was 107.2 volts with a baseline inter-cell reading of 27.2 micro-ohms (Ref. WO 572075-02). This leaves a voltage margin of 2.2 Vdc. The Unit 2 battery was chosen since it has the least margin and is limiting."

The licensee concluded that only 41.9 of 60 (~70 percent) inter-cell connections could reach the 150 micro-ohm limit before the available margin (2.2 Vdc) was used. However, 196 inter-cell connections – greater than three times the calculated actual connections - would have to be at 33 micro-ohms (baseline plus 20 percent) to reduce the margin to zero. The licensee entered IR 00534101, "Basis for Battery Inter-Cell Resistance in Tech Spec," dated September 14, 2006, into the corrective action program to further assess the 150 micro-ohm inter-cell connection value.

On October 13, 2006, the team was informed that the licensee performed the required battery inter cells resistance calculation to verify the TS specified value. As a result, IR 00543848, "Non-Conservative TS SR For Battery Inter-cell Resistance," had been generated and compensatory measures had been commenced to ensure safety-related battery operability by declaring the 125 Vdc batteries inoperable if any inter-cell resistance exceeds 70 micro-ohms. The licensee determined that the value of 70 micro-ohms was appropriate in calculation EC 262983, Revision 000. The team reviewed the EC and agreed with the licensee's interim actions. Further evaluation is needed to determine the final resistence value.

Analysis: The team determined that licensee's failure to perform the required calculation and verify that the resistance value ($\leq$ 150 micro-ohms) specified in TS SR 3.8.4.2 and TS SR 3.8.4.5 was sufficient to ensure safety-related battery operability was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because if left uncorrected, the finding could become a more significant safety concern. Specifically, the 125 Vdc safety-related batteries would become incapable of meeting their design basis if the inter-cell and connection resistance were allowed to increase to the TS allowed value.

The team evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because the team answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the Mitigating Systems column.

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 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. It further states that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, from the mid 1990s until October 12, 2006, the licensee failed to verify by calculation or design review that Technical Specification SR 3.8.4.2 and TS SR 3.8.4.5 specified battery inter-cell and terminal connection resistance value was sufficient to ensure plant safety. Specifically, the licensee failed to verify that the use of 150 micro-ohms criteria would be sufficient to ensure safety-related battery operability in accordance with the design basis. However, because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-02(DRS)). The licensee entered the finding into their corrective action program as IR 00543848, IR 00534101and IR 00540524.

.3 **Calculation Input Design Data Discrepancies for the Auxiliary Power Analysis and EDG Loading**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving calculation input design data discrepancies in Calculation QDC-6700-E-1503, Revision 1, dated May 15, 2006, "Auxiliary Power Analysis," and Calculation 9390-02-19-3, Revision 3A, dated February 12, 1999, "Diesel Generator 1/2 Loading Under Design Basis Accident Conditions" for the 1/2 EDG loading.

Description: Calculation QDC-6700-E-1503, which utilized Electrical Transient Analyzer Program (ETAP) Power Station, was identified by the licensee to be the calculation of record for the plant auxiliary power analysis. The calculation was updated on Revision 1 to include the analysis for the EDG's.

Calculation 9390-02-19-3 was identified by the licensee as the calculation of record for the Unit 1/2 EDG Design Basis Accident (DBA) loading and was used to provide design input to Calculation QDC-6700-E-1503. Revisions 2 and 3 of Calculation 9390-02-19-3 each revised the 1/2 EDG peak capacity ratings. The rating was changed on calculation Revision 2, from 3500 to 3560 kVA (and from 2800 to 2850 kW), but the subject data in Input Data, Item B, EDG nameplate data was not updated to 3560 kVA and a value of 3500 kVA remained incorrectly stated. Revision 3 of the calculation revised Item B for the generator peak rating from the incorrectly stated value of 3500 kVA (it should have been 3560 kVA per calculation Revision 2) to 3575 kVA.

UFSAR Section 8.3.1.6.1 stated the EDG's are rated 3250 kVA @ 0.8 pf, 2600 kW continuous, and have a 2000-hour/year rating of 2860 kW (3575 kVA @ 0.8 pf).

The team identified that Input Data Item B did not agree with the Unit 1/2 EDG nameplate data for the rated full load current (at 4160 volts), rated kVA, rated KVA and peak rating KVA temperature rise, and for excitation volts and amperes. The team found that although the generator nameplate was changed on September 30,1997, calculation Reference 38 for the generator nameplate data, Reference 68 for generator characteristics, and Input Data Item B (with the exception of the peak 3575 kVA rating), were not updated when the calculation was revised on February 12, 1999.

Calculation 9646-04-19-1, Revision 0, dated June 21, 1995, "ELMS-AC Load Data for the Emergency Diesel Generator" was identified by the licensee as the calculation of record for generator parameter design input data for different machine ratings for the EDG loading Calculation 9390-02-19-3. The team found that the Unit 1/2 EDG design data in calculation 9646-04-19-1 was evaluated for a 3500 kVA rating and the Units 1 and 2 EDG's were evaluated for a 3575 kVA rating. The calculation determined a X/R value of 29 for each generator.

The team questioned which generator data was used as design input (e.g., impedance, X/R, short circuit time constant, etc.) in the ETAP Auxiliary Power Analysis calculation for the Units 1, 2 and 1/2 EDG's. The licensee identified that calculation QDC-6700-E-1503, design input Section 5.25, contained the generator design data used in the ETAP program and that calculation 9390-02-19-3, Revision 3, was a source of data for Section 5.25. The team found the Section 5.25 data to be inconsistent with the Unit 1/2 EDG generator nameplate data, apparently since calculation 9390-02-19-3 also contained similar incorrect design data, as follows: rated current (434 amperes was used versus 452 amperes on the generator nameplate); rated kVA (3125 was used versus 3250 on the generator nameplate); excitation volts and amperes (137 and 98 respectively were used versus 144 and 100 respectively on the generator nameplate); rated stator temperature rise (65 degrees Celsius by thermometer versus 85 degrees Celsius on the generator nameplate); and stator temperature rise for peak kVA (80 degrees Celsius by thermometer versus 105 degrees Celsius on the generator nameplate). The team also found that the value of 1.82 per unit in Section 5.25 that was used for direct axis synchronous reactance was not consistent with the value of 1.83 per unit in Calculation 9390-02-19-3, Reference 66, for a 3575 kVA generator. The Unit 1 EDG rated current (425 amperes was used versus 452 amperes on the generator nameplate) was also found to be incorrect in Section 5.25.

Furthermore, the team found that some generator design input data that was identified in Section 5.25 was not correctly entered as the input data into the ETAP analysis. An X/R of 30 was entered as input into ETAP versus the X/R of 60 in the Section 5.25. When questioned by the team as to the basis for the X/R of 60, the licensee acknowledged that X/R of 60 was an erroneous value taken from the wrong reference in a previous calculation. In addition, the licensee stated that the X/R of 30 was entered into ETAP during sensitivity and case studies during the development of the model for the EDG systems and was inadvertently left in the analysis as an input. It should be noted that both of the X/R values in the ETAP analysis (X/R of 30 and 60) did not agree with the X/R data that was determined in the identified calculation of record, Calculation 9390-02-19-1, which determined an X/R of 29 for the generators.

Preliminary study cases by the licensee during the inspection determined that the change in X/R from 30 to 29, which was the original design input, resulted in the maximum load for Units 1, 2 and 1/2 EDGs (aligned to Unit 1) to increase by approximately 1 kW, which decreased the load margin. However, upon further review by the licensee, the design input X/R of 29 was changed to 33.33 in the ETAP calculation, based on vendor information, and the diesel loading decreased by approximately 4 kW, which increased the load margin.

Several other items of ETAP calculation design input data were also found to be incorrectly entered as input to the ETAP program. The percent efficiency data for the Unit 2 EDG was incorrectly inputted into ETAP as 100 percent versus the correct value which was 97.2 percent. The number of poles and speed for the Unit 1/2 EDG were also incorrectly entered as 4 poles and 1800 rpm, whereas the correct values were 8 poles and 900 rpm.

Analysis: The team determined that licensee's failure to perform adequate reviews of design input data in Auxiliary Power Analysis and EDG loading calculations was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of design control, which affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the EDG's to respond to initiating events to prevent undesirable consequences. Specifically, use of incorrect design input data, could result in inaccurate electrical systems margin.

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

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 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, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, during this inspection, the team identified that on May 15, 2006, during the Auxiliary Power Analysis calculation revision of calculation QDC-6700-E-1503, Revision 1, the licensee failed to perform an adequate design review of the design input data for the Units 1, 2 and 1/2 EDGs to ensure that proper design control was maintained. Specifically, the licensee failed to adequately identify design data and to perform an adequate design review of the vendor nameplate data for the Unit 1/2 EDG, and also had incorrectly identified and reviewed design input data for the Units 1, 2 and 1/2 EDGs that was used in the ETAP auxiliary power analysis, which resulted in an inaccurate determination of EDG loading. However, because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-03(DRS)). The licensee entered the finding into their corrective action program as IR 00521503, IR 00521248, and IR 00526373.

.4 **Licensee Used Inappropriate Vortex Analysis Methodology**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) contaminated condensate water storage tank (CCST) volume's design analysis. Specifically, the team identified that the licensee failed to select an appropriate method for calculating the onset of vortexing at the intake of the HPCI and RCIC pumps' suction lines from the CCST water storage tank. Additionally, the licensee failed to fully account for the impact of instrument uncertainty in the tank level switch setpoint which determines the point where suction for the pumps is switched from the CCST to the torus.

Description: The team reviewed Calculation QDC-3300-0489, "Usable Water Volume of Contaminated Condensate Storage Tanks for HPCI and RCIC, Including Vortexing Considerations," Revision 2. The purpose of the calculation was to determine the appropriate analytical level (i.e., elevation of water) where vortexing would occur above the HPCI and RCIC pumps' suction lines. The analytical level was adjusted by 0.25 inches of instrument uncertainty, then was used as a design input to calculate the allowable time for automatic CCST storage tank to suppression pool low level suction switchover for the HPCI and RCIC pumps.

There are numerous methodologies available to calculate the minimum submergence level associated with pumps, primarily based on correlations of experimental data. It is important to find a methodology that best suits the geometric configuration or arrangement of the intake design, and the manner by which the submergence is calculated.

The team determined that the methodology used in Calculation QDC-3300 - 0489 to determine the minimum height of water above the HPCI and RCIC pump's intake lines to preclude vortex formation was not appropriate. The calculation's methodology did not account for the actual fluid configuration with respect to the suction piping submergence where air ingestion into the HPCI and RCIC pumps' suction lines would potentially occur. The onset of vortexing was calculated using a methodology contained in NUREG/CR-2772, "Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors," June 1982. The method was not appropriate because:

- Critical height of submergence developed from the test data in the subject NUREG calculated the intake Froude number (Fr) as:

  (Fr) as $Fr = v/(g*s)^{½}$

  where (v) is the fluid velocity in the intake piping, (g) is gravitational constant, and (s) was defined in the subject NUREG as the distance from the inlet pipe centerline to the water surface. When calculation QDC-3300-0489 calculated the Froude number, it defined the submergence as the distance from the top of the pipe to the water surface. Although this was stated as a conservatism by the licensee, the team determined that it was not consistent with the test conditions. This is because the calculated submergences from calculation QDC-3300-0489 indicated that, if applied in accordance with the similarity to the subject NUREG testing, would result in the inlet piping being uncovered. In other words, the piping would be partially filled with air. This condition was not tested for predicting the onset of vortexing in the subject NUREG because there was at least 2 feet of submergence.

- When determining the onset of vortexing, the subject NUREG used test results that were based upon a constant tank level of at least 2 feet of submergence, with a varying discharge flow rate. This differed from the conditions in the Quad Cities CCST, which had only a few inches of submergence, a constant discharge flow rate, and a draining tank. Since the prediction of vortex formation is based on similarity to model testing, the controlled conditions of the test must closely match the condition where the test data was being applied.

In addition to the inappropriate selection of vortex methodology, Calculation QDC-3300-0489 did not correctly apply the affects of instrument uncertainty to the CCST level switch setpoint. The calculation applied an instrument uncertainty of 0.25 inches to the analyzed vortex level when determining the required setpoint of the CCST low level switches (LS 1/2-2350-A/B/C/D). The licensee informed the team that this was based upon the setting tolerance for the level switches. The team reviewed calculation number QDC-2300-I-0964, "HPCI/RCIC Level Switch Setpoint Error Analysis," Revision 0, which determined the instrument uncertainty for the level switches. The calculation determined that the uncertainty in tank level was 1.27 inches. Therefore, the vortex calculation (QDC-3300-0489) should have used 1.27 inches uncertainty, instead of 0.25 inches. The team concluded that the analyzed vortex level was non conservative by 1.02 inches.

The team asked the licensee to provide technical justification for their use of the test data from the subject NUREG to predict the onset of vortexing. The licensee provided an evaluation of why they thought the vortex methodology was appropriate, but the team determined that it was not appropriate due to the constant level and filled pipe used in the NUREG testing, versus the draining tank and redefining of submergence in the configuration of the CCST at Quad Cities.

The licensee stated they would consider other methods applicable to this configuration that were more readily accepted by the industry. The licensee chose a method for predicting the onset of vortexing that was based upon test data of a partially filled horizontal suction line. This method is only applicable at a low intake Froude number (around 0.5), which closely matches the intake Froude number at the CCST intake for the HPCI and RCIC pumps at Quad Cities. The licensee entered the finding into their corrective action program as IR 00524923 to identify the methodology and instrument uncertainty concerns with QDC-3300-0489 and to track update of the calculation. The licensee revised the calculation to incorporate the new methodology and instrument uncertainty and determined that vortexing would not occur. The team reviewed the revised calculation and had no further concerns.

<u>Analysis</u>: The team determined that failure to select an appropriate method for calculating the onset of vortexing at the intake of the HPCI and RCIC suction lines from the CCST was a performance deficiency warranting a significance evaluation. The team also determined the instrument uncertainty for the level switches was not properly applied in calculation QDC-3300-0489, and also was a performance deficiency warranting a significance evaluation. The team concluded that the finding was greater than minor because it was similar to example 3j of Appendix E in IMC 0612, "Power Reactor Inspection Reports." Specifically, the calculation of record was not correct and

there was reasonable doubt of the successful outcome of a re-analysis. This finding affected the Mitigating System cornerstone.

The team completed a significance determination of this finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At - Power Situations.” The team answered “no” to all five screening questions in the Phase 1 Screening Worksheet under the Mitigating Systems column. Based on subsequent calculations, the team agreed with the licensee's position that the HPCI and RCIC systems would have performed their safety functions. Therefore, the team concluded that the finding did not represent an actual loss of a safety function and the finding screened out as having very low safety significance or (Green).

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures shall be established for the selection and review for suitability of application of processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, as of September 15, 2006, the licensee failed to select and review for suitability an appropriate method for calculating the onset of vortexing at the intake of the HPCI and RCIC suction lines from the CCST, and failed to properly apply instrument uncertainty to the calculated vortex level. Specifically, Calculation QDC-3300-0489, “Usable Water Volume of Contaminated Condensate Storage Tanks for HPCI and RCIC, Including Vortexing Considerations,” Revision 2 used a method that did not account for the actual fluid configuration where air ingestion into the HPCI and RCIC pump’s intake would potentially occur and instrument error uncertainty of 0.25 inches was used instead of 1.27 inches. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy(NCV 05000254/265/ 2006003-04(DRS)). The licensee entered the finding into their corrective action program as IR 00524923 to revise the affected calculation.

.5 **Non-conservative Sizing Calculation for ADS/SRV Air Accumulator Storage Tank**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving the sizing calculation for the Target Rock ADS/SRV air accumulator tank. Specifically, the team identified that the licensee failed to correctly specify the minimum differential air pressure required to actuate the ADS/SRV valves, failed to include the volume of the piping from the solenoid to the ADS/SRV actuator, and had the wrong assumption for leakage rate used as acceptance criteria in air drop testing.

Description: The accumulator on the Target Rock ADS/SRV valve provides pneumatic pressure, and volume, for valve actuation, on a loss of normal pneumatic supply. There is one Target Rock ADS/SRV per unit. The other four ADS/SRV valves per unit are DC operated electromatic relief valves (ERVs), and do not require pneumatics. The team reviewed Calculation number NUC-60, “Accumulator Air Leakage for SO-203-3A,” Revision 0, whose purpose was to determine the air volume and pressure required in the storage tank for the Target Rock ADS/SRV air actuators. Specific problems with the calculation included:

- The calculation provided documentation that the Target Rock Safety Relief Valve (SRV), 1(2)-0203-3A, will be capable of actuating for at least 30 minutes after the loss of instrument air supply after performing five actuations. The calculation assumed that drywell pressure remained constant between 0 to 2 psig and the drywell temperature remained at 334 degrees F. The actual design basis of the accumulator is to perform five actuations at atmospheric drywell pressure after a 1 hour loss of drywell pneumatic supply pressure. This corresponds to two actuations at a drywell pressure of 70 percent of drywell design pressure.

- The calculation assumed that the accumulator length was 18 inches. Actual walkdowns by the licensee have determined the length was 24 inches. This was conservative because the accumulator would have a greater capacity.

- The calculation assumed that the initial accumulator pressure was 90 psig with an assumed leakage rate of 1 standard cubic foot per hour (SCFH). Technical Specification Surveillance Requirement (SR) 3.5.1.12 required verification every 31 days that the ADS/SRV pneumatic supply header pressure is $\geq$ to 80 psig to ensure adequate pneumatic pressure for Target Rock ADS/SRV actuation. The current licensing basis is that the initial pressure will be 80 psig, and after 1 hour the accumulator pressure will be at least 70 psig. The one hour requirement at 70 psig is verified in IST of accumulator check valve. The 70 psig requirement is the minimum pneumatic pressure required to stroke the ADS/SRV valve, based upon previous laboratory testing of the SRV and accumulator. The 1 hour requirement was determined by the licensee to be adequate to depressurize the reactor in a small break loss of coolant accident (LOCA), since the subsequent maintenance of low RPV pressure will be adequately insured by the remaining four ERVs.

- The calculation did not credit the drywell pneumatic piping volume from the check valve to solenoid on the actuator. This was conservative because the accumulator would have a greater capacity.

- The calculation did not penalize for the volume of the piping from the solenoid to ADS/SRV actuator.

Based upon a preliminary calculation review performed by the licensee, the licensee determined that the calculation required revision to correct the problems that were identified by the team.

<u>Analysis</u>: The team determined that failure to correctly specify the minimum differential air pressure required to actuate the ADS/SRV valves, the failure to include the volume of the piping from the solenoid to the ADS/SRV actuator, and the failure to specify the correct leakage rate, based on periodic test acceptance criteria, was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue

Dispositioning Screening," because it was associated with the attribute of design control, which affected the mitigating systems cornerstone objective of ensuring the availability and reliability of the ADS/SRV valves to respond to initiating events to prevent undesirable consequences. Specifically, the errors identified in the sizing calculation could result in over-predicting the air storage tanks' performance (i.e., creating design margin capability that would not exist). This could potentially render the ADS/SRV valves incapable of performing their required safety function.

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

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 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, as of September 22, 2006, the licensee failed to assure that the ADS/SRV pneumatic accumulator minimum operability limits were correctly translated into specifications, drawings, procedures, and instructions. Specifically, calculation number NUC-60, "Accumulator Air Leakage for SO-203-3A" did not use the minimum air pressure requirements as specified in the plant TS's, it did not consider the volume of air lost between the of the piping from the solenoid to ADS/SRV actuator, and had the wrong assumption for leakage rate used in air drop testing. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-05(DRS)). The licensee entered the finding into their corrective action program as IR 00525397 to revise the affected calculation.

.6 **Discrepant MCC Voltages Used in Degraded MOV Voltage Drop Calculations (Power and Control Circuits)**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance for failure to ensure a control circuit calculation contained an appropriate input for the voltage for motor control center (MCC) 28-1B.

Description: While reviewing electrical design calculations for safety-related MOVs, the team noted that Calculation QDC-E-0000-E-0206, "Motor Terminal Voltage for Units 1 and 2 GL [Generic Letter] 89-10 MOVs," Revision 1, was used to determine the terminal voltage at the MOV's motor to ensure that the voltage was adequate for the motor to develop the necessary torque. The team noted that the assumed voltage at the motor control center, 28-1B was 429.5 volts.

The team also reviewed Calculation 8913-69-19-4, "Justification of the Adequacy of MCC Contactor Circuits Fed from Switchgears 19 and 28," Revision 0, and noted that the calculation determined the available voltage at the motor contactor to ensure that there was adequate voltage for the contactor to pick up. However, the team identified that the assumed voltage at the motor control center 28-1B was 435.5 volts.

The licensee stated that the two differing voltages for the same MCC were derived from two versions of Electrical Load Monitoring System, ELMS-AC, runs. The team noted that 429.5 V was derived from the 1996 version of ELMS-AC and 435.5 V was derived from the 1992 version of ELMS-AC. The team identified that the licensee failed to ensure that the control circuit voltage drop calculation for the safety-related MOVs was updated to reflect the latest available design information for the available voltage at the MCC (ELMS-AC 1996 version) so that the correct voltage available for motor contactor pick up could be ensured.

Subsequently, the licensee determined, based on review of other electrical design calculations, that the affected circuits will have adequate voltage to ensure proper function of the valves components. The licensee initiated IR 00526361 and planned to formally revise several existing calculations to confirm correct voltage is available to ensure continued operability of the safety-related MOVs.

<u>Analysis</u>: The team determined that the failure to use proper and most current design input for the calculation was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of design control, which affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the safety-related MOVs to respond to accident conditions. Specifically, the licensee failed to verify and ensure that the control circuit voltage drop calculation for the safety-related MOVs was revised to reflect the latest design parameters for control circuit components available voltages, so that the affected circuits have adequate voltage to ensure proper function of the control circuit components of the valves . Subsequently, on September 1, 2006, the licensee determined, based on review of other electrical design calculations, that the affected circuits will have adequate voltage to ensure proper function of the valves components.

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

The team concluded this finding did not have a cross-cutting aspect.

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

Contrary to the above, the licensee failed to maintain adequate design control concerning safety-related MOVs voltage drop calculations. Specifically, in Calculation 8913-69-19-4, Revision 0, the licensee incorrectly used 435.5 volts for the assumed voltage at MCC 28-1B and should have used the more conservative calculated value of 429.5 volts. However, because this violation was of very low safety significance and because the issue 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 Enforcement Policy (NCV 05000254/265/2006003-06(DRS)). The licensee entered the finding into their corrective action program as IR 00526361 to revise the affected calculations.

.7 Inadequate Load Tabulation in Operations Procedure QCOP 6500-28

Introduction: The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” having very low safety significance (Green), for failing to maintain adequate procedures/instructions to establish the load on Bus 24-1 during crosstie operations with Bus 14-1.

Description: The stated purpose of procedure QCOP 6500-28, “4KV/480V Bus Loading Profiles,” Revision 0, was to provide reference loading values (current or KW) for 4KV/480V distribution system loads. The loading values may then be used in conjunction with procedures that place the electric plant in an off-normal lineup (crosstied busses, etc.) to ensure that bus overload, breaker trips on overcurrent, and/or Diesel Generator loading limits are not exceeded. UFSAR Section 8.2.1.2, Switchyard, discusses the design capability of the 4KV Division I and II crossties. Specifically, for the crosstie between buses 14-1 and 24-1, the UFSAR states that, “The crosstie between buses 14-1 and 24-1 has differential relays for cable fault protection” and that “Overload protection is also provided by administrative control which limits the crosstie load to 600A.”

Procedure QCOP 6500-28, Section B.4, stated that, “The maximum loading values provided in the attachments of the procedure include adequate margin to prevent overloading of bus supply breakers and take into account cycling loads.” The maximum breaker loading criteria that was identified in the procedure attachments was 580 amperes (A). The licensee stated that this limit was based on equipment protection, to ensure that the rating of the current transformers (CT’s) was not exceeded. Therefore, the overload protection for the limit on the maximum loading on the breakers and their respective CT’s, during the crosstie condition between buses 24-1 and 14-1, was provided by administrative control through the use of Attachments in the subject procedure. Should the procedure’s administrative controls fail to provide adequate control of the breaker loading during a bus crosstie, electrical equipment (e.g., CT) damage or failure due to the overcurrent/overload condition could result.

The team noted that Attachment H, of procedure QCOP 6500-28, which determined the load on Bus 24-1 when the bus is cross connected to Bus 14-1, did not include a row in the Bus 24-1 load tabulation for the operator to enter the expected load data (kW and running current) from Bus 14-1. The procedure was used to determine the load on Bus 24-1 and to verify that the expected load was less than the bus feed breaker maximum loading during bus crosstie conditions. The procedure was also used to help in predicting the kW load under procedure QCOP 6600-17, "Unit 2 Diesel Generator Simultaneous Supply to Buses 24-1 and 14-1," Revision 5, when in accordance with that procedure the 2 EDG supplies the cross connected buses, "in Operational Mode 4, 5 and none," for Units 1 and 2.

The team noted that in Attachment H, of procedure QCOP 6500-28, the tabulated load on Bus 24-1 for the running current for the pumps and the Transformer 29 Feed can be as high as 385 amperes. The team also noted that the crosstie load that was not included in the Bus 24-1 load tabulation, for the load from Bus 14-1 that is determined in procedure Attachment D, can be 355 amperes (or more) based on the tabulated running current for the pumps, Transformer 19 Feed and the gatehouse feed listed in the attachment. Therefore, for the subject bus tie condition the Bus B-24 to Bus B-24-1 feed breaker load could be as high as 740 amperes (385 + 355 amperes), which would exceed the licensee's administrative limit of 580 amperes for the maximum bus feed breaker loading limit and which could result in equipment damage.

Analysis: The team determined that the failure to include a step in the procedure to address the load from the crosstie bus was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of procedure quality, which affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to evaluate the load from the crosstie bus could result in overloading the Bus 24-1 feeder breaker and impact the performance equipment.

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

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions.

Contrary to the above, procedure QCOP 6500-28, Attachment H, used in part, to determine the load on Bus 24-1 during crosstie operation with Bus 14-1 was inadequate in that it did not include a procedure step for entering the expected load data (kW and running current) from crosstie Bus 14-1 into the load tabulation and thereby ensure that the expected load is less than the bus feed breaker maximum loading during bus crosstie conditions. Because this violation was determined to be of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000254/265/2006003-07 (DRS)). The licensee entered the finding into their corrective action program as IR 00521012.

.8 **Inconsistency in Procedures for Cleaning Batteries**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," having very low safety significance (Green) for the 125 Vdc batteries surveillance procedures being discrepant from vendor specific instructions and other station procedures. Specifically, the procedures stated that "if electrolyte is spilled on batteries, then use only demineralized water for cleaning." This differed from vendor's specific instructions which stated that electrolyte spill should be neutralized with baking soda water solution.

Description: During a review of Procedures QCOS 6900-01, "Station Battery Weekly Surveillance," Revision 20, QCOS 6900-02, "Station Battery Quarterly Surveillance," Revision 23, and QCOS 6900-14, "Station Battery Allowable Value Verification Surveillance," Revision 10, the team identified an inconsistency between the procedures with respect to cleaning electrolyte spill on 125 Vdc safety-related batteries. Section F.5 of the weekly and quarterly surveillance procedures and Section F.4 of the allowable value surveillance procedure, stated that "if electrolyte is spilled on batteries, then use only demineralized water for cleaning the batteries. Do not use any other cleaning solutions on batteries." However, the team noted that procedure QCEPM 0100-01, "Station Battery Systems Preventive Maintenance," Revision 22 stated in Sections 3.1.7 and 4.1.1.B that only a baking soda water solution should be used to clean electrolyte spilled on battery cells. The team also noted that IEEE 450-1975 (which the plant is committed to) also stated that only a baking soda water solution should be used to clean spills on batteries. Section 18.10 of the battery vendor manual required electrolyte spills on batteries be neutralized with baking soda water solution and this activity shall be continued until fizzing action ceases.

The team determined that using only demineralized water to clean the electrolyte spill will not ensure the neutralization of the electrolyte and could lead to undesirable consequences including corrosion and intermittent grounds on the battery system. During walkdown of both the units' batteries the team identified numerous instances corrosion on batteries which could be attributed to improperly removing spilled electrolyte. In addition, the team reviewed IRs that documented numerous instances of 125 Vdc system grounds on both Unit 1 and 2 125 Vdc batteries, in the past several years.

Analysis: The team determined that the failure to follow vendor's specific instructions, which resulted in inconsistent requirements in battery surveillance procedures, was considered to be a performance deficiency and a finding. Specifically, vendor's specific instructions required that baking soda water solution be used to clean electrolyte spill on batteries whereas the surveillance procedures required the use of only demineralized water. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Disposition Screening,” in that the finding was associated with the attribute of procedure quality and equipment performance and affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the DC power system to respond to initiating events to prevent undesirable consequences. Specifically, failing to neutralize spilled battery acid could lead to undesirable consequences for the battery and could potentially affect the battery’s design function.

The team evaluated the finding using IMC 0609, Appendix A,“Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per GL 91-18, did not represent an actual loss of a system’s safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. In addition, although the licensee was not following the vendor instructions for cleaning electrolyte spill on batteries, the station batteries were capable of performing their required safety function as evidenced by the availability of the required battery terminal voltage.

The team determined the finding was not related to a cross-cutting aspect.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with the procedures of a type appropriate to the circumstances and shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, the batteries weekly, quarterly and allowable value surveillance procedures failed to incorporate specific vendor’s instructions concerning use of baking soda water solution instead of only demineralized water to clean electrolyte spill on batteries. Because the violation was of very low safety significance and the licensee entered the finding into their corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-08(DRS)). The licensee entered the finding into their corrective action program as IR 00525492.

.9 **Failure to Comply with Preventive Maintenance Procedure Requirements Concerning Re-Torquing of Corroded Electrical Terminal Connections**

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” having very low safety significance (Green) for failure to follow the station battery systems preventive maintenance procedure for 125 Vdc safety-related battery. Specifically, the licensee failed to comply with a requirement in the procedure not to re-torque corroded battery connections Additionally, the team noted that the as-left re-torque values were not documented as required by the procedure.

Description: During a walkdown of the Unit 1, 125 Vdc safety-related battery, the team identified corrosion on multiple inter-cell connections of the battery. Upon raising this issue with the licensee, the licensee cleaned some of the corroded cells. After cleaning, the licensee measured the resistance values of the inter-cell connections and the end plate connectors. Some of the measured resistance values were outside the acceptable range (recommended by the Vendor) and the electricians re-torqued the battery cell to cable plate connectors twice in an attempt to achieve acceptable resistance values. However, the resistance values did not come down.

The team noted that Precaution Step 3.1.5 of Procedure QCEPM 0100-01,"Station Battery Systems Preventive Maintenance," Revision 22 stated that "if battery connectors are corroded, then do not re-torque." Step 4.5 of Procedure QCEPM 0100-01stated that "if corrosion is found between contact surfaces, then an Issue Report (IR) should be written to disassemble connections, clean and re-torque." In addition, Section 19.2 of vendor manual C0004, specifically cautioned against too frequent re-torquing and stated that "too frequent re-torquing of connections is not recommended as this will result in distortion of cell posts, connectors, etc. thus degrading rather than improving the connection." The vendor manual further stated, "re-torquing should not be done if visual inspection shows evidence of corrosion. Re-torquing when corrosion is present only restores mechanical compression but will not improve electrical integrity. Just cleaning the corrosion on and around the connectors does not guarantee that there is no corrosion between the battery post and the connector."

The inspectors concluded that the electricians failed to follow step 3.1.5 when they re-torqued the battery cell to cable plate connectors after cleaning corrosion. The inspectors also noted that the electricians failed to document the as-left re-torque values, as required by the procedure. As a result of the team raising this issue, the licensee's immediate corrective action included measuring the resistance values after cleaning the terminal corrosion and initiating an IR to investigate and evaluate the effects of re-torquing on the battery inter-cell resistance values. The team determined that the resistance readings were high relative to the allowable values provided by the vendor; however, the values were still within the Technical Specification specified value.

Analysis: The team determined that the failure to follow the maintenance procedure with respect to re-torquing connectors was considered a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," in that, the finding was associated with the attribute of procedure quality and equipment performance and affected the Mitigating Systems cornerstone objective of ensuring the availability and reliability of the DC power system to respond to initiating events to prevent undesirable consequences. Specifically, frequent re-torquing could potentially damage the battery cell posts and connectors by distorting them and thus degrading the electrical integrity of the battery resulting in potential impact on battery's design function.

The team evaluated the finding using IMC 0609, Appendix A,"Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per GL 91-18, did not represent an actual loss of a system's safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. Despite the failure to follow the procedure, the station batteries were capable of performing their required safety function as evidenced by the availability of the required battery terminal voltage.

This finding had a cross-cutting aspect in the area of human performance because the licensee failed to ensure that the written instructions (i.e., traveler) were consistent with other approved maintenance procedures.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be accomplished in accordance with the procedures and shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, on August 29, 2006, the licensee failed to follow Step 3.1.5 of Procedure QCEPM 0100-01,"Station Battery Systems Preventive Maintenance," Revision 22, when maintenance personnel re-torqued corroded battery electrical connections. Additionally, the team noted that the as-left re-torque values were not documented after re-torquing was performed. Because the violation was of very low safety significance and the licensee entered the finding into their corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000254/265/2006003-09(DRS)). The licensee entered the finding into their corrective action program as IR 525113.

.10 Non-Conservative HPCI Pump Test Acceptance Criteria

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, having very low safety significance (Green) involving the HPCI system technical bases for pump test acceptance criteria. Specifically, the team identified that the licensee failed to correctly specify the minimum pump's hydraulic operability limits to be used in surveillance testing of the HPCI system that would ensure the system's design basis requirements could be met.

Description: The team noted that in calculation number QDC-2300-0486, "Verification of HPCI Pump Discharge Flow to the Reactor," Revision 0, the design flow rate for the HPCI pump was 5600 gpm at a discharge pressure of 1189 psig. This flow rate included 5000 gpm to the reactor vessel and 600 gpm minimum flow. The team reviewed the surveillance testing performed on the HPCI system and determined that the existing acceptance criteria did not verify this design basis requirement.

The team reviewed the ASME Section XI pump testing acceptance limits to determine if the established limits would have ensured to pumps remained operable. The test procedure established initial conditions which held pump speed constant at 3500 RPM and discharge flow constant at 5600 GPM. After these conditions are established the measured differential pressure across the pump was recorded. The testing differential pressure low limit acceptable criteria for the HPCI pump was 804 psig. Using pump affinity laws, the team determined that the HPCI pump discharge pressure would be over 100 psi lower than the design basis discharge pressure requirement of 1189 psig at the maximum automatically controlled pump speed of 4000 RPM. Therefore, since the speed control can not exceed 4000 RPM, the team determined that if the HPCI pumps had degraded to this differential pressure, it would not have been able to meet the design basis discharge pressure and flow requirements during an accident.

The licensee performed an evaluation of recent HPCI pump test results using pump affinity laws and determined that the pump had adequate discharge pressure, flow and speed to meet design basis requirements as stated above. The team reviewed the evaluation and agreed with the licensee's conclusion.

Analysis: The team determined that failure to establish adequate acceptance criteria to ensure the HPCI pump was capable of performing its safety function was a performance deficiency and a finding. The performance deficiency associated with this finding was that the licensee did not set pump test acceptance criteria for the HPCI equipment that ensured they would be capable of providing the required design basis flow during accident conditions.

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 the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of the HPCI system. If the HPCI pump had degraded to the lower limit of the test acceptance criteria, it would not have been able to meet the design basis discharge pressure and flow requirements.

The team evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding was of very low safety significance (Green), because subsequent analysis showed that the HPCI system remained operable based on the actual results of previously performed surveillance tests.

The team concluded this finding did not have a cross-cutting aspect.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, 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, the licensee failed to translate the HPCI system flow and discharge pressure design requirements into surveillance test QCOS 23—27, "HPCI Pump Performance Test," dated September 23, 2001, acceptance criteria. Because this issue 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 05000254/265/2006003-10 (DRS)). The licensee entered the finding into their corrective action program as IR 525592.

.11 Non-Conservative Safety-Related Air Storage Tank Capacity Test

Introduction: The team identified a NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," having very low safety significance (Green) for failure to specify the technical bases for Target Rock ADS/SRV air drop test acceptance criteria. Specifically, the team identified that the licensee failed to correctly specify the minimum air pressure to be used in surveillance testing of the Target Rock ADS/SRV pneumatic accumulator that would ensure the system's design basis requirements could be met.

Description: On August 13, 2001, the licensee submitted a proposed TS change to support the transition to GE14 fuel and to support extended power uprate (EPU). The new TS added a surveillance requirement (SR) 3.5.1.12, to verify every 31 days that automatic depressurization system (ADS) pneumatic supply header pressure was > 80 psig. It was stated in the submittal that the > 80 psig pressure ensured that adequate nitrogen pressure is available for reliable Target Rock ADS valve operation (valves 1(2)-203-3A). The accumulator on the Target Rock ADS valve provides pneumatic pressure for valve operation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70 percent of design pressure. The ECCS safety analysis assumed only one actuation to achieve the depressurization required for operation of the low pressure ECCS. The submittal stated that the proposed SR verified that pneumatic supply header pressure was greater than 80 psig, and, together with a current Quad Cities In-Service Testing (IST) program periodic test, ensured that the accumulator remained pressurized to at least 70 psig for one hour following a loss of makeup to the accumulator. This test was performed each refueling outage. As part of the GE14 fuel transition, the licensee should have revised the IST test to ensure that the starting pressure for the test is no greater than 80 psig. The team noted that IST procedure number QCOS 4700-02, "Inboard MSIV and Target Rock Valve Pneumatic System Leak Test," Revision 2, had not been revised to ensure that the starting pressure for the test was no greater than 80 psig. A search of commitment tracking (Passport) by the licensee indicated that no action tracking assignments were initiated to track the IST program change.

The last performance of this test under WO 00598882-01 (Unit 1, April 13, 2006) and WO 00738139-01 (Unit 2, April 12, 2006) had starting pressures of 104 psig and 101.5 psig, respectively. Testing from a higher pressure than 80 psig would be technically acceptable and conservative as long as the pressure dropped less than 10 psig in 1 hour. The team noted the Unit 1 accumulator pressure dropped 29 psig to 75 psig. Had the starting pressure been 80 psig, the pressure drop would have been unacceptable (less than 70 psig.) Therefore, the Unit 1 pneumatic accumulator system appeared to be degraded. For Unit 2, the accumulator pressure only dropped 2 psig to 99.5 psig and therefore, was functional.

The licensee performed an operabililty evaluation of the Unit 1,Target Rock ADS/SRV valve, and concluded that the valve was operable. This evaluation was contained in document number EC 362300, "Unit 1 Target Rock SRV Accumulator Check Valve Degraded," dated August 30, 2006. The team reviewed the OE and agreed with the licensee's conclusion.

Analysis: The team determined that the failure to establish an acceptable pneumatic supply header pressure starting pressure during testing was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it affected the equipment performance attribute associated with the mitigating systems cornerstone as related to the availability, reliability, and capability of the ADS/SRV system. Specifically, the pneumatic accumulator test conditions did not ensure the Target Rock ADS/SRV equipment would be capable of providing the required design basis number of actuations 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, and determined that the finding was of very low safety significance (Green), because subsequent analysis, and an operability evaluation showed that the Target Rock ADS/SRV equipment remained operable based on the actual results of previously performed surveillance tests.

The team determined this finding did not have a cross-cutting aspect.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, 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 is identified and 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, the licensee failed to include the appropriate minimum pneumatic accumulator pressure in surveillance testing of the Target Rock ADS/SRV pneumatic equipment. Specifically, IST procedure number QCOS 4700-02, "Inboard MSIV and Target Rock Valve Pneumatic System Leak Test," Revision 2, did not require the licensee to verify the pneumatic supply header pressure starting pressure was less than or equal to 80 psig, as required by Technical Specification SR 3.5.1.12. Because this issue 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 05000254/265/2006003-11(DRS)). The licensee entered the finding into their corrective action program as IR 0052383 to revise the test procedure.

.12 **Shift Management Failed to Adequately Document Basis for Operability Determination**

Introduction: The team identified a finding of very low safety significance (Green) involving decision-making supporting operability determinations. Specifically, the team identified several examples where the basis for operability of the Unit 1 and Unit 2, 125 Vdc was not supported.

Description: During the inspection activities for the 125 Vdc safety-related batteries, the team reviewed the condition reports associated with corrosion on the Unit 1 and Unit 2 battery connections. The team noted that procedure LS-AA-105, "Operability Determinations," Revision 1, Section 3.2 stated: "Operations Shift Management is responsible for the determination of whether an SSC is operable." Section 4.1.5 of this procedure stated: "determine and document the operability status of the affected SSC in accordance with the site cap procedure."

The team reviewed seven IRs related to corrosion on the Unit 1 or Unit 2, 125 Vdc battery inter-cell or terminal connections with dates from 2004 through 2006 and noted that only one of the seven IRs had an appropriate basis for declaring the affected battery operable. This basis noted that the battery met all requirements for TS 3.8.6 (battery cell parameters), TS SR 3.8.4.1 (battery terminal voltage), and that electrical maintenance had been contacted to take connection resistance measurements (to meet TS SR 3.8.4.2 requirements). The shift manager further stated that the weekly surveillance would not be closed until the resistance checks were completed satisfactorily. Based on meeting these parameters, the shift manager concluded that the battery would meet its design function and was operable. The team considered this to be good documentation with clear logic and reasons to support the prompt operability call. In the remaining six IRs, the shift supervisors did not provide an appropriate basis or justification for operability. For example, the following justifications were provided:

- "Battery surveillance completed SAT [satisfactory]. EMD [electrical maintenance department] notified of corrosion." No note was made if the resistance checks were completed or the corrosion cleaned off;

- "Small amounts of corrosion does not affect safety function." This reason was stated on two different IRs; however, corrosion can degrade a connection and affect the battery safety function. A required connection resistance check was not performed;

- "Minor corrosion is not an operability issue addressed in Tech Specs." The team noted that TS 3.8.4.2 clearly address visible corrosion on the battery connections;

- "This IR relates to the U-1 125 battery. TS 3.8.4.4 discusses removal of all visible corrosion and verification that cell to cell connections are coated with anti-corrosion material every 24 months. This is what prompted the IR when the operator found visible corrosion. This does not make the battery inoperable." The shift manager quoted the wrong TS when dealing with identified visible corrosion and did not understand the failure mechanism from corrosion; and

- "Amount of corrosion will not impact battery performance." This conclusion was contrary to the manufacturer's and IEEE standard information. Furthermore, the shift manager did not give any basis for deciding that the battery was operable.

Based on the results of weekly and quarterly surveillance tests, the team did not have a concern for the actual operability of the batteries; however, the team considered the norm for shift management to document the basis for prompt operability calls to be very low. The licensee agreed with the concern and documented it in IR 00531359, "NRC I'D Insufficient documentation for documentation,"September 15, 2006. The recommended actions for the IR included, "Additional coaching /training to include in prompt operability call section, the actual values of Tech Spec parameters used in making the call, ...or more detail as to the reasoning used supporting the call being made.... Additional action to be taken as a result of Root Cause being performed on the 125 Vdc system corrosion."

Analysis: The team determined that inadequate justification in making prompt operability calls on safety-related SSCs was a performance deficiency. Specifically, although procedure guidance for making prompt operability calls existed, some of the reasons given for operability demonstrated a lack of understanding of the design basis of the 125 Vdc system. The finding was more than minor because it could reasonably be viewed as a precursor to a significant event and if left uncorrected, the finding could become a more significant safety concern. Specifically, failing to maintain adequate rigor in ascertaining and verifying the basis for operability calls could lead to an incorrect conclusion which could result in an SSC not fulfilling its design basis in an event. The issue was reviewed using the Phase 1 SDP worksheet for mitigating systems and determined the finding was of very low safety significance (Green), because subsequent analysis did not reveal any instance of an actual incorrect prompt operability call occurring.

Enforcement: Because this finding did not involve a violation of regulatory requirements and has very low safety significance, it is identified as FIN 05000254/265/2006003-12(DRS)

.4 **Operating Experience**

a. Inspection Scope

The team reviewed six operating experience issues (6 samples) to ensure these issues, either NRC generic concerns or identified at other facilities, had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection effort:

- IN-2002-12 and IN-2002-13; Submerged Electrical Safety-Related Cables;
- IN 97-90; Use of Non-Conservative Acceptance Criteria in Safety-Related Pump Surveillance Tests;
- 00519119; Quad Cities Review of Dresden TIA 2005-009 (EDG testing);
- IN-2002-01; Swgr Failure;
- GE SIL 615; ADS /HPCI Functional Redundancy; and
- OE 19296; CR 105 Contactor Slow to Operate Due to Binding in Auxiliary Contacts.

b. Findings

No findings of significance were identified.

.5 **Modifications**

a. Inspection Scope

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

- EC 343933, EC 334104; Replace PORV with ERV G2R17;
- DCP-MOD 01028115; Fabricate sleeve for Upper Thrust Bearing collar for 2A RHR pump;
- EC0000018160; 1/2 Diesel Instrumentation 7300002/M04-0-73-002;
- MOD 24042; Unit 1/2 EDG auto start TD;
- EC No. 24165; Revision 1; Condensate Pumps Trip Logic; and
- EC 340735; Modify high level trip logic to 1 out of 2 taken twice.

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 six risk significant, time critical operator actions (6 samples). These actions were selected from the licensee's PRA rankings of human action importance based on risk achievement worth and Birnbaum values. Where possible, margins were determined by the review of the assumed design basis and USAR response times and performance times documented by job performance measures results. For the selected operator actions, the team performed a walk through of associated procedures 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:

- Responses to station blackout (SBO);
- Failure to align a fire pump to SSMP [standby safety makeup pump] during a loss of service water;
- Failure to initiate automatic depressurization system (ADS) during a transient;

- Failure to initiate ADS and suppression pool cooling bypass for high torus water level switchover;
- Failure to manually initiate Torus cooling; and
- Failure to initiate ADS and SSMP.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

a. Inspection Scope

The team reviewed a sample of the selected component problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exits

.1 Exit Meeting Summary

The team presented the inspection results to Mr. Gideon and other members of licensee management at the conclusion of the inspection on September 15, 2006. A second telephone exit was conducted on November 3, 2006, to inform the licensee of changes to the findings discussed during the exit meeting on September 15, 2006. Proprietary information was reviewed during the inspection and was be handled in accordance with NRC policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee</u>
T. Tulon, Station Vice President
R. Gideon, Plant Manager
W. Beck, Regulatory Assurance Manager
S. Boline, Design Manager
D. Craddick, Maintenance Director
S. Darin, Plant Engineering Manager
F. Lantine, Corporate Engineer
T. Fuhs, Regulatory Assurance
D. Moore, Nuclear Oversight Manager
K. Moser, Engineering Director
R. Sualeson, Operations Director
J. Bailey, Plant Engineer
R. Buttke, Electrical Engineer
S. Laughlin, System Engineer
J. Taft, Design Engineering
J. Fredrichsen, Senior Staff Engineer
D. Boyles, Operations Support Manager

<u>Nuclear Regulatory Commission</u>
A. Boland, Deputy Director, DRS
K. Stoedter, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000254/265/2006003-01	NCV	Failure to Comply with TS SR 3.8.4.2 for 125 Vdc Battery Terminal Connection Corrosion and Resistance Measurements (Section 1R21.3.b.1)
05000254/265/2006003-02	NCV	Battery Connection Resistance Value Specified in TS SRs Insufficient to Ensure Operability (Section 1R21.3.b.2)
05000254/265/2006003-03	NCV	Calculation Input Design Data Discrepancies for the Auxiliary Power Analysis and EDG Loading (Section 1R21.3.b.3)
05000254/265/2006003-04	NCV	Licensee Used Inappropriate Vortex Analysis Methodology (Section 1R21.3.b.4)
05000254/265/2006003-05	NCV	Non-Conservative Sizing Calculation for ADS/SRV Air Accumulator Storage Tank (Section 1R21.3.b.5)
05000254/265/2006003-06	NCV	Discrepant MCC Voltages Used in Degraded MOV Voltage Drop Calculations (Section 1R21.3.b.6)
05000254/265/2006003-07	NCV	Inadequate Load Tabulation in Operations Procedure QCOP 6500-28 (Section 1R21.3.b.7)
05000254/265/2006003-08	NCV	Inconsistency in Procedures for Cleaning Batteries (Section 1R21.3.b.8)
05000254/265/2006003-09	NCV	Failure to Comply with Preventive Maintenance Procedure Requirements Concerning Re-Torquing of Corroded Electrical Terminal Connections (Section 1R21.3.b.9)
05000254/265/2006003-10	NCV	Non-Conservative HPCI Pump Test Acceptance Criteria (Section 1R21.3.b.10)
05000254/265/2006003-11	NCV	Non-conservative Safety-Related Air Storage Tank Capacity Test (Section 1R21.3.b.11)
05000254/265/2006003-12	FIN	Shift Management Failed to Adequately Document Basis for Operability Determination (Section 1R21.3.b.12)

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

1R21 Component Design Bases Inspection

Calculation

Number	**Title**	**Revision**
QDC-1000-S-1235	Seismic Evaluation of RHR Hx. Expansion Joint	2
QDC-1000-1019	QC EPU Eval of RHR/CS NPSH Analysis: Post-LOCA Long and Short Term	1
BSA-Q-00-03	Analysis for 2 and 4 RHR pump rates used in LOCA analysis	0
QC-NPD-96-0002	MO-1001-5AandB Post-Mod Vibration Test Evaluation	00
QDC-1000-0587	RHR Pump Required Discharge Pressure	0
QDC-4600-1112	Design Review of EDG Starting Air System Capability	0
DG-1	Diesel Generator Room Ventilation Load Verification	1A
0591-171-008	Diesel Fuel Oil Consumption and Tank Volume	1
QDC-6700-E-1503	Auxiliary Power Analysis	1
9390-02-19-3	Calculation For Diesel Generator1/2 EDG Loading Under Design Basis Accident Condition	3
9646-04-19-1	Calculation For ELMS-AC Load Data for the Emergency Diesel Generators	0
QC-019-E002	4kV Bus 13-1/23-1 and 14-1/24-1 Cross Tie - Coordination Study	3
QDC-6600-E-0949	Operating Setpoint, Tolerance, and Characteristics Of The TD1 Time Delay Relay, TD5 Time Delay Relay, TD7 Repeat Cycle Timer, and the 4641-45 Pressure Switch	2
7318-32-19-1	Calculation for Inputting 125 Vdc Load Profiles into ELMS-DC for Units 1 and 2	38J
557031-19-1	125 Vdc Fault Currents	2

Calculation Number	Title	Revision
QDC-0000-E-0206	Motor Terminal Voltage Calc for Quad Cities Unit 1 and Unit 2 GL 89-10 Motor Operated Valves	1
8913-69-19-4	Calc for Justification of the Adequacy of MCC Contactor	0
8913-73-19-6	Calc for Nonsize 2 Motor Control Center (MCC) Control Voltage Contactor Circuit Lengths fed from Switchgear 29	2
QDC-6700-E-1498	Second Level Undervoltage Relay Setpoint	1
QDC-8300-E-0482	Evaluation of 125 Volt DC System Coordination for Appendix R	4
5570-31-19-2	125 Vdc System Circuit Breaker and Fuse Coordination	0
QDC-3300-0489	Usable Water Volume of Contaminated Condensate Storage Tanks for HPCI and RCIC, Including Vortexing Considerations	2
QDC-2300-I-0964	HPCI/RCIC Level Switch Setpoint Error Analysis	0
QDC-2300-0486	Verification of HPCI Pump Discharge Flow to the Reactor	0
NUC-60	Accumulator Air Leakage for SO-203-3A	0
ATD-0189	DG Cooling Hydraulics	0
QDC-1000-0131	NPSH Availability versus Requirements for DGCW and RHRSW Pumps	2B
QDC-1000-0252	Actuator Torque MOV 2-1001-5A	0
QDC-1600-0545	ECCS Strainer Head Loss	3
QDC-1000-1019	EPU Evaluation NPSH RHR and CS Overpressure	1
QDC-1600-0545	ECCS Strainer Head Loss	3
QDC-1600-1153	ECCS Strainer	0
QDC-2300-0189	HPCI NPSH	2
QUA-1-2301-35	MIDACALC Results DC Motor Operated Gate Valve	3
XCE064.0200.001	EPU Hardened Vent	1A
QDC-0287-0701	ADS Accumulator Accident Pressure Rating	0
EMD-025814	SandL Calc, Stress Analysis, Unit 1 Target Rock Air Line Mod	0
64.305.2029	Torus Pitting Corrosion Acceptance Criteria for Quad Cities	1

Calculation Number	Title	Revision
2175C	Torque Requirements for Containment Purge	0
VR-10	Emergency Cooler Performance at Varied Service Water Flow Rates for Core Spray, HPCI, and RHR Systems	1

Condition Reports Generated In Response to Team Findings

IR No.	Title	Date
519883	NRC Identified Damaged Screen on SW Pump Motor	8/14/06
519885	Grounding Strap Improperly Fastened	8/14/06
520001	HPCI Fire System Support Missing a Fastener	8/15/06
520064	NRC Identified Dirt inside 1C RHRSW Pump Motor	8/14/06
520627	Performance of Maintenance Was Untimely	8/11/06
520654	NRC Questions May Lead to Potential Revision to RHR Pump Calculation	8/15/06
520716	4E-2306 Reference 4E-2351C Not In EDMS or AP Card	8/16/06
520851	CDBI Team Identified Incorrect Step Referencing In Procedure	8/17/06
521012	CDBI Identified: Missing Step in QCOP 6500-28	8/17/06
521056	NRC Identified Procedure Enhancements	8/17/06
521062	CDBI Identified Valve Locking Method not Adequate	8/31/06
521153	QCTS Does Not Meet Requirements of MA-AA-721-1001	8/14/06
521248	Input Data for 1/2 EDG is Incorrect Based on Nameplate	8/18/06
521252	Battery Corrosion is Not Acceptable	8/18/06
521503	Incorrect Input Parameters for Calc	8/18/06
523803	Inadequate Testing of Target Rock SRV Accumulator Check Valve	8/16/06
523884	Calculation ATD-0189 and Atd-0057 Assume Outdated Conditions	8/18/06
524658	Micro Ohm Reading Not Meeting Criteria	8/28/06
524923	Need to Update QDC-3300-0489 to Use Newer Methodology	8/28/06
525113	NRC Identified Procedural Error in QCEPM 0100-01	8/29/06

Condition Reports Generated In Response to Team Findings

IR No.	Title	Date
525397	Calculation NUC-60 is Outdated and Needs Revision	8/16/06
525492	Battery Surveillance Guidance Differs from Vendor	8/30/06
525592	HPCI Pumps Have Non-Conservative IST Criteria	8/23/06
526124	Revision Required to GE SIL 615 Response	8/29/06
526361	Voltage Drop Calculation Discrepancies	9/01/06
526373	ETAP Input Discrepancies for EDG	9/01/06
526481	Calculation Discrepancy, Superseded Calculation	8/31/06
530021	Incorrect Calculation Reference in Procedure MA-QC-773-524	9/12/06
530043	Undervoltage Relay Calibration Sheets Have Misleading Info	9/12/06
530329	Design Analysis Clarification	9/13/06
530416	Design Analysis Documentation Correction Required	9/13/06
530544	Failure to Meet SR 3.8.4.2 Concerning Battery Corrosion	9/13/06
530663	NRC Identified WO 905552 Traveler Conflicts with QCEPM 0100-01	9/13/06
531359	NRC Identified Insufficient Documentation for Operability	9/15/06
531361	HELB Evaluation Required for HPCI Steam Line Inside Drywell	9/15/06
534101	Basis for Battery Inter-Cell Connection Resistance in Tech Spec	9/21/06
534947	Corrosion Found on Unit 1 125 Vdc Batteries	9/22/06
535147	Corrosion on Unit 2 125 Volt Battery Cell Post	9/23/06
537864	Deficiency Tag Audit Results	9/13/06
540524	Basis for Battery Inter-Cell Resistance in Tech Specs	10/5/06
543848	Non Conservative TS SR for Battery Intercell Resistance	10/13/06

Condition/Issue Reports Reviewed During the Inspection

Number	Title	Date
00425566	LCO Critique: 2A RHR LCO Lesson Learned	11/18/05
00430814	Station LCO Execution	12/06/05
00378587	RHR HX Shell Inlet Piping Flanges 150 PSI Vs 300 PSI	9/27/05

Condition/Issue Reports Reviewed During the Inspection

Number	Title	Date
00378829	Procedure Needs Additional Notes For Clear CO Directions	9/28/05
00379390	LCO Critique: OPS Lessons Learned From 2A RHR LCO	9/29/05
460615-02	Functional Failure Cause Determination Evaluation.	5/1/06
324585	1D RHRSW Pump Tripped After Running For 4 Seconds	4/14/2005
438650	1B CS Pump Tripped Immediately When Starting	1/4/2006
493816	1C RHRSW Pmp Tripped On Startup	5/25/2006
482166	RHRSW Vault Sump Discharge Valve Failed to Seat	5/22/06
452716	Failed RHRSW Vault Check Valve	9/17/2006

Drawings

Number	Title	Revision
4E-2438Q	Schematic Diagram RHR Sys Pumps 1002A, B, C, and D, 4160V Bkr Cont Div 1 and 2, Sh 15	P
4E-2346	Schematic Diagram 4160V Bus 24-1 Standby Diesel 2 Feed and 24-1 Tie Breaker	AM
4E-1346	Schematic Diagram 4160V Bus 14-1 Standby Diesel 1 Feed and 24-1 Tie Breaker	AS
4E-1351C	Schematic Diagram Diesel Generator 1/2 Auxiliaries and Start Relays	T
4E-2306	Key Diagram, Reactor Building, 480V Swgr 28	W
4E-1304	Key Diagram, 4160V Switchgears 13-1 and 14-1	AD
4E-1304A	Key Diagram, 4160V Switchgears 13-1 and 14-1	C
4E-2304	Key Diagram, 4160V Switchgears 23-1 and 24-1	W
4E-1318B	Overall Key Diagram 125V DC Distribution Centers	J
4E-1318A	Key Diagram Turbine Building 125V DC Main Bus Distribution Panel	T
4E-6870K	Key Diagram Station Blackout 125V DC Switchboard 6A	C
4E-7870K	Key Diagram Station Blackout 125V DC Switchboard 7A	C
728E953	Process Diagram, High Pressure Coolant Injection System	4
M-34	Diagram of Pressure Suppression Piping, Sheet 1	BB
M-4A	Environmental Zone Map, Main Floor Plan. El. 647-6, Figure 4	E

Drawings

Number	Title	Revision
M-13	Diagram of Main Steam Piping, Sheet 1	AR
M-24	Instrument Air Piping, Reactor Building, Sheet 13	H
M-60	Main Steam Piping, Sheet 1	AR
M-71	Instrument Air Piping, Reactor Building, Sheet 8	B
M-69	Service Water Piping, Diesel Generator Cooling Water, Sheets 3 and 5	N
M-305	Target Rock Valve Instrument Air Line	B
M-22	Service Water Piping, Diesel Generator Cooling Water	X
M-87	HPCI Piping, Sheet 1	BH
M-87	HPCI Piping, Sheet 2	H
M-87	HPCI Piping, Sheet 3	G

Engineering Changes/Modifications

Number	Title	Date
[QC] DCP-MOD 19815	Fabricate Sleeve for Upper Thrust Bearing Collar for 2A RHR Pump	5/30/85
EC 24042	DCP 9900277, Revise the 1/2 EDG Auto- Start Circuit	9/6/00
EC 23512	Install New Breakers at Bus 14-1	10/17/01
EC 340735	Modify High Level Feedwater and Turbine Trip Logic	4/24/06
EC 24165	Trip Condensate/Booster Pump “1D” on LOCA With All Four Condensate/Booster Pumps Running”	1/03/01
EC 334115	Evaluate the Effect of the Installation of a Dehumidifier in the RHRSW Vaults	1/11/02

Miscellaneous Documents

Number	Title	Revision/Date
NRC IN 2002-01	NRC Information Notice 2002-01: Metalclad Switchgear Failures and Consequent Losses Of Offsite Power	1/8/02
NRC IN 97-90	Use of Nonconservative Acceptance Criteria in Safety-Related Pump Surveillance Tests.	12/30/97

Miscellaneous Documents

Number	Title	Revision/Date
Self Assessment Report	Quad Cities Nuclear Power Station Pre-Inspection Based on the NRC TI 2515/158, Review of Low Margin/High Risk Significant Components and Human Actions	12/12/05
DMA 35N	Portable Density/Specific Gravity/Concentration Meter Instruction Manual	12/09/05
VETIP C0004	Vendor Manual for GNB Batteries	-----
Spec. R-2380	Spec. For Diesel Engine-Generator Set	9/21/67
GE-NE-A22-00103-19-02	Dresden and Quad Cities Extended Power Uprate Task 0300, Nuclear Boiler Systems	3
GE-NE-A22-00103-08-01	Dresden and Quad Cities Extended Power Uprate Task 0400, Containment Systems	1
GE-NE-A22-00103-33-01	Dresden and Quad Cities Extended Power Uprate Task 0404, High Pressure Coolant Injection System	0
GE-NE-A22-00103-75-02	Dresden and Quad Cities Extended Power Uprate Task 0903(Q), Station Blackout (Quad Cities)	0
GE-NE-A22-00103-76-01	Dresden and Quad Cities Extended Power Uprate Task 0608, Ultimate Heat Sink	0
N/A	Letter from K. K. Niyogi to Charles Alguire, "Independent Review of Calc. No. QDC 3300-0489, Usable Water Volume of CCST for HPCI and RCIC, Including Vortex"	9/8/06
NUREG/CR-2772	Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors	6/1982
N/A	Final Report-Torus Immersion Area, ECCS Suction Strainer Inspection	11/6/00
N/A	Letter from Quad Cities to NRC: Quad Cities Station Units 1 and 2 Response to Generic Letter 89-16	10/30/89
N/A	Letter from T. M. Ross, NRC to T. J. Kovach, Quad Cities; Mark I Containment Hardened Wetwell Vent Generic Letter 89-16	2/1/90
N/A	Station Response to IE Bulletin 88-04, Potential Safety-Related Pump Loss	7/ 11/88; 2/27/89 1/8/90

Operability Determinations/Engineering Condition Evaluations

Number	Title	Revision/Date
523803	1-0203-3AD Target Rock SRV Accumulator Check Valve	0
EC 362983	Evaluate the Maximum Allowable Inter-Cell Resistance Acceptance Criteria for the Safety-Related 125/250 VDC Batteries	000

Operating Experience Reports

Number	Title	Date
AT 41939-02	NRC IN number 2000-20, “Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High Energy Line Break Barriers”	3/8/01
NTS Item 254-455-98-61501	GE SIL number 615, “HPCI/ADS Functional Redundancy	0

Procedures

Number	Title	Revision
QCOP 6600-17	Unit 2 Diesel Generator Simultaneous Supply to Busses 24-1 and 14-1	5
QCOP 6600-16	Unit 1 Diesel Generator Simultanious Supply to Busses 14-1 and 24-1	5
QCOP 6500-28	4KV/480V Bus Loading Profiles	0
CC-AA-309	Control of Design Analyses	5
CC-AA-309-1001	Guidelines for Preparation And Processing Design Analyses	2
QCOS 6900-01	Station Battery Weekly Surveillance	20
QCOS 6900-02	Station Safety-Related Battery Quarterly Surveillance	23
QCOS 6900-14	Station Battery Allowable Value Verification Surveillance	10
QCEPM 0100-01	Station Battery Systems Preventive Maintenance	22
QCTS 0210-03	Station Battery Individual Cell Discharge/Charge	4
QCTS 0210-02	Battery Charger Testing for Safety-Related 125 VDC and 250 VDC Batteries	11

Procedures

Number	Title	Revision
MA-QC-773-303	Quad Cities Nuclear Operational Analysis 1/2 Emergency Diesel Generator Relay Routine	1
CC-AA-206	Fuse Control	5
WC-AA-106	Work Screening and Processing	5
QCOP 1600-13	Post-Accident Venting of the Primary Containment	19
QP.10.09A	Underwater Construction Corp.	0
MA-QC-021-722	SDIV/CCST/Torus Level Switch Calibration	10
WC-AA-106	Work Screening and Processing	5

Surveillances (completed)

Number	Title	Dates performed/Rev.
QCOS 1000-06	2A RHR Pump Flow Rate Group a Test (IST and Operability)	12/20/05
QCOS 1000-06	2A RHR Pump Flow Rate Group a Test (IST and Operability)	3/21/06
QCOS 1000-06	2A RHR Pump Flow Rate Comprehensive Test (IST and Operability)	9/23/05
Special Test 1-167	EDG Fuel Consumption Test	0
QCOS 6600-15	Functional Test for EDG Vent N2 Backup System	7/11/06
QCOS 6600-03	EDG Fuel Oil Transfer Pump Monthly Operability	8/7/06
QCOS 6900-01	Station Battery Weekly Surveillance	20
QCOS 6900-02	Station Safety-Related Battery Quarterly Surveillance	23
QCEPM 0100-01	Station Battery Systems Preventive Maintenance	22
QCOS 2300-27	HPCI Pump Performance Test	9/23/01
QCOS 2300-06 (Unit 1)	IST Valve Test Acceptance Criteria Sheet	10/19/04
QCOS 2300-06 (Unit 2)	IST Valve Test Acceptance Criteria Sheet	6/21/04
QCOS 0010-11	D RHRSW Vault Sump Pump Check Valve High Level Alarm Test	8/25/04

Surveillances (completed)

Number	Title	Dates performed/Rev.
QCOS 0010-11	D RHRSW Vault Sump Pump Check Valve High Level Alarm Test	1/2/06
QCOS 6600-06	DG Cooling Water Pump Group B Flow Test	4/12/06 and 7/8/06
QCOS 6600-08	Unit 1/2 DGCW to Unit 1 and Unit 2 ECCS Room Cooler Check Valve	1/22/06
TIC-459	RHRSW and DGCW Pump Suction Piping DP Monitoring	Not listed
ER-AA-340-1002	Attachment 1, 1/2 DG Heat Exchanger Inspection Data Sheet	1/10/06
QCOS 1600-14	Pressure Suppression Valve Timing Test	8/15/06 5/28/06 2/16/06
QCTP 0820-10	Heat Exchanger Inspection Report, ½ EDG	5/8/01
QCOS 2300-07	HPCI System Turbine Overspeed Test	4/24/04

Work Orders

Number	Title	Date/Rev.
00379825-01	Post-Maintenance Test for 2-1001-5A.	3/27/03
00495063	Post-Maintenance Test for 2-1001-5B.	4/25/03
00770560	SW Strainer Backwash Qualitative Verification Performances	4/28/05
00481692-01	Bus 24 to 24-1 Feed Relay Routine	9/9/2003
00505226-01	Bus 14-1 to 24-1 Xtie Relay Routine	9/25/2003
98028544-01	Pre-outage Insp. and Test 4KV BRKR No.303	5/9/2002
99268727-01	2A RHR PP Relay Routine	5/24/2002
00479078-01	EM Perform 4KV Horizontal Breaker Inspection (Merlin Gerin), 4KV Breaker 232	7/12/2005
00797374-01	EM Perform 4KV Horizontal Breaker Inspection (Merlin Gerin), 4KV Breaker 243	12/10/2005
00877434	OP QCOS 6900-02 125 VDC Station Batteries	3/23/06
00905465	OP QCOS 6900-02 125 VDC Station Batteries	6/23/06

Work Orders

Number	Title	Date/Rev.
00608082	EM Tech Spec U-1 125 VDC Battery No. 1 Inspection	4/06/05
00324338	EM Tech Spec U-1 Battery and Rack Inspection	1/04/03
00850916	OA1/2 Diesel Generator Routine	11/02/05
00598882-01	Inboard MSIV and Target Rock Valve Pneumatic System Leak Test	4/13/05
00738139-01	Inboard MSIV and Target Rock Valve Pneumatic System Leak Test	4/12/06
00633034	Shared Unit CCST Low Level Switch Calibration (A)	6/9/05
00633033	Shared Unit CCST Low Level Switch Calibration (B)	6/9/05
00633032	Shared Unit CCST Low Level Switch Calibration (C)	6/9/05
00633031	Shared Unit CCST Low Level Switch Calibration (D)	6/9/05
00748123 01	Unit 2 Normal 125 Vdc Battery Inspection	5/26/06
00883795 01	FNE Corrosion Found on Unit 1 125 Vdc Battery	7/20/06
00930774 01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	6/20/06
00902418-01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	3/21/06
00891313 01	OP QCOS 6900-02 125 Vdc Station SR Battery Quarterly Surveillance	5/12/06
00905552 01	EM Unit 2 125 Vdc Battery Cells Have Corrosion	5/24/06 and 8/29/06
00881144 01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	1/10/06
00877434 01	OP QCOS 6900-02 125 Vdc Station SR Battery Quarterly Surveillance	3/23/06
00905465 01	OP QCOS 6900-02 125 Vdc Station SR Battery Quarterly Surveillance	6/23/06
00934403 01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	7/4/06
00608082 01	EM Tech Spec Unit 1 125 Vdc Battery No. 1 inspection	4/6/06
0084700 01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	1/14/06

Work Orders

Number	Title	Date/Rev.
00882826 01	OP QCOS 6900-01 125 Vdc Station SR Battery Weekly Surveillance	1/17/06
00867452 01	OP QCOS 6900-02 125 Vdc Station SR Battery Quarterly Surveillance	2/8/06
00922525 01	OP QCOS 6900-02 125 Vdc Station SR Battery Quarterly Surveillance	8/9/06

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
AR	Action Request
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CA	Corrective Action
CCST	Contaminated Condensate Water Storage Tank
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CCST	Condensate Storage Tank
CT	Current Transformer
DC	Direct Current
DCR	Design Change Request
DG	Diesel Generator
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling Systems
ERV	Electromatic Relief Valve
ETAP	Electrical Transient Analyzer Program
GL	Generic Letter
gpm	gallons per minute
HPCI	High Pressure Coolant Injection
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IR	Issue Report
IST	Inservice Testing
ISTS	Improved Standard Technical Specifications
JPM	Job Performance Measure
kV	Kilovolt
LAR	Licensee Amendment Request
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
psig	pounds per square inch gauge
psid	pounds per square inch differential
RCIC	Reactor Core Isolation Cooling
RCA	Root Cause Analysis
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SDP	Significance Determination Process

SE	Safety Evaluation
SPAR	Standardized Plant Analysis Risk
SRV	Safety Relief Valve
SSC	System, Structure, or Component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WO	Work Order