October 31, 2007

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

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000254/2007004;
05000265/2007004

Dear Mr. Crane:

On September 30, 2007, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Quad Cities Nuclear Power Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on October 2, 2007, with Mr. Tulon and other members of your staff.

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

Based on the results of this inspection, the inspectors identified six findings of very low safety significance (Green). All of these issues involve violations of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations in accordance with Section V1.A.1 of the NRC's Enforcement Policy.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000254/2007004; 05000265/2007004
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
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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
Chief Radiological Emergency Preparedness Section,
Dept. Of Homeland Security
D. Tubbs, Manager of Nuclear
MidAmerican Energy Company

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Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
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Chairman, Illinois Commerce Commission
Chief Radiological Emergency Preparedness Section, Dept. Of Homeland Security
D. Tubbs, Manager of Nuclear
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Letter to C. Crane from M. Ring dated October 31, 2007

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000254/2007004;
05000265/2007004

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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:	05000254/2007004 and 05000265/2007004
Licensee:	Exelon Nuclear
Facility:	Quad Cities Nuclear Power Station, Units 1 and 2
Location:	Cordova, Illinois
Dates:	July 1, 2007, through September 30, 2007
Inspectors:	K. Stoedter, Senior Resident Inspector M. Kurth, Resident Inspector G. Kolcum, Acting Resident Inspector R. Daley, Senior Reactor Inspector Z. Falevits, Senior Reactor Inspector A. Koonce, Reactor Engineer R. Langstaff, Senior Reactor Inspector D. Melendez-Colon, Reactor Engineer R. Orlikowski, Senior Resident Inspector - Duane Arnold W. Slawinski, Senior Health Physicist R. Winter, Reactor Engineer R. Ganser, Illinois Emergency Management Agency
Observers:	K. Streit, Nuclear Safety Professional Development Program Engineer
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000254/2007004, 05000265/2007004; 07/01/2007 - 09/30/2007; Quad Cities Nuclear Power Station, Units 1 & 2; Refueling and Outage Activities; Event Followup; and Other Activities.

The report covered a three-month period of inspection by resident and regional inspectors and announced inspections by radiation protection and maintenance effectiveness specialists. Six Green findings, all of which were Non-Cited Violations (NCVs), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding and a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, was identified on August 22, 2007. The finding occurred due to the failure to have instructions and procedures appropriate to the circumstance for performing valve operation test and evaluation systems (VOTES) testing on a high pressure coolant injection valve. This contributed to the unexpected isolation of the Unit 1 reactor water cleanup system due to keying a hand held radio during the VOTES test. Corrective actions for this issue included restoring the reactor water cleanup system, performing training on radio use, briefing personnel on the event, and updating other VOTES testing work instructions to ensure that the presence of a radio-free zone was clearly specified.

  This issue was more than minor because, if left uncorrected, the continued use of inadequate procedures would lead to additional initiating events and equipment isolations. This issue was of very low safety significance because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigating systems equipment would not be available. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Work Practices, Human Error Prevention because the licensee's human error prevention techniques were not used to ensure that the work activities were performed safely. (Section 4OA3.3)

- Green. An inspector-identified finding and a Non-Cited Violation of a Quad Cities Nuclear Power Station license condition for fire protection was identified on May 3, 2007, due to the failure to adequately control transient combustible materials in a transient combustible exclusion zone. Specifically, the inspectors discovered two large cardboard boxes and an aerosol spray can that contained

methyl alcohol improperly controlled and unattended in the cable spreading room. Corrective actions for this issue included removing the materials from the cable spreading room, providing additional oversight of the transient combustibles control program, and clearly labeling the cable spreading room as a transient combustible exclusion zone.

The inspectors determined that this issue was more than minor because it could be viewed as a precursor to a significant event, i.e., fire impacting multiple pieces of safety-related equipment. Specifically, multiple vertical cable risers were located within the zone of influence for the aerosol can. The inspectors determined that this issue was of very low safety significance based upon the criteria established in Inspection Manual Chapter 0609F, Table 2.9.1, "Risk Significance Based on ΔCore Damage Frequency." The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Work Practices, Oversight, in that the licensee did not ensure that supervisory and management oversight of work activities, including contractors, was appropriate such that nuclear safety was supported. (Section 4OA5.3)

Cornerstone: Mitigating Systems

- Green. A self-revealing finding and Non-Cited Violation of Technical Specification 5.4.1 was identified due to the failure to properly preplan and perform maintenance on safety-related equipment in May 2007. This failure resulted in unexpected leakage on two safety-related valves. Corrective actions for this issue included repairing the valves, revising the maintenance procedures to ensure they complied with procedural requirements, and providing additional training to maintenance and maintenance planning personnel on the planning and performing of maintenance activities.

  The inspectors determined that this issue was more than minor because, if left uncorrected, the failure to properly pre-plan and perform safety-related maintenance would lead to the continued degradation of equipment important to safety. This finding was of very low safety significance because the leakage did not result in the total loss of safety function for the main steam, high pressure coolant injection, or the containment isolation systems. The inspectors determined that this finding was cross-cutting in the area of Human Performance, Resources, Documentation because the licensee failed to have complete, accurate, and up to date procedures for performing safety-related maintenance. (Section 1R20)

- Green. A self-revealing finding and a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, was identified in September 2007 for the failure to identify and correct a condition adverse to quality. Specifically, the licensee failed to assure that the cause of the March 2007 failure of the 1D residual heat removal pump breaker was promptly identified and corrected. This resulted in an additional 1D residual heat removal pump breaker failure in May 2007. Corrective actions for this issue included performing an extent of condition

review and modifying all of the Unit 1 Merlin Gerin breakers and cubicles. At the conclusion of the inspection period, 17 of the 47 Unit 2 breaker cubicles had also been modified. The remainder will be modified during the next Unit 2 refueling outage.

This issue was more than minor because, if left uncorrected, the failure of safety-related breakers would continue to result in the inoperability of risk significant equipment. This finding was of very low safety significance because it was not a design deficiency, did not result in the total loss of a safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding was determined to be cross-cutting in the area of Problem Identification and Resolution, Corrective Action Program, Evaluation, because the licensee failed to thoroughly evaluate the March 2007 breaker failure to ensure that the resolution addressed the cause and extent of condition. (Section 4OA5.2)

Cornerstone: Barrier Integrity

- Green. A self-revealing finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, was identified on September 8, 2007, due to the failure to follow procedures during the performance of Unit 1 125 Vdc ground detection activities. The failure to follow procedures resulted in the inadvertent isolation of the Unit 2 reactor building ventilation system. Corrective actions for this issue included restoring the isolated plant equipment, briefing personnel on the event, revising the ground detection procedure to ensure consistency with other Exelon stations, requiring additional oversight of ground detection activities, and implementing additional human performance improvement initiatives.

  The inspectors determined that this issue was more than minor because if left uncorrected, it would lead to additional equipment issues. The inspectors determined that this issue was of very low safety significance because it did not represent a degradation of a radiological barrier provided by the standby gas treatment system, did not represent a degradation of the barrier function of the control room ventilation system against smoke or a toxic atmosphere, and did not represent an actual open pathway in the physical integrity of the reactor containment. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Work Practices, Human Error Prevention because the licensee's human error prevention techniques were not used to ensure that the work activity was performed safely. (Section 4OA3.4)

- Green. A self-revealing finding and Non-Cited Violation of Technical Specification 5.4.1 was identified on July 30, 2007, due to the failure to properly implement OP-AA-109-101, "Clearance and Tagging." This failure resulted in tripping the 1A fuel pool cooling pump during clearance and tagging activities. Corrective actions for this issue included restoring the fuel pool cooling system to normal operation, establishing a clearance order review board to thoroughly review clearance orders, holding training sessions to ensure that the clearance

order writers clearly understood that each clearance order step should contain only one equipment manipulation, and implementing additional actions to improve Operations Department performance.

This issue was more than minor because, if left uncorrected, the failure to properly implement the clearance and tagging program would become a more significant safety concern. The inspectors determined that this finding was of very low safety significance because the finding only represented a degradation of the radiological barrier provided by the spent fuel pool. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Resources, Documentation in that operations personnel did not ensure that Clearance Order 55101 was complete and accurate prior to use. (Section 4OA3.5)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power until September 8, 2007, when the unit was shut down to perform repairs on the inboard high pressure coolant injection system steam isolation valve. This valve failed to close during the performance of routine maintenance activities conducted on September 5. During the 3.5 day outage, the licensee repaired two other leaking valves, replaced multiple feedwater pump seals, and conducted switchyard maintenance. Unit 1 returned to power on September 10 and synchronized with the electrical grid on September 11. Unit 1 operated at normal power levels for the remainder of the inspection period with the exception of planned power reductions for routine activities.

Unit 2 operated at or near full power for the entire inspection period with the exception of planned power reductions for routine surveillances and control rod maneuvers.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed a partial walkdown of the following systems to verify the operability of redundant or diverse trains and components when safety-related equipment was inoperable. The inspectors compared actual plant configurations to procedural requirements to identify any discrepancies that could impact the function of the system and increase risk. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers.

- Unit 1 Torus Cooling and Residual Heat Removal Service Water during maintenance on the Unit 1 Hardened Vent and
- Unit 1 Diesel Generator Cooling Water during maintenance on the Unit 1 and Unit 2 Station Blackout Diesel Generators.

This inspection represented the completion of two quarterly samples.

.2 Complete Walkdown

a. Inspection Scope

The inspectors used system drawings, procedures, the Updated Final Safety Analysis Report, and Technical Specifications to perform an equipment alignment verification of the Unit 1 and Unit 2 scram discharge volume systems. These systems were selected because they were used to limit the loss of, and contain, the water from the control rod drives during a scram. The scram discharge volume vent and drain valves were also primary containment isolation valves. The failure of either redundant vent valve and/or drain valve represented a safety system degradation and could result in unnecessary challenges to the reactor protection and primary containment isolation systems. The inspectors reviewed the licensee's corrective action program records from July 2006 to July 2007 to verify that issues were being identified at the appropriate threshold and resolution of issues was appropriate. The inspectors performed a visual inspection of components and reviewed open work orders to determine if there were any outstanding issues that could impact performance of the system.

This review represented the completion of one semi-annual sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted a tour of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with the licensee's administrative procedures; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with the licensee's fire plan.

- 2A Residual Heat Removal Corner Room;
- 2B Residual Heat Removal Corner Room;
- Unit 1 Battery Room;
- Station Blackout Diesel Generator Building General Work Area;
- Station Blackout Diesel Generator Rooms;
- Station Blackout Diesel Generator Day Tank Rooms; and
- Station Blackout Diesel Generator Switchgear and Battery Areas.

This inspection represented the completion of seven quarterly samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the following items as part of the annual internal flooding inspection:

- Operating Experience Smart Sample FY 2007-02 - Flooding Vulnerabilities Due to Inadequate Design and Conduit/Hydrostatic Seal Barrier Concerns and
- Review of the Plant Floor Drain Maintenance Program.

As part of the smart sample review, the inspectors reviewed the licensee's internal flooding analysis, design documentation, inspection procedures and the Updated Final Safety Analysis Report to gain an understanding of the licensee's internal flooding program. Surveillance and test procedures were evaluated to ensure that the procedures appropriately tested all equipment credited by the licensee's internal flooding analysis. The inspectors also conducted a plant tour to identify areas where a piping failure could have a significant internal flooding impact. Lastly, the inspectors reviewed the licensee's evaluation of NRC Information Notice 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design," to determine whether the licensee had identified any previously unrecognized internal flooding vulnerabilities.

As part of the floor drain system inspection, the inspectors reviewed procedures to determine how the temporary plugging of floor drains was controlled. The inspectors also conducted a search of the corrective action program to determine how frequently workers identified that plant floor drains were plugged and the impact that the plugged floor drain could have in exacerbating an internal flooding event. The inspectors reviewed the maintenance work history for each plugged floor drain identified during the corrective action program review to determine whether each drain was unplugged in a timely manner.

This inspection represented the completion of two internal flooding samples.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On August 27 and 31, 2007, the inspectors observed operations crews in the simulator. The first scenario required operations personnel to respond to the loss of Transformer 12, a loss of feedwater heating, the loss of 4160 Volt Bus 13-1, and the initiation of steam cooling. For the second scenario, the operations crew performed simulator manipulations to address a loss of annunciator power, a reactor recirculation pump trip, the loss of Transformer 12, a loss of coolant accident inside containment, the loss of all high pressure injection systems, and the accumulation of hydrogen inside the drywell.

The inspectors evaluated crew performance in the areas of:

- clarity and formality of communications;
- ability to make timely actions in the safe direction;
- prioritization, interpretation, and verification of alarms;
- procedure use;
- control board manipulations;
- oversight and direction from supervisors; and
- group dynamics.

The inspectors verified that the crews completed the critical tasks listed in the above scenarios. If critical tasks were not met, the inspectors verified that crew and operator performance errors were detected and adequately addressed by the evaluators. The inspectors verified that the evaluators effectively identified crews requiring remediation and appropriately indicated when removal from shift activities was warranted. Lastly, the inspectors observed the licensee's critique to verify that weaknesses identified during this observation were noted by the evaluators and discussed with the respective crews.

These inspections represented the completion of two inspection samples.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors reviewed the licensee's handling of performance issues and the associated implementation of the Maintenance Rule to evaluate the maintenance effectiveness for the items listed below. These items were selected based on them

being designated as risk significant under the Maintenance Rule, being in increased monitoring, or due to an identified issue or problem that potentially impacted system work practices, reliability, or common cause failures.

- Offgas and Auxiliary Steam Systems (Maintenance Rule Functions Z5400 and Z3000-02);
- Feedwater System (Maintenance Rule Function Z3200);
- Electrohydraulic Control System (Maintenance Rule Function Z5650); and
- 4160 Volt Switchgear (Maintenance Rule Function Z6700).

The inspectors review included an examination of specific issues documented in issue reports, an evaluation of maintenance rule performance criteria and maintenance work practices, an assessment of common cause issues and extent of condition reviews, and trending of key parameters. The inspectors also reviewed the licensee's maintenance rule scoping goal setting, performance monitoring, functional failure determinations, and current equipment performance status.

This inspection represented the completion of four samples.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors examined the latest Maintenance Rule periodic evaluation report completed for the period of May 1, 2004 through May 1, 2006. The inspectors reviewed a sample of (a)(1) Action Plans, Performance Criteria, Functional Failures, and Condition Reports to evaluate the effectiveness of (a)(1) and (a)(2) activities. These same documents were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the inspectors reviewed the Maintenance Rule Procedures and processes. The inspectors focused the inspection on the following systems (samples):

- 480V MCCs;
- High Pressure Coolant Injection (HPCI);
- Unit 2 Fuel;
- Refueling Bridge Crane; and
- Control Room HVAC.

The inspectors verified that the periodic evaluations were completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 24 months). The inspectors also ensured that the licensee reviewed its goals, monitored Structures,

Systems, and Components (SSCs) performance, reviewed industry operating experience, and made appropriate adjustments to the Maintenance Rule Program as a result of the above activities.

The inspectors verified that:

- the licensee balanced reliability and unavailability during the previous cycle, including a review of high safety significant SSCs;
- (a)(1) goals were met, that corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and that (a)(1) activities and related goals were adjusted as needed; and
- the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, and reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

In addition, the inspectors reviewed Maintenance Rule Self-assessments and Audit Reports that addressed the Maintenance Rule Program implementation.

This review represented five triennial inspection samples.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the following activities to verify that the appropriate risk assessments were performed prior to removing equipment for maintenance. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors verified the appropriate use of the licensee's risk assessment tool and risk categories in accordance with procedures.

- Risk Management Documentation No. SA-1621, "Risk Assessment for Missed Technical Specification Surveillance Requirement for Quad Cities Main Steam Safety Valves," Revision 0;
- Work Week 27 (July 1 - 8) including planned maintenance on the Unit 2 emergency diesel generator, the 2C and 2D residual heat removal pumps, and the 2C and 2D residual heat removal service water pumps;
- Work Week 33 (August 12 - 18) including emergent maintenance on offsite power line 0404;
- Work Week 37 (September 9 - 15) including planned maintenance on the Unit 1 and Unit 2 Station Blackout Diesel Generators;

- Work Week 38 (September 17 - 23) including planned maintenance on the 1/2B fire diesel, the Unit 1 250 V battery charger, the 1B instrument air compressor; the 2A standby liquid control pump, and the 2A residual heat removal loop; and
- Work Week 39 (September 24-29) including planned maintenance on the 1B core spray system, the Unit 1 emergency diesel generator, the Unit 2 250 Vdc battery charger, and the Unit 2 instrument air compressor.

This inspection represented the completion of six samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

For the operability evaluations listed below, the inspectors evaluated the technical adequacy of the evaluations to ensure that Technical Specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the Updated Final Safety Analysis Report to verify that the system or component remained available to perform its intended function. In addition, the inspectors reviewed compensatory measures implemented to verify that the compensatory measures worked as stated and the measures were adequately controlled. The inspectors also reviewed a sampling of issue reports to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations.

- Issue Report 648646 - Unit 1 Maximum Combined Flow Limit Setting;
- Issue Report 639290 - 2-1201-85 Valve Reach Rod Misaligned, Valve Won't Operate;
- Issue Report 650893 - Wire in CAM Does Not Match Replacement Wire;
- Issue Report 649225 - 2A Residual Heat Removal Cooler Needs Cleaned; and
- Issue Report 653838 - Emergency Diesel Generator Heat Load Deficiency.

This inspection represented the completion of five samples.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the licensee's test procedure to verify that the procedure adequately tested the safety function(s) that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed the test data, to verify that test results adequately demonstrated restoration of the affected safety function(s).

- Work Order 1051763 - Troubleshoot failure of 2A reactor recirculation motor generator set to change speed during Unit 2 power reduction;
- Work Order 1034756 - Furmanite Repair (by peening) Packing Leak Off Plug;
- Work Order 863091 - Train B Standby Gas Treatment System Charcoal Replacement, Freon Leak Test;
- Work Order 1021847- 1D Residual Heat Removal Pump Breaker Troubleshooting;
- Work Order 1059022 - Troubleshoot failure of High Pressure Coolant Injection Valve 1-2301-4; and
- Work Order 1060659 - Repair of Main Steam Line Drain Line Isolation Valve 1-220-01.

This inspection represented the completion of six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

On September 7, 2007, the licensee shut down Quad Cities Unit 1 to perform repairs on high pressure coolant injection valve 1-2301-4. The inspectors attended meetings to assess the adequacy of the licensee's troubleshooting efforts, the thoroughness of the forced outage plan and schedule, and the risk associated with the outage related activities. During the 3.5 day outage, the inspectors conducted daily tours of the control room to assess the adequacy of decay heat removal methods, electrical distribution configuration, reactor vessel inventory controls, and compliance with Technical Specifications. The inspectors also observed the licensee's response to several emergent issues which were identified following the shutdown. During maintenance activities, the inspectors performed a drywell tour to assess the licensee's adherence to foreign material exclusion controls. The inspectors also observed various portions of

the shutdown and startup to ensure that operations personnel maintained procedural compliance and operated the plant in accordance with NRC requirements.

b. Findings

Documentation Issues Result in Safety Related Valve Leakage

Introduction: A self-revealing Green finding and Non-Cited Violation of Technical Specification 5.4.1 was identified due to the failure to properly pre-plan and perform maintenance on safety-related equipment. The failure to properly pre-plan and perform maintenance activities resulted in unexpected leakage on two safety-related valves.

Description: On September 8, 2007, operations personnel entered the Unit 1 drywell shortly after the reactor was shut down. During the entry, a self-revealing leak was discovered on Main Steam Line Drain Line Isolation Valve 1-220-01. The inspectors noted that valve 1-220-01 was also an inboard containment isolation valve. The inspectors discussed the valve leakage with licensee management and learned that the valve was replaced during the May 2007 Unit 1 refueling outage. Although this valve passed all of the associated post maintenance tests, historical drywell floor drain sump leakage results indicated that this valve had been leaking for some time. In addition, the leakage resulted in the creation of a ground on one of the inboard main steam isolation valve limit switches.

The licensee and the inspectors reviewed the maintenance work package and procedure used to replace valve 1-220-01 in May 2007. Both parties determined that the valve likely leaked because an incorrectly sized seal ring was installed in the valve in May 2007. The incorrectly sized seal ring was selected for installation because the guidance provided in Procedure QCMM 1515-17, “Pressure Seal Gate Valve Maintenance,” was not specific enough to ensure that correctly sized seal rings were installed for this valve type and size.

Not long after the leak on valve 1-220-01 was identified, another self-revealing leak was found on High Pressure Coolant Injection Discharge Check Valve 1-2301-7. This valve was also disassembled and inspected during the May 2007 Unit 1 refueling outage. The inspectors discussed the valve leakage and the previous maintenance activities with licensee personnel. Through these discussions, the inspectors concluded that the high pressure coolant injection valve leakage occurred because workers failed to use the correct torque value when reassembling the valve. The incorrect torque value was used because maintenance planning personnel directed the workers to select a torque value listed on a generic torque table rather than clearly providing the torque value as required by Procedure MA-MW-716-010-1000, “Passport Work Planning Manual.” The inspectors reviewed the generic torque table with licensee personnel and discovered that none of the possible torque values matched the vendor specified torque value.

Analysis: The inspectors determined that the failure to properly pre-plan and perform maintenance on safety-related equipment was more than minor because if left uncorrected this condition would result in the continued degradation of safety-related

equipment. The inspectors completed a Phase 1 Significance Determination Process screening and concluded that this finding was of very low safety significance (Green) because the leakage did not result in the total loss of safety function for the main steam, high pressure coolant injection, or the containment isolation systems. The inspectors determined that this finding was cross-cutting in the area of Human Performance, Resources, Documentation because the licensee failed to have complete, accurate, and up to date procedures for pre-planning and performing the safety-related maintenance activities (H.2(c)).

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented, and maintained for the items specified in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 9a, "Procedures for Performing Maintenance," requires that maintenance that can affect the performance of safety-related equipment be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstance. Contrary to the above, in May 2007, safety-related maintenance was pre-planned and performed on the main steam and the high pressure coolant injection systems using procedures which were not appropriate to the circumstance. Specifically, the procedures used failed to provide appropriate guidance for the selection of valve seal rings and torque values. Because this violation was determined to be of very low safety significance, and because you entered this issue into your corrective action program as Issue Reports 669178, 669191 and 669784, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the Enforcement Policy **(NCV 05000254/2007004-01)**. Corrective actions included repairing the valves, revising the maintenance procedures to ensure they complied with procedural requirements, and providing additional training to maintenance and maintenance planning personnel on the planning and performing of maintenance activities.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of selected risk-significant structures, systems, and components listed below, to assess whether the structures, systems, and components met the requirements of the Technical Specifications, the Updated Final Safety Analysis Report, and the American Society of Mechanical Engineers Section XI. The inspectors also determined if the testing effectively demonstrated that the structures, systems, and components were operationally ready and capable of performing their intended safety functions.

- Review of Methods Used to Verify Ultimate Heat Sink Operability;
- Testing of High Pressure Coolant Injection Valve 1-2301-10;
- Review of Recommendation to Discontinue On-line Surveillance Testing of High Pressure Coolant Injection and Reactor Core Isolation Cooling Primary Containment Isolation Valves; and
- QCOS 1400-01 - Quarterly Core Spray Flow Rate Test.

This inspection represented the completion of one routine surveillance and three inservice testing samples. See Section 4OA3.3 for additional details regarding the testing of the high pressure coolant injection valve.

b. Findings

On August 21, 2007, Dresden Station experienced a grass intrusion event which impacted the operation of several systems which take a suction from the Kankakee River. The following day, the inspectors reviewed the Quad Cities Technical Specifications and procedures to determine how operations personnel would respond to a similar intrusion event. The inspectors identified the following concerns:

- Technical Specification 3.7.3 required the ultimate heat sink to be operable in Modes 1, 2, and 3. To maintain operability, operations personnel recorded the water level at the intake bay and the ultimate heat sink temperature every 24 hours. The inspectors determined that the licensee's current Technical Specification Surveillance Requirement for water level may be non-conservative under a postulated intrusion event because the water level at the suction of the safety-related cooling water pumps could be significantly lower than the water level at the intake bay. This would result in the safety-related cooling water pumps and the ultimate heat sink being inoperable.

- The licensee's approved Emergency Action Levels were also based on water level at the intake bay. The inspectors determined that the approved Emergency Action Levels were adequate to address ultimate heat sink inoperability due to a lock and dam failure. However, they did not appear to address ultimate heat sink inoperability due to a biological or ice intrusion event.

- No procedural guidance existed regarding the measuring of water level at the intake of the safety-related pumps.

The licensee initiated Issue Report 663343 to document the inspectors concerns. Issue Report 665703 was also initiated to document specific concerns regarding the Emergency Action Levels. At the conclusion of the inspection period, the licensee was continuing to evaluate the inspectors concerns. The inspectors considered this item to be unresolved pending a review of the licensee's resolution to the above issues **(URI 05000254/2007004-02; 05000265/2007004-02)**.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the temporary modification listed below and the associated 10 CFR 50.59 screening, and compared each against the Updated Final Safety Analysis Report and Technical Specifications to verify that the modification did not affect the operability or availability of the affected system. The inspectors also compared the licensee's information to operating experience information to ensure that lessons

learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance.

- Operation of the 2B1 feedwater heater using the emergency level control valve instead of the normal level control valve.

This inspection represented the completion of one sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors evaluated the conduct of a routine emergency drill on August 28, 2007. The inspectors observed emergency response operations in the Technical Support Center to verify that event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to ensure that the inspector-observed weakness were also identified by the licensee's drill evaluators.

This inspection represented the completion of one annual drill evaluation sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System Description and Waste Generation

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system descriptions in the Updated Final Safety Analysis Report (UFSAR), and reviewed the 2005 and 2006 Annual Radioactive Effluent Release Reports for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit and self-assessment activities with regard to the

radioactive material processing and transportation programs to determine if those activities satisfied the requirements of 10 CFR 20.1101(c), and the quality assurance audit requirements of Appendix G to 10 CFR Part 20, and of 10 CFR 71.137.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors walked down portions of the liquid and solid radwaste processing systems to determine if these systems were consistent with the descriptions in the UFSAR and in the Process Control Program, and to assess the material condition of various areas of the radwaste building and radwaste equipment. No changes were made to the radwaste processing systems since the last inspection of this program area. The inspectors reviewed the status of radioactive waste process equipment that was not operational and/or abandoned in place, and the associated administrative and physical controls that were implemented to ensure that this equipment would not contribute to an unmonitored release path or be a source of unnecessary exposure.

The inspectors walked down the Interim Radwaste Storage Facility (IRSF) and satellite radiologically controlled areas where radioactive waste was stored to assess material conditions, inventory control, and to determine whether the facilities/equipment were consistent with descriptions in the UFSAR, as applicable, or otherwise that changes were reviewed by the licensee in accordance with 10 CFR 50.59.

The inspectors reviewed the licensee's processes for transferring and dewatering various waste stream resins into shipping containers to determine if appropriate mixing was performed so as to obtain representative waste stream samples for analysis. The inspectors reviewed the licensee's practices for the collection of area smear surveys to represent the dry-active waste (DAW) stream and the methods used for determining the radionuclide mix of filter media to determine if they were representative of the intended radwaste stream. Additionally, the inspectors reviewed the methodologies for quantifying gamma emitting radionuclide waste stream content, for determining waste stream tritium concentrations and for waste concentration averaging to determine if representative samples of the waste products were provided for the purposes of waste classification as required by 10 CFR 61.55.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's methods and procedures for determining the classification of radioactive waste shipments including the use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides and those that decay by electron capture). The inspectors reviewed the last two radiochemical sample analysis results (i.e., 10 CFR Part 61 analyses) including vendor laboratory data for each of the licensee's waste streams, and the associated calculations used to account for difficult-to-measure radionuclides. These waste streams consisted of condensate system and fuel pool resins, various filter media, DAW and irradiated hardware (activated metals). The licensee had not shipped reactor water cleanup system resins since the last inspection in this area. The inspectors also reviewed the minimum detectable concentrations achieved for each waste stream as determined by the licensee's contract analytical laboratory compared to the corresponding radionuclide groupings in 10 CFR 61.55 to determine whether the concentration values satisfied the NRC Branch Technical Position on Radioactive Waste Classification. These reviews were conducted to determine if the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to determine if reactor coolant chemistry data was periodically evaluated to account for changing operational parameters that could potentially affect waste stream classification and thus validate the continued use of existing scaling factors between sample analysis updates.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipment Manifests

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, radiation surveys, package labeling and marking, vehicle inspections and placarding, emergency instructions, and licensee verification of shipment readiness for six non-excepted radioactive material and radwaste shipments made between May 2005 and August 2007. For those shipments made in Type B casks, the inspectors selectively determined if the cask Certificate of Compliance was met for the shipment. The shipment documentation reviewed consisted of:

- Two Type B Packages of Activated Metals to a Low Level Waste Burial Site;
- Type B Package of Mechanical Filters to a Low Level Waste Burial Site;
- Low Specific Activity (LSA) Shipment of DAW to a Waste Processor;

- Surface Contamination Object (SCO) Shipment (Valves) to a Vendor; and
- Type A Package of Contaminated Equipment to a Low Level Waste Burial Site.

For each shipment, the inspectors determined if the requirements of 10 CFR Parts 20 and 61, and those of the Department of Transportation (DOT) in 49 CFR Parts 170-189 were met. Specifically, records were reviewed and staff involved in shipment activities were interviewed to determine if packages were labeled and marked properly, if package and transport vehicle surveys were performed with appropriate instrumentation, and whether radiation survey results satisfied DOT requirements, and if the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also determined whether shipment manifests were completed in accordance with DOT and NRC requirements, if they included the required emergency response information, if the recipient was authorized to receive the shipment, and if shipments were tracked as required by 10 CFR Part 20, Appendix G.

Selected staff involved in shipment activities were interviewed by the inspectors to determine if they had adequate skills to accomplish shipment related tasks, and to determine if the shippers were knowledgeable of the applicable regulations to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172 Subpart H. Also, the inspectors observed radiation protection technicians conduct surveys of an outbound shipment of spent resins in a Type A cask to assess the adequacy of the surveys, and examined the package marking and labeling, vehicle placarding, and driver instructions for compliance with DOT requirements. Additionally, the lesson plans for Safety Training, General Awareness/ Familiarization and Function Specific Training for radiation protection technicians and station laborers, and the training certificates for storeroom personnel were reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems for Radwaste Processing and Transportation

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (as applicable), selected condition reports (assignment reports (ARs)), self-assessment and audit reports that involved the radioactive waste and radioactive materials shipping program performed between 2005 and August 2007 to determine if the licensee had effectively implemented its corrective action program, and that problems were identified, characterized, prioritized, and corrected. The inspectors determined whether the licensee's oversight mechanisms (audits, self-assessments, etc.) collectively were capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also selectively reviewed ARs generated since the previous inspection that dealt with the radioactive material/radwaste shipping program or waste processing activities, and interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner, commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of Non-Cited Violations tracked in the corrective action program; and
- Implementation/consideration of risk significant operational experience feedback.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Public Radiation Safety

Radiation Safety Strategic Area

a. Inspection Scope

The Inspectors sampled the licensee's Performance Indicator (PI) submittals for the period indicated below. The inspectors used PI definitions and guidance contained in Revision 5 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following PI was reviewed:

- Radiological Environmental Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM) Radiological Effluent Occurrence.

The inspectors reviewed data associated with the RETS/ODCM PI to determine if the indicator was accurately assessed and reported. The inspectors reviewed the licensee's AR database and selected individual reports generated between mid-2006 through mid-2007, to identify any potential occurrences such as unmonitored, uncontrolled or improperly calculated effluent releases that may have impacted offsite dose. The inspectors also selectively reviewed gaseous and liquid effluent summary data and the results of associated offsite dose calculations for selected dates between October 2006 and July 2007 to determine if indicator results were accurately reported. The inspectors

also discussed with the licensee the methods for quantifying gaseous effluents and for determining effluent dose.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program:

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors screened all items entered into the licensee's corrective action program. This was accomplished by reviewing the description of each new issue report and attending daily management review committee meetings.

.2 Review of Licensee's Corrective Actions Initiated to Address CDBI Identified Issues

Introduction

During the Quad Cities Component Design Bases Inspection (CDBI) which ended on September 15, 2006, the team identified 11 Non-Cited Violations (NCVs) and one finding which were documented in IR 05000254/265/2006003 (DRS). In addition, the team identified other issues that were characterized as minor issues and were entered into the licensee's corrective action program for resolution, but were not documented in the report. The issues identified by the team related mainly to inadequate design calculations; untimely corrective actions to address long-term corrosion on 125 Vdc safety related battery terminals and failure to meet battery Technical Specification requirements; procedure deficiencies and failure to follow procedures; inadequate testing and related acceptance criteria; and lack of adequate documentation for prompt operability calls in issue reports.

During this inspection activity, the inspectors assessed the licensee's prioritization and evaluation of the identified issues. This fulfills an inspection objective (one sample) of NRC Inspection Procedure 71152.

Prioritization and Evaluation of Issues

Inspection Scope

As part of the corrective action plan, the licensee generated 48 Action Tracking items (ATs) to address the identified CDBI related issues. At the time of this inspection, required corrective actions identified in 40 of the 48 ATs had been completed. The remaining 8 have been assigned and scheduled for completion by end of 2007. The

inspectors performed focused reviews by sampling related issue reports, ATs, engineering improvement initiatives, self assessments and Root Cause Analysis findings initiated to determine and correct root and contributing causes of related CDBI findings. The inspectors also reviewed the revised portions of calculations, operating, maintenance, and surveillance procedures, Engineering Changes, and Training Lesson Plans which were changed to address the CDBI related findings. The inspectors verified if actions were timely and commensurate with the significance of the identified issues.

Issues

The inspectors noted that the licensee performed technical evaluations and extent of condition and cause reviews for some of the CDBI identified issues (e.g. battery corrosion, inadequate calculations and testing related concerns) to identify potential adverse trends in plant processes and to initiate needed improvements. For example, to address the testing acceptance criteria concerns identified by the team the licensee tasked all System Engineers with performing technical evaluation reviews of system surveillance acceptance criteria to ensure consistency with the plant's original design bases and calculations. In addition, the licensee has been re-evaluating the modified Technical Specification Bases acceptance criteria used for battery cell resistance measurement to ensure uniform implementation in all Exelon plants.

During the CDBI in 2006, the NRC identified a number of corrosion management issues relating to safety related 125 Vdc batteries which required implementation of extensive licensee corrective actions during and after the inspection. On July 17, 2007, the inspectors performed a sample field inspection and observed licensee operations and maintenance personal perform weekly battery surveillance activities on Unit 1 125 Vdc batteries. The licensee identified several battery inter-cell terminal connections that contained 2-Dimensional surface corrosion. No 3-Dimensional corrosion was identified by the licensee during this surveillance. Subsequently, the inspectors visually inspected the battery inter-cell connections and noted one 3-Dimensional corrosion on cell Number 31 (NEG) which was not identified by the licensee during this weekly surveillance inspection. The licensee promptly followed procedure requirements (when 3-Dimensional corrosion was identified) and performed resistance measurements on all battery cell connections and cleaned the terminals identified as having corrosion. The inspectors noted that approximately 200 issue reports had been issued in the last year documenting identification of corrosion on 24/28 V and 125 V Unit 1 and 2 batteries. This warrants vigilance and close management attention to ensure batteries are adequately maintained operational. The licensee was trending this condition and evaluating improvements in training and procedures. In addition, the licensee was planning to replace the aging (about 18 years) Units 1 and 2, 125 Vdc safety-related batteries in 2009 and 2010 respectively.

Based on review of corrective action documents, revised design documents and procedures, field observations and interviews with plant staff members, the inspectors concluded that the licensee had completed a majority of the corrective actions items to address the findings identified by the CDBI team. The corrective actions reviewed appeared to be appropriate and sufficient to address the identified causes.

Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report 05000254/06-001-01: Failure of the Unit 1B Core Spray Pump Breaker to Operate due to Racking Deficiency.

This licensee event report was a revision to a report submitted to the NRC on March 6, 2006. The review of the original report was documented in NRC Inspection Report 05000254/2006005; 05000265/2006005. The inspectors reviewed the information provided in the licensee event report revision and determined that the new information had not changed the NRC's assessment of the original issue. Therefore, no further NRC actions were needed.

.2 (Closed) Licensee Event Report 05000254/07-001: Two Main Steam Safety Valves and One Main Steam Safety/Relief Valve Outside of Technical Specification Allowed Tolerance Due to Setpoint Drift.

On November 7, 2006, the licensee submitted a License Amendment to the NRC to revise the values associated with Technical Specification Surveillance Requirement 3.4.3.1. Once approved, this amendment would revise Technical Specification Surveillance Requirement 3.4.3.1 such that the main steam safety valves would be required to actuate within plus or minus three percent of their setpoint rather than the plus or minus one percent currently required. The licensee requested that the NRC approve the License Amendment by November 2007.

On May 16, 2007 the licensee determined that the as-found setpoints for two main steam safety valves and the main steam safety/relief valve were outside of the plus or minus one percent currently specified in Technical Specification Surveillance Requirement 3.4.3.1. The licensee determined that the valves failed to meet the Technical Specifications due to setpoint drift.

The inspectors reviewed the licensee's corrective action documents for this event and determined that the setpoint drift would not have prevented the main steam safety valves from protecting the reactor vessel from an overpressure condition. In addition, the inspectors determined that the licensee would have met Technical Specification Surveillance Requirement 3.4.3.1 if the License Amendment would have been approved by the NRC prior to May 16, 2007. Based upon this information, the inspectors determined that this failure to comply with Technical Specification Surveillance

Requirement 3.4.3.1 was a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the out of tolerance condition in Issue Report 630204. Corrective actions included replacing the valves and the continued pursuit of the License Amendment.

.3 Failure to Have Adequate Procedure and Use Human Performance Tools Results in Reactor Water Cleanup Isolation

a. Inspection Scope

The inspectors interviewed licensee personnel and reviewed procedures and corrective action documents to determine the circumstances which resulted in the inadvertent isolation of the Unit 1 reactor water cleanup system.

b. Findings

Introduction: A self-revealing Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, was identified due to the failure to have instructions and procedures appropriate to the circumstance for performing valve operation test and evaluation systems (VOTES) testing on a high pressure coolant injection valve. The lack of an appropriate procedure contributed to the unexpected isolation of the Unit 1 reactor water cleanup system during the VOTES testing activity.

Description: On August 22, 2007, electrical maintenance personnel were preparing to perform VOTES testing on high pressure coolant injection motor operated valve 1-2301-10. To complete the testing, two electricians were stationed at the valve in the high pressure coolant injection room. Two additional electricians were located at the valve's breaker which was located on the second floor of the Unit 1 reactor building near the reactor water cleanup isolation instrumentation. The electricians planned to communicate with each other using hand held radios.

At 10:40 a.m., Unit 1 control room personnel received numerous alarms associated with an unexpected isolation of the reactor water cleanup system. Operations personnel responded to the second floor of the reactor building to determine the cause of the isolation. Upon arriving at the scene, the operators discovered the two electricians located at the breaker for valve 1-2301-10. In addition, the operators identified that the electricians at the breaker were working in a "radio free zone." At Quad Cities, radio-free zones were designated by the presence of orange and white markings on the floor. In addition, the words "radio free zone" were also stenciled on the floor. The operators determined that the reactor water cleanup system isolation had occurred because the electricians located at the valve breaker had used a radio to communicate with the electricians in the high pressure coolant injection room.

The inspectors discussed this event with one of the electricians located at the valve breaker. The inspectors also reviewed the licensee's prompt investigation report, procedures, and apparent cause report. Based upon the results of these activities, the inspectors determined the following:

- At least one of the electricians at the valve's breaker was aware of the orange and white markings on the reactor building floor;
- The electrician that noticed the markings did not question their purpose nor did he associate the markings with a radio free zone;
- The pre-job briefing, work package, and procedural information failed to include information that the breaker for valve 1-2301-10 was located in a radio-free zone;
- Operations Procedure QCOP 9000-04 contained information regarding the proper use of hand held radios and the specific location of the radio-free zones; there was not a similar maintenance procedure;
- Maintenance personnel have not received specific training on radio use; and
- The electricians failed to use human performance tools to identify the radio-free zone.

Analysis: The inspectors determined that the failure to have an appropriate procedure for the VOTES test was a performance deficiency that should be evaluated using the Significance Determination Process. The inspectors determined that this issue was more than minor because, if left uncorrected, the continued use of inadequate procedures would lead to additional initiating events due to unexpected equipment isolations. The inspectors determined that this issue was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Work Practices, Human Error Prevention because the licensee's human error prevention techniques were not used to ensure that the work activities were performed safely (H.4(a)).

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by documented instructions, procedures, and drawings appropriate to the circumstance. Contrary to the above, on August 22, 2007, instructions and procedures for performing VOTES testing were not appropriate to the circumstance. Specifically, the procedures and instructions failed to include information warning personnel that the VOTES test would be performed in a radio free zone. Because this violation was of very low safety significance, and because the issue was entered into the corrective action program as Issue Report 663190, the issue is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2007004-03)**. Corrective actions for this issue included restoring the isolated plant equipment, performing additional training on radio use, briefing personnel on the event, and updating other VOTES testing work instructions to ensure that the presence of a radio-free zone was clearly specified.

.4 Failure to Follow Procedure and Use Human Performance Tools Results in Reactor Building Ventilation Isolation

a. Inspection Scope

The inspectors interviewed licensee personnel and reviewed procedures and corrective action documents to determine the circumstances which resulted in the inadvertent isolation of the Unit 1 reactor building ventilation system during ground detection activities.

b. Findings

Introduction: A self-revealing Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, was identified due to the failure follow procedures during the performance of Unit 1 125 Vdc ground detection activities on September 8, 2007. The failure to follow procedures resulted in the inadvertent isolation of the Unit 2 reactor building ventilation system.

Description: On September 8, 2007, operations personnel performed 125 Vdc ground detection activities in Distribution Panel 2B-1 using the guidance contained in QOP 6900-06. As part of these activities, a non-licensed operator opened Circuit #18, "Division B Main Feed Auto Blow-Down Logic," as required by Step F.6.d.4 of QOP 6900-06. After opening the circuit, a second non-licensed operator monitored the Unit 1 125 Vdc ground recorder for any changes. After verifying that the ground recorder had not changed, the operator at the distribution panel was instructed to close Circuit #18 per Step F.4.e.1 of QOP 6900-06. Instead of closing Circuit #18, the operator opened Circuit #16 which resulted in the unexpected isolation of the Unit 2 reactor building ventilation system.

The inspectors interviewed operations personnel and reviewed the associated work package and procedure following this event. The inspectors determined that Circuit #16 was mistakenly opened because the non-licensed operator failed to use appropriate human performance tools to ensure that procedural compliance was maintained. The licensee also failed to provide additional oversight of the ground detection activities to minimize the potential for human performance errors.

Analysis: The inspectors determined that the failure to follow the procedures for the ground detection activity was a performance deficiency that should be evaluated using the Significance Determination Process. The inspectors determined that this issue was more than minor because, if left uncorrected, it would lead to additional unexpected equipment issues. The inspectors determined that this issue was of very low safety significance (Green) because it did not represent a degradation of a radiological barrier provided by the standby gas treatment system, did not represent a degradation of the barrier function of the control room ventilation system against smoke or a toxic atmosphere, and did not represent an actual open pathway in the physical integrity of the reactor containment. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Work Practices, Human Error Prevention because the

licensee's human error prevention techniques were not used to ensure that the work activities discussed above were performed safely (H.4(a)).

Enforcement: Title10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by documented instructions, procedures, and drawings appropriate to the circumstance and that the activities be accomplished in accordance with these instructions, procedures, and drawings. Contrary to the above, on September 8, 2007, operations personnel failed to accomplish Unit 1 125 Vdc ground detection in accordance with QOP 6900-06. Specifically, operations personnel failed to close Circuit #18 of Distribution Panel 2B-1 in accordance with Step F.4.e.1 of QOP 6900-06. Because this violation was of very low safety significance, and because the issue was entered into the corrective action program as Issue Report 669304, the issue is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2007004-04; 05000265/2007004-04)**. Corrective actions for this issue included restoring the isolated plant equipment, briefing personnel on the event, revising the ground detection procedure to ensure consistency with other Exelon stations, requiring additional oversight of ground detection activities, and implementing additional human performance improvement initiatives.

.5 Inadequate Clearance Order Results in Fuel Pool Cooling Pump Trip

a. Inspection Scope

The inspectors interviewed licensee personnel and reviewed procedures and corrective action documents to determine the circumstances which resulted in tripping the 1A fuel pool cooling pump during the performance of clearance and tagging activities.

b. Findings

Introduction: A self-revealing Green finding and Non-Cited Violation of Technical Specification 5.4.1 was identified due to the failure to properly implement OP-AA-109-101, "Clearance and Tagging." This failure resulted in tripping the 1A fuel pool cooling pump.

Description: On July 30, 2007, non-licensed operations personnel were tasked with removing the 1B fuel pool cooling filter/demineralizer from service using Clearance Order 55101. During the clearance order activities, Unit 1 control room personnel received numerous alarms associated with the skimmer surge tank and both fuel pool cooling pumps. Three seconds later, the control room received an alarm indicating that the 1A fuel pool cooling pump had tripped. The operations crew immediately responded to the alarms, stopped the clearance order activities, and dispatched a field supervisor to the fuel pool cooling area. Once the fuel pool cooling system performance was stabilized, the licensee initiated a review of this event.

The inspectors held several discussions with operations personnel and management involved in this event. The inspectors also reviewed Procedure OP-AA-109-101, "Clearance and Tagging." Step 7.1.6 of the procedure required that clearance orders

written on systems remaining in service, or partially removed from service, have no adverse impacts on the overall system operation. Step 7.1.12 of the same procedure also required that components manipulated during clearance orders be tracked by tagging the component with a tag, listing the component as a non-carded step, or through the use of an existing procedure which governed the manipulations. Lastly, Step 3.34 required that a senior reactor operator review the clearance order to ensure that it was safe for employees.

The inspectors reviewed Clearance Order 55101 and determined that the 1A fuel pool cooling pump tripped due to a clearance order sequencing error. The sequencing error occurred due to the inappropriate use of a non-carded step within the clearance order, the inadequate review of the clearance order prior to its use, and the failure to ensure that the clearance order had no impacts on the continued operation of the fuel pool cooling system. Specifically, Step 8 of the clearance order directed the non-licensed operators to drain the Unit 1 fuel pool demineralizers and inlet header through the use of a non-carded step and the associated non-carded step note. The inspectors reviewed the non-carded step note and determined that the note contained ten specific component manipulations. This was contrary to the requirements of OP-AA-109-101, Step 7.1.12. The inspectors also determined that the order in which the non-carded step was placed in the clearance order was incorrect. This resulted in inadvertently draining the skimmer surge tank and tripping the 1A fuel pool cooling pump contrary to Step 7.1.6 of the same procedure. Lastly, the senior reactor operator that reviewed Clearance Order 55101 failed to recognize the sequencing error or that the instructions contained in the non-carded step note constituted a procedure.

Analysis: The inspectors determined that this issue was more than minor because, if left uncorrected, the failure to properly implement the clearance and tagging program would become a more significant safety concern. The inspectors evaluated this finding using the Significance Determination Process and determined that this finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier provided by the spent fuel pool. The inspectors concluded that this finding was cross-cutting in the area of Human Performance, Resources, Documentation in that operations personnel did not ensure that Clearance Order 55101 was complete and accurate (H.2(c)).

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented, and maintained for the items specified in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 1c, requires that procedures be established, implemented, and maintained for equipment control. OP-AA-109-101, “Clearance and Tagging,” was the licensee’s procedure that governed equipment control. Step 7.1.6 of the procedure required that clearance orders written on systems remaining in service, or partially removed from service, have no adverse impacts on the overall system operation. Step 7.1.12 of the same procedure also required that components manipulated during clearance orders be tracked by tagging the component with a tag, listing the component as a non-carded step, or through the use of an existing procedure which governed the manipulations. Lastly, Step 3.34

required that a senior reactor operator review the clearance order to ensure that it was safe for employees. Contrary to the above, on July 30, 2007, operations personnel failed to implement Steps 7.1.6, 7.1.12, and 3.34 of OP-AA-109-101. This resulted in draining the Unit 1 skimmer surge tank and tripping the 1A fuel pool cooling pump. Because this violation was determined to be of very low safety significance, and was entered into the corrective action program as Issue Report 655447, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the Enforcement Policy **(NCV 05000254/2007004-05)**. Corrective actions included restoring the fuel pool cooling system to normal operation, establishing a clearance order review board to thoroughly review clearance orders, holding training sessions to ensure that the clearance order writers clearly understood that each clearance order step should contain only one equipment manipulation, and implementing additional actions to improve Operations Department performance.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000254/2005007-02; 05000265/2005007-02: Lack of a Design Analysis Evaluating Secondary Fire Effects of Non-Fused 120 Vac Control Circuitry on the Plants' Fire Protection Safe Shutdown Analysis.

During the Quad Cities Modification/50.59 Inspection, the team discovered that fuse isolation did not exist in certain 120 Vac circuitry configurations. The primary configuration of concern was 120 Vac ungrounded circuits being powered from Control Power Transformers (CPTs) (480 - 120 Vac). The inspectors were concerned that if a fire were to occur in a fire area and cause a fault on one of these unfused 120 Vac circuits, a concurrent fire could occur somewhere else in the circuitry due to the high amperage caused by the faulted condition. Of primary concern was the eventual failure of the CPTs in Motor Control Cabinet (MCC) cubicles.

Because of these concerns, the licensee issued Engineering Analysis QDC-7800-E-1564, "Effect of a Range of Short Circuit Current on the Connected Cables of the Unfused and Ungrounded Control Transformer Secondary Side." This engineering analysis showed that if a fault occurred, the weak link in the circuitry would be the CPT. Additionally, the analysis concluded that even though the CPT would fail, a fire would not occur in the MCC breaker bucket. While the inspectors agreed that it was most probable that the CPTs would fail, they could not agree with the conclusion that a fire would not occur in the MCC bucket. The inspectors did agree, however, that it was most likely that the CPT and connected electrical equipment in the bucket would only overheat and smoke rather than produce flame.

Because the inspectors could not agree that there was no chance of fire in the bucket, the licensee performed an extensive operating experience review. In that review, the licensee was able to show the inspectors at least 50 instances of CPTs failing (at various nuclear plants). In all cases, no fire damage occurred outside of the specific MCC bucket that contained the CPT. The licensee affirmed that they could find no instances of CPT failure that resulted in fire that spread outside of the MCC bucket. The inspectors determined that these examples of CPT failures provided enough evidence to

prove that fire propagation due to a fault in this type of low energy control circuit (120 Vac) was highly improbable. Consequently, the inspectors concluded that adverse secondary fire effects for this type of circuitry at the Quad Cities Nuclear Power Station were highly improbable.

.2 (Closed) Unresolved Item 05000254/2007003-06; 05000265/2007003-06: Review of Unit 1 4160 Volt Breaker Failures.

Introduction: A self-revealing Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to identify and correct a condition adverse to quality. Specifically, the licensee failed to assure that the cause of the March 2007 failure of the 1D residual heat removal pump breaker was promptly identified and corrected. This resulted in the repeated failure of the 1D residual heat removal pump breaker in May 2007.

Description: On January 4, 2006, the 1B core spray pump breaker tripped after attempting to close. The licensee identified that the breaker was misaligned and was protruding ½ inch outside of the breaker cubicle. This event was discussed in Section 4OA3 of NRC Inspection Report Number 05000254/2006005; 05000265/2006005. The event resulted in a self-revealing Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion V, for failing to have an adequate procedure for installing the breakers. The procedure was subsequently revised to ensure that breakers were installed plumb and flush with the breaker cubicles.

On March 9, 2007, the 1D residual heat removal pump's breaker closed-in for six seconds and then tripped open. The licensee initiated troubleshooting efforts but was unable to identify an obvious problem. On March 10 the licensee attempted to re-close the breaker. The breaker tripped immediately. The licensee removed the breaker and noted that the breaker's rack door lever gap was out of tolerance. Another breaker with the proper tolerances was installed and cycled multiple times. No deficiencies were noted during the breaker cycling. Following this testing, operations declared the 1D residual heat removal pump operable. The inspectors questioned the licensee multiple times during the troubleshooting efforts to ensure that all possible failure mechanisms were considered. Based on the January 2006 core spray breaker to cubicle interface issue, the inspectors were specifically concerned that a breaker to cubicle interface issue could have resulted in, or contributed to, the residual heat removal pump breaker failure. However, the licensee failed to take any action to investigate the breaker cubicle interface. In addition, the licensee failed to identify whether the 1D residual heat removal pump breaker was flush and plumb with the cubicle. Therefore, the licensee failed to identify a likely cause of the breaker failure as demonstrated by the subsequent May 7, 2007, breaker failure.

On May 7, 2007, the 1D residual heat removal pump failed to start on demand. The licensee developed and implemented troubleshooting efforts in response to the failure, but little emphasis was initially placed on the breaker cubicle. Under close observation by engineering and maintenance personnel, the operating crew attempted to close the breaker again. The breaker tripped immediately after attempting to close. Following the

second attempt, the licensee removed the breaker from the cubicle and identified strike marks on the cubicle's mechanism operated cell (MOC) switch cam follower. The strike marks were a result of the breaker's spring discharge roller contacting the cam follower when the breaker was installed in the racked in position. In turn, the cam follower applied a load to the breaker's spring discharge roller. The spring discharge roller translated this load through a linkage to the breaker's discharge trip paddle. This made the breaker very susceptible to opening after attempting to close due to internal movement of the breaker as the breaker was closing.

During the licensee's May 2007 troubleshooting efforts, the inspectors noted that the licensee measured various components inside the breaker cubicle. The licensee found that the MOC switch cam follower rod and the vertical dimension of the MOC switch linkage were slightly out of tolerance. This contributed to the contact between the spring discharge roller and cam follower. The out of tolerance condition was corrected. Six days later the licensee attempted to close the 1D residual heat removal breaker as part of the post maintenance testing. The breaker tripped open immediately. The inspectors questioned multiple licensee personnel regarding the potential that changes made to the breaker installation procedure following the 2006 1B core spray breaker failure could have contributed to both of the 1D residual heat removal pump breaker failures. Specifically, the inspectors were concerned that over-racking of the breaker could cause decreasing tolerances between breaker and cubicle components and contribute to interference between components. Although the licensee informed the inspectors that the change to the breaker installation guidance may have contributed to the breaker failures, there were no plans to review breaker racking activities. When the inspectors asked whether additional breaker racking guidance would be provided to operations personnel to minimize the potential for additional breaker failures, they were informed that no additional guidance would be provided. Instead, the licensee modified the breaker cubicle's cam follower by removing approximately 1/4 inch of material. This allowed an additional gap between the cubicle's cam follower and the breaker's discharge roller. Post-maintenance testing was completed satisfactorily.

The inspectors reviewed the licensee's May 2007 breaker installation and maintenance history. The inspectors also reviewed the corrective actions taken after the March 2007 1D residual heat removal breaker failure. The inspectors determined that the front-bottom section of the 1D residual heat removal pump breaker was installed 1/4 inch inside the front of the cubicle. This was contrary to QCOP 6500-07, "Racking in a 4160 Volt Horizontal Type AMHG or G26 Circuit Breaker," which was revised following the failure of the 1B core spray pump breaker in January 2006. Specifically, the procedure required that the breaker be installed nominally plumb and flush with the cubicle. The failure to install the 1D residual heat removal pump breaker plumb and flush with the cubicle allowed the breaker's spring discharge roller to reside further in the cubicle. This contributed to the contact between the MOC switch and the cubicle's cam follower. The inspectors also noted that the licensee was challenged to meet the procedural step to install the 4160 Volt Merlin Gerin AMHG breakers nominally plumb and flush with the cubicles due to inconsistent floor leveling at specific cubicle locations.

Because of this, some of the breakers were installed such that the front-bottom portion of the breaker was slightly inside the cubicle rather than being nominally plumb and flush.

The inspectors also found that the licensee replaced the 4160 Volt GE Magnablast breakers with Merlin Gerin AMHG 4160 Volt breakers between 1995 and 2002. The GE Magnablast breakers had a collar attached to the racking screw that was a positive stop to prevent the breaker from being pulled too far into the cubicle during installation. The Merlin Gerin AMHG 4160 Volt breaker was not designed with a positive stop on the racking screw. The inspectors determined that the absence of a positive stop also contributed to the licensee's breaker failures.

Analysis: The inspectors reviewed multiple documents to determine if the licensee's failure to promptly identify and correct the March 2007 breaker failure constituted a condition adverse to quality. As discussed in Chapter 16 of the licensee's Quality Assurance Topical Report, examples of conditions adverse to quality included failures, malfunctions, adverse trends, deficiencies, deviations, defective material, design errors, equipment, and nonconformance to specified requirements. Based on the examples provided in the Topical Report, the inspectors concluded that the failure to promptly identify and correct the 1D residual heat removal pump breaker failure in March 2007 was a condition adverse to quality which was required to be promptly corrected.

The inspectors determined that this issue was more than minor because if left uncorrected, the misalignment between safety-related breakers and cubicles would continue to result in the inoperability of equipment important to safety. In addition, the previous failure of the 1D residual heat removal pump breaker impacted the operability, availability, and reliability of this mitigating system. The inspectors concluded that this finding was of very low safety significance (Green) because it was not a design deficiency, did not result in the total loss of a safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding was determined to be cross-cutting in the area of Problem Identification and Resolution, Corrective Action Program, Evaluation because the licensee failed to thoroughly evaluate the March 2007 breaker failure to ensure that the resolution addressed the cause and extent of condition (P.1(c)).

Enforcement: Title10 CFR Part 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, between March 9 and May 7, 2007, the licensee failed to ensure that the cause of the 1D residual heat removal pump breaker failure was promptly identified and corrected. However, because this violation was of very low safety significance, and because the issue was entered into the corrective program as Issue Report 626108, the issue is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000254/2007004-06; 05000265/2007004-06)**. Corrective actions for this issue included performing an extent of condition review and modifying all of the Unit 1 Merlin Gerin breakers and cubicles. Five additional Unit 1 breakers showed evidence of contact between the MOC switch cam follower and the breaker's spring discharge roller. At the conclusion of the

inspection period, 17 of the 47 Unit 2 breaker cubicles had also been modified. The remainder will be modified during the next Unit 2 refueling outage. The licensee was also evaluating the need for additional changes to the breaker installation procedure.

.3 (Closed) Unresolved Item 05000254/2007003-01; 05000265/2007003-01: Aerosol Can Found in Cable Spreading Room.

Introduction: An inspector identified Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix R, Section III, Criterion K(2) were identified due to the licensee's failure to adequately control transient combustible material in a transient combustible exclusion zone. Specifically, the inspectors discovered two large cardboard boxes and an aerosol spray can that contained methyl alcohol improperly controlled and unattended in the cable spreading room.

Description: On May 3, 2007, the inspectors conducted an inspection of the cable spreading room. The inspectors identified two cardboard boxes and an aerosol can left unattended in the cable spreading room with no associated transient combustible permits. The boxes were located in a staging area and were labelled as material for an upcoming Unit 1 modification. The aerosol can was located in a tool pouch that was placed on the floor behind a vertical cable riser. The label on the can identified its contents to be a product by the name of EZY-TAP, which has a fire hazard rating of 2 and contained methyl alcohol as one of its ingredients.

Procedure OP-AA-201-009, "Control of Transient Combustible Material," defined a transient combustible exclusion zone as "an area in the plant in which transient combustible material is prohibited when **not** constantly attended, depending on plant operating conditions." Furthermore, Attachment 8 of this procedure stated that the cable spreading room was a transient combustible exclusion zone.

Analysis: The inspectors determined that this issue was more than minor because it could be viewed as a precursor to a significant event, i.e., fire impacting multiple pieces of safety-related equipment. Specifically, multiple vertical cable risers were located within the zone of influence for the aerosol can (based on review of Inspection Manual Chapter (IMC) 0609, "Fire Protection Significance Determination Process," Appendix F, Table 2.3.2, for a 70 kilowatt fire for thermoplastic cables). Consequently, the aerosol can presented a credible fire scenario involving equipment important to safety (such as the cables for equipment important to safety located within the vertical cable tray risers and a horizontal cable tray directly above). As such, the finding warranted a significance determination evaluation.

The inspectors determined that the unattended aerosol can represented a high degradation of the licensee's transient combustible controls program due to the flammable nature of the propellant used in the aerosol can. Based on discussions with the licensee, it was determined that the aerosol can was most likely left in the cable spreading room by contract personnel in the 3 to 30 day time frame. As such, the inspectors determined that the duration factor (DF) of 0.1 was appropriate based on IMC 609F, Step 1.4.

Based on a review of IMC 0609F, Attachment 4, "Fire Ignition Source Mapping Information: Fire Frequency, Counting Instructions, Applicable Fire Severity Characteristics, and Applicable Manual Fire Suppression Curves," the inspectors determined that the base fire frequency, F, was 1.7×10^{-4} per year for transient combustibles having a medium likelihood rating. The inspectors considered the medium likelihood rating appropriate because the cable spreading room was generally accessible and access to the room was not controlled. In addition, the room did contain some components other than cables. As discussed in IMC 0609, Attachment 5, "Characterizing Non-Simple Fire Ignition Sources," no weighting factor was applied to the fire frequency because the inspectors observed transient fuels in a specific location. The fire frequency, F, was raised by one level of likelihood to 1.7×10^{-3} per year for transient combustibles having a high likelihood rating because the finding was associated with the licensee's combustible controls program. The inspectors conservatively assumed that the probability of non-suppression was 1.0 because the affected cables were within the zone of influence for the aerosol can and located where a fire would not be readily suppressed by the installed suppression system (i.e., a wet-pipe sprinkler system) due to obstructions. (The inspectors noted that the installed suppression system would control a fire such that it would be unlikely that additional cables beyond the zone of influence would be affected by the fire.) Based on the cable risers and the one cable tray within the zone of influence, the inspectors determined that the following Unit 2 equipment would potentially be affected by a fire:

- Main Steam Isolation Valves;
- Feedwater Pumps;
- Condensate and Condensate Booster Pumps;
- Safe Shutdown Make-Up Pump (control room control affected only, local operation still available);
- High Pressure Coolant Injection;
- Reactor Core Isolation Cooling;
- Division 2 of Residual Heat Removal/Low Pressure Coolant Injection;
- Division 2 of Low Pressure Core Spray; and
- 1B Instrument Air Compressor (other air compressors sufficient to supply instrument air).

The inspectors determined that the following equipment and functions could be credited:

- Local Control of Safe Shutdown Make-Up Pump;
- Automatic Depressurization System;
- Division 1 of Residual Heat Removal/Low Pressure Coolant Injection;
- Division 1 of Low Pressure Core Spray;
- Containment Venting;
- Control Rod Drive;
- Survival of Injection Path Following Containment Failure; and
- Instrument Air.

Based on review of the transients work sheet of the "Risk Informed Notebook for the Quad Cities Nuclear Power Station, Units 1 and 2," Revision 2.1, the inspectors

determined that at least 4 points of mitigating systems credit was available resulting in a conditional core damage probability, CCDP, of 1×10^{-4}.

The inspectors determined that the change in core damage frequency, ΔCDF, would be 1.7×10^{-8} per year based on the following equation from IMC 0609F, Step 2.9:

$$\Delta CDF = DF \times F \times SF \times AF \times PNS \times CCDP$$

Where:

DF	Duration factor = 0.1 as discussed above.
F	Fire Frequency = 1.7×10^{-3} per year as discussed above.
SF	Severity Factor = 1.0 because the fire scenario does not involve any severity factors.
AF	Ignition Source Specific Frequency Adjustment Factor = 1.0 because the fire frequency adjustment for a finding involving the combustible controls program was performed for the fire frequency.
PNS	Probability of Non-Suppression = 1.0 as discussed above.
CCDP	Conditional Core Damage Probability = 1×10^{-4} as discussed above.
ΔCDF	= DF × F × SF × AF × PNS × CCDP = $0.1 \times 1.7 \times 10^{-3} \times 1.0 \times 1.0 \times 1 \times 10^{-7}$ = 1.7×10^{-8} per year

Based on a review of IMC 0609F, Table 2.9.1, "Risk Significance Based on ΔCDF," the inspectors determined that the issue was of very low safety significance (Green). The inspectors also determined that this finding was cross cutting in the area of Human Performance, Work Practices, Oversight, in that the licensee did not ensure that supervisory and management oversight of work activities, including contractors, was appropriate such that nuclear safety was supported (H.4(c)).

Enforcement: The Quad Cities Nuclear Power Station's Operating Licenses DPR-29 and DPR-30, Section h.3.F, stated that the licensee shall implement and maintain in effect all provisions of the approved Fire Protection Plan as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated July 27, 1979, and subsequent supplements. OP-AA-201-009, "Control of Transient Combustible Material," required combustibles in a transient combustible exclusion zone to be attended. Attachment 8 of OP-AA-201-009 also designated the cable spreading room as a transient combustible exclusion zone. Contrary to the above, on May 3, 2007, the licensee failed to adequately implement OP-AA-201-009 in that unattended transient combustible material was found staged in the cable spreading room. Because this violation was of very low safety significance, and because the issue

was entered into your corrective action program as Issue Reports 625097 and 636793, the issue is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000265/2007004-07)**. Corrective actions for this issue included removing the NRC-identified transient combustible material from the cable spreading room, informing personnel of the ineffective control of transient combustible materials, providing additional oversight of the transient combustible control program, and clearly labeling the cable spreading room as a transient combustible exclusion zone.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Tulon and other members of licensee management at the conclusion of the inspection on October 2, 2007. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the CDBI followup corrective action inspection were presented to Mr. W. Beck, Regulatory Assurance Manager, and other licensee engineering staff members at the conclusion of the inspection on July 19, 2007.
- Maintenance Effectiveness Periodic Evaluation with Mr. T. Tulon, Site Vice President, on September 28, 2007.
- Public radiation safety radioactive waste processing and transportation program inspection with Mr. R. Gideon and other licensee staff on August 31, 2007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

T. Tulon, Site Vice President
R. Gideon, Plant Manager
B. Adams, Engineering Manager
D. Barker, Work Control Manager
W. Beck, Regulatory Assurance Manager
D. Craddick, Maintenance Manager
D. Moore, Nuclear Oversight Manager
K. Moser, Training Manager
V. Neels, Chemistry/Environ/Radwaste Manager
K. Ohr, Radiation Protection Manager
R. Svaleson, Operations Manager

Nuclear Regulatory Commission personnel

M. Ring, Chief, Reactor Projects Branch 1
M. Thorpe-Kavanaugh, NRR Project Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000254/2007004-01	NCV	Documentation Issues Result in Unexpected Safety-Related Valve Leakage
05000254/2007004-02; 05000265/2007004-02	URI	Review of Ultimate Heat Sink Surveillance Requirements and Emergency Action Levels
05000254/2007004-03	NCV	Failure to Have Adequate Procedures and Use Human Performance Tools Results in Reactor Water Cleanup Isolation
05000254/2007004-04; 05000265/2007004-04	NCV	Failure to Follow Procedures and Use Human Performance Tools Results in Reactor Building Ventilation Isolation

05000254/2007004-05	NCV	Inadequate Clearance Order Results in Fuel Pool Cooling Pump Trip
05000254/2007004-06; 05000265/2007004-06	NCV	Failure to Promptly Correct March 2007 1D Residual Heat Removal Pump Breaker Failure
05000265/2007004-07	NCV	Failure to Control Transient Combustibles in the Cable Spreading Room

Closed

05000254/2007004-01	NCV	Documentation Issues Result in Unexpected Safety-Related Valve Leakage
05000254/2007004-03	NCV	Failure to Have Adequate Procedures and Use Human Performance Tools Results in Reactor Water Cleanup Isolation
05000254/2007004-04; 05000265/2007004-04	NCV	Failure to Follow Procedures and Use Human Performance Tools Results in Reactor Building Ventilation Isolation
05000254/2007004-05	NCV	Inadequate Clearance Order Results in Fuel Pool Cooling Pump Trip
05000254/2007004-06; 05000265/2007004-06	NCV	Failure to Promptly Correct March 2007 1D Residual Heat Removal Pump Breaker Failure
05000265/2007004-07	NCV	Failure to Control Transient Combustibles in the Cable Spreading Room
050002547/06-001-01	LER	Failure of the 1B Core Spray Pump Breaker to Operate due to Racking Deficiency
05000254/07-001	LER	Two Main Steam Safety Valves and One Main Steam Safety/Relief Valve Outside of Technical Specification Allowed Tolerance due to Setpoint Drift
05000254/2005007-02; 05000265/2005007-02	URI	Lack of a Design Analysis Evaluating Secondary Fire Effects of Non-Fused 120 Vac Control Circuitry on the Plants' Fire Protection Safe Shutdown Analysis
05000254/2007003-01; 05000265/2007003-01	URI	Aerosol Can Found in Cable Spreading Room

05000254/2007003-06; 05000265/2007003-06	URI	Review of 4160 Volt Breaker Failures

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R04 Equipment Alignment

QCIS 0300-01; Unit 1 Scram Discharge Volume FCI Level Switch Calibration and Functional Test; Revision 13
QCIS 0300-05; "Unit 2 Scram Discharge Volume FCI Level Switch Calibration and Functional Test;" Revision 0
QCIS 0300-02; "Unit 1 Division I Scram Discharge Volume Rochester Instruments Calibration and Functional Test;" Revision 8
QCIS 0300-07; "Unit 1 Division II Scram Discharge Volume Rochester Instruments Calibration and Functional Test;" Revision 1
QCIS 0300-03; "Unit 1 Division I Scram Discharge Volume Barton Level Transmitter Calibration;" Revision 8
QCIS 0300-10; "Unit 1 Division II Scram Discharge Volume Barton Level Transmitter Calibration;" Revision 1
QCIS 0300-08; "Unit 2 Division I Scram Discharge Volume Rochester Instruments Calibration and Functional Test;" Revision 1
QCIS 0300-09; "Unit 2 Division II Scram Discharge Volume Rochester Instruments Calibration and Functional Test;" Revision 1
QCIS 0300-11; "Unit 2 Division I Scram Discharge Volume Barton Level Transmitter Calibration;" Revision 1
QCIS 0300-12; "Unit 2 Division II Scram Discharge Volume Barton Level Transmitter Calibration;" Revision 1
QCTS 0820-04; "Leak Rate Test of the Scram Discharge Volume Vent and Drain Valves;" Revision 5
QCOP 0500-04; "Inserting Manual Scrams;" Revision 12
QCOP 0300-28; "Alternate Control Rod Insertion;" Revision 25
QCGP 2-3; "Reactor Scram;" Revision 61
QCOS 0300-20; "Scram Discharge Volume Level Switch Functional Test;" Revision 8
QOS 0005-01; "Operations Department Weekly Summary of Daily Surveillance;" Revision 123
QCOS 0300-13; "Scram Discharge Volume Bypass Switch Rod Block Functional Test;" Revision 11
QCOS 0300-05; "SDV Vent Ball Check Valve Exercise;" Revision 1
QCOS 0500-05; "Mode Switch in Shutdown, Scram Solenoid Valves, and SDV Vent & Drain Test at Shutdown;" Revision 20
QCOS 0300-11; "Scram Discharge Volume Vent and Drain Valve Testing;" Revision 9
IST Valve Test Acceptance Criteria Sheet
Drawing M-41; Unit 1 Diagram of Control Rod Drive Hydraulic Piping; Sheet 3
Drawing M-83; Unit 2 Diagram of Control Rod Drive Hydraulic Piping; Sheet 3
QOM 1-6600-01; Unit 1 Diesel Generator Valve Checklist; Revision 21

QCOP 6600-01; Diesel Generator 1(2) Preparation for Standby Operation; Revision 33

1R06 Flood Protection Measures

Issue Report 644122; Floor Drain Sock Plugged with Paint; dated June 23, 2007
NRC Information Notice 92-69; Water Leakage From Yard Area Through Conduits Into Buildings; dated September 22, 1992
NRC Information Notice 2003-08; Potential Flooding Through Unsealed Concrete Floor Cracks; dated June 25, 2003
NRC Information Notice 2005-11; Internal Flooding/Spray Down on Safety-Related Equipment due to Unsealed Equipment Hatch Floor Plugs and/or Blocked Floor Drains; dated May 6, 2005
NRC Information Notice 2007-01; Recent Operating Experience Concerning Hydrostatic Barriers; dated January 31, 2007
Updated Final Safety Analysis Report
CC-AA-201; Plant Barrier Control Program; Revision 6
QCOA 3900-01; Service Water System Failure; Revision 11
QOA 0030-01; Condenser Pit Room or Condensate Pump Room Flooding; Revision 10
QCAP 0250-06; Control of In-Plant Flood Barriers and Watertight "Submarine" Doors; Revision 10

1R11 Licensed Operator Requalification

IR 665157; Licensed Operator Requalification Training As-Found Failure; dated August 27, 2007

1R12 Maintenance Effectiveness

ER-AA-310-1004; Attachment 8 - Functional Failure Cause Determination; 4160 Volt Switch Gear; Maintenance Rule Function Z6700-05; Revision 5
Maintenance Rule Periodic Refueling Assessment; May 1, 2002 to
May 1, 2004; dated July 29, 2004
Maintenance Rule Periodic Refueling Assessment; May 1, 2004 to
May 1, 2006; dated July 31, 2006
ER-AA-310-1007; Maintenance Rule - Periodic (a)(3) Assessment; Revision 4
ER-AA-310-1004; Maintenance Rule - Performance Monitoring; Revision 5
Maintenance Functional Failures from May 2004 to May 2006; dated May 2006
Quad Cities Station Maintenance Rule Performance Criteria; dated September 2007
Maintenance Rule (a)(1) Action Plan for Control Room HVAC; dated September 22, 2005
Maintenance Rule (a)(1) Action Plan for Refueling Bridge Crane; dated September 9, 2004
Maintenance Rule (a)(1) Action Plan for 480 V MCCs; dated March 28, 2005
Maintenance Rule (a)(1) Action Plan for Unit 2 Fuel; dated December 8, 2005
Expert Panel Meeting Minutes; dated July 14, 2005
Expert Panel Meeting Minutes; dated October 13, 2005
Health and Status Reports for 480 Vac System, High Pressure Core Injection, Unit 2 Fuel, Refueling Bridge Crane and Control Room HVAC; dated September 2007
Issue Report 209029; 1/2A CR HVAC Dampers Failed to Position on System Restart; March 17, 2004

Issue Report 221464; Review of Peach Bottom HPCI Failure; dated May 16, 2004
Issue Report 294136; Z7800-03 MR Function exceeded CBM Criteria; dated December 13, 2004
Issue Report 315350; MO 2-1001-26B, Inbrd DW Spray Isolation Valve Will not Close; dated March 21, 2005
Issue Report 346220; 1/2A SB HVAC (B Compressor) Tripping; dated June 22, 2005
Issue Report 378095; Fuel Failure on Quad Unit Two; dated September 26, 2005
AT-563845-04; Quad Cities Maintenance Rule Pre-NRC Inspection Check-In-Self-Assessment, dated July 31, 2007

1R13 Risk Assessments and Emergent Work
Work Week Safety Profiles
Operations Department Daily Orders
Shutdown Safety Risk Assessments

1R15 Operability Evaluations
IR 666841; QGA Required Equipment Work Request/Work Order Priority Not Addressed in WC-AA-106; dated August 31, 2007
QCOP 1200-10; Injection of Boron Using the Reactor Water Cleanup System; Revision 20
Issue Report 650893; Wire in Hotbox (CAM) Does Not Match What Was Presented from Vendor; dated July 17, 2007
Issue Report 649225; 2A Residual Heat Removal Cooler Needs Cleaned; dated July 12, 2007
QCOS 5750-09; ECCS Room and DGCWP Cubicle Cooler Monthly Surveillance; Revision 30
Issue Report 669981; Additional Information Added to EDG HVAC Operability Determination; dated September 11, 2007
Issue Report 656537; Prompt Op Evaluation Not Requested for Degraded Condition; dated August 2, 2007
Letter from E.W. Ralls, Electro-Motive Division to Brad Abernathy, MKW Power Systems; Maximum Control Cabinet Temperatures; dated March 1, 1991

1R19 Post Maintenance Testing
QCTS 0430-03; Standby Gas Treatment System Inplace Charcoal Adsorber Freon 11 Leak Test; dated July 17, 2007
Issue Report 650820; “B” Standby Gas Treatment System Trays Replacement Missing 6 Test Canisters
Issue Report 626108; 1D RHR Pump Breaker Tripped; dated May 7, 2007
Work Order 1021847; 1D RHR Pump Breaker Troubleshooting; dated May 7, 2007
Issue Report 669786; As-left LLRT of 220-01 Valve Above Action Limit; dated September 10, 2007
Engineering Change Evaluation 367324; Evaluate Required Action Limit Increase for 1-220-01 and 1-220-02; dated September 10, 2007

1R20 Refueling and Outage Activities
Issue Report 669784; Outage Lessons Learned Pressure Seal Rings; dated September 8, 2007
Issue Report 669178; High Pressure Coolant Injection Valve Leaking 0.5 Gallons Per Minute; dated September 8, 2007
QCMM 1515-17; Pressure Seal Gate Valve Maintenance; Revision 9

1R22 Surveillance Testing

Issue Report 669224; Discontinuation of Online Exercising of High Pressure Coolant Injection and Reactor Core Isolation Cooling Primary Containment Isolation Valves; dated September 8, 2007

2PS2 Radioactive Material Processing and Transportation

Annual Radioactive Effluent Release Reports for 2005 and 2006; Tables Summarizing Solid Waste and Irradiated Fuel Shipments; dated April 28, 2006 and April 30, 2007
RW-AA-100; Process Control Program for Radioactive Wastes; Revision 5
Interim Radwaste Storage Facility Waste Liner Inventory; dated August 27, 2007
RP-AA-605; 10 CFR 61 Program; Revision 1
RP-AA-601; Surveying Radioactive Material Shipments; Revision 6
RP-AA-600-1001; Exclusive Use and Emergency Response Information; Revision 3
RP-AA-602; Packaging of Radioactive Material Shipments; Revision 12
FO-OP-032-44506; Setup and Operating Procedure for the RDS-1000 Unit at Quad Cities; Revision 5
FO-OP-048; Procedure for Installation and Operation of the In-Line Sampler; Revision 3
RP-QC-605-1001; 10 CFR 61 Waste Stream Sampling and Analysis; Revision 1
10 CFR 61 Database Analyses and Associated Sample Validation Information for Various Waste Streams; dated various periods in 2006 and 2007
Teledyne Brown Engineering, Inc. Report of Analyses for 10 CFR 61 Waste Steams; dated September 01, 2006, March 30, 2007 and January 22, 2007
RP-QC-605-1001, Attachment 1; Trending for Shifts in Scaling Factors; Weekly Results for 2006 thru June 28, 2007
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-05-004; Irradiated Hardware to Low Level Waste Burial Site; dated October 17, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-07-010; Irradiated Hardware to Low Level Waste Burial Site; dated August 23, 2007
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-05-001; De-watered Mechanical Filters to Low Level Waste Burial Site; dated May 24, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-05-323; Contaminated Valves to Vendor; dated February 22, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-05-159; Contaminated Equipment to Low Level Waste Burial Site; dated December 13, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment No. QC-06-120; DAW to Waste Processor; dated August 22, 2006
Training Lesson Plan; Nuclear General Employee Training; Revision 33
Training Lesson Plan; Radiation Protection Technician Retraining - Radioactive Material Shipments; Revision 11
Training Lesson Plan; Station/Contract Laborers - Low Level Waste Processing; Revision 05
Hazmat Training Certificates (General Awareness and Function Specific) for Storeroom/Warehouse Staff; dated various periods in 2005 and 2006
RP-AA-500-1001; Requirements for Radioactive Materials Stored Outside; Revision 0
Outdoor Container Satellite RCA Surveillance Records; dated August 10, 2007

Focused Area Self-Assessment Reports; Radioactive Material Processing/Transportation; dated August 7, 2007; and Type A or Greater Waste Shipments to Disposal Sites; dated October 19, 2006
Nuclear Oversight Audit No. DRE-06-04; Chemistry, Radwaste, Effluent and Environmental Monitoring; dated May 3, 2006
Nuclear Oversight Objective Evidence Reports for Audit NOSA-QDC-06-04; Waste Classification/Characteristics and General Requirements for Shipments/Packaging; Audit dates April 17 - 28, 2006
AR 00223877; Nuclear Oversight Identified Improperly Abandoned Equipment; dated May 26, 2004
Radwaste Abandonment Projects Report; dated June 29, 2006
AR 00650483; Incorrect Control Rod Blade Shipped for Burial in 2005; dated July 16, 2007
AR 00552513; Radwaste Shipment Container Error; dated November 1, 2006

4OA1 Performance Indicator Verification
Noble Gas, Gaseous Tritium and Liquid Effluent Summary Data for 2006 and 2007.
Monthly Calculation Summaries of Liquid and Gaseous Effluent Doses for October 2006 thru August 2007
Unit 1 and 2 Gaseous Release Permit Reports for the Main Chimney & Reactor Building Vent, and associated Gamma Isotopic Analyses Results; selected periods in 2007

4OA2 Identification and Resolution of Problems
QDC-3300-M-0489; Useable Water Volume of Contaminated Condensate Storage Tanks for HPCI and RCIC, Including Vortexing Considerations; Revision 3
QCOP 6500-28; 4KV/480V Bus Loading Profiles; Revision 3
QCOP 6600-10; Diesel Generator ½ Simultaneous Supply to Buses 13-1 and 23-1; Revision 10
QCOP 6600-16; Unit 1 Diesel Generator Simultaneous Supply to Buses 14-1 and 24-1; Revision 6
QCOP 6600-17; Unit 2 Diesel Generator Simultaneous Supply to Buses 24-1 and 14-1; Revision 6
Issue Report 00650819; U1 Alt. 125 Vdc and 3D Battery Corrosion; July 17, 2007
Issue Report 00521503; CDBI NRC Inspection - Incorrect Input Parameter for Calculation; August 17, 2006
Issue Report 00556340; Review all CDBI Calculation Issues for Possible CCA; November 10, 2006
Issue Report 00582097**;** OPS Department Fundamental Windows for December 2006; January 19, 2007
Issue Report 00591442; Effect of EDG Frequency on Loading and Pump Flows; February 14, 2007
Issue Report 00642890; U2 HPCI - Raise MSC/MGU High Speed Stops; June 21, 2007
Issue Report 00647072; U1 125 Vdc Battery Corrosion; July 3, 2007
Issue Report 00647070; Issues Noted with Maintenance of Station Batteries; July 3, 2007
Issue Report 00647244; Out of Tolerance on Battery Intercell Resistance Readings; July 3, 2007
LN-6900; DC Distribution and Batteries; April 11, 2007
N-2007-2; Battery Corrosion DLA; February 22, 2007

Nuclear Event Report (NER) QC-06-107 Green; Technical Specification for Safety Related Intercell Resistance Determined to be Non-Conservative; November 28, 2006
ATIN 520627; Root Cause Investigation Report - Inadequate Corrosion Management of Safety Related Batteries; November 9, 2006
Status of Corrective Actions to Various Engineering issues Identified in Issue Reports initiated during CDBI in 2006
EC 366520; Review of Design Basis Differential Pressure Requirements for Flowrates for Pumps in the IST Program; July 13, 2007
EC 364821; Evaluate 2-Dimensional Battery Corrosion; April 24, 2007
LS-AA-125; Corrective Action Program (CAP) Procedure; Revision 11
LS-AA-125-1004; Effectiveness Review Manual; Revision 2
QCEPM 0100-01; Station Battery Preventive Maintenance; Revision 28
QCOS 6900-01; Station Battery Weekly Surveillance; July 17, 2007
QCOS 6900-02; Station Battery Quarterly Surveillance; Revision 28
QCOS 6900-14; Station Battery Allowable Value Verification Surveillance; Revision 12
QCOS 6900-15; Station Battery Monthly Surveillance; Revision 25
QCOS 2300-05; Quarterly HPCI Pump Operability Test; Revision 59

4OA3 Event Followup

Apparent Cause Report 655447; Improper Sequencing of a Clearance Order Caused Trip of 1A Fuel Pool Cooling Pump; dated September 26, 2007
Issue Report 655447; Unexpected Fuel Pool Cooling Alarms; dated July 30, 2007
QCOP 9000-04; Plant Radio System Operation; Revision 6
Issue Report 663190; Unit 1 Reactor Water Cleanup Isolation; dated August 22, 2007
HU-AA-101; Human Performance Tools and Verification Practices; Revision 4
Issue Report 669304; Unexpected Isolation of Unit 2 Reactor Building Vents; dated September 9, 2007

4OA5 Other Activities

QDC-7800-E-1564; Effect of a Range of Short Circuit Current on the Connected Cables of the Unfused and Ungrounded Control Transformer Secondary Side; dated January 31, 2007
Issue Report 382847; MCC Control Power Transformer Circuit Issues; dated October 6, 2005
Micron Test Report; CPT Destructive Test - Direct Short Circuit Applied to Secondary Winding; dated February 8, 1984
Zion Interoffice Memorandum; Fusing of Control Power Transformers; dated February 11, 1985
QCOP 6500-07; Racking in a 4160 Volt Horizontal Type AMHG or G26 Circuit Breaker; Revisions 18, 19, 20, and 21
Quad Cities Quality Assurance Topical Report; Chapter 16 - Company Program to Identify and Correct Conditions Adverse to Quality; Revision 79
Vendor Document; Golden Gate Switchboard Company; FIP 3081-3; MOC Assembly Installation Procedure; Revision 6
Issue Report 631363; Plant Startup Issue - Q1R19 Breaker Tripped at Start Attempt; dated May 19, 2007
Modification Work Package M4-1-91-033B; Perform Bus 14 Switchgear Enhancement; dated June 8, 1994
Work Request 604-1-93-282; Install New MOC Switch Linkage Assembly for Bus 14-1; dated July 13, 1994

Work Order 97081771; EM Cubicle Modification Bus 14-1 Cubcile 4, 1D RHR Pump Motor Breaker; dated November 13, 2002
Work Order 1028556; Strike Marks on MOC Switch CAM Follower at Bus 14 Cubicle 14; dated May 13, 2007
Issue Report 626108; 1D RHR Pump Breaker Tripped; dated May 7, 2007
Work Order 1021847; 1D RHR Pump Breaker Troubleshooting; May 7, 2007

LIST OF ACRONYMS USED

AF	Ignition Source Specific Frequency Adjustment Factor
AR	Assignment Report
AT	Action Tracking Items
CCDP	Conditional Core Damage Probability
CDBI	Component Design Basis Inspection
ΔCDF	Change in Core Damage Frequency
CPT	Control Power Transformer
DAW	Dry-Active Waste
DF	Duration Factor
DOT	Department of Transportation
F	Fire Frequency
HVAC	Heating Ventilation Air Conditioning
IMC	Inspection Manual Chapter
IR	Issue Report
IRSF	Interim Radwaste Storage Facility
LSA	Low Specific Activity
MCC	Motor Control Cabinet
MOC	Mechanical Operated Cell
NCV	Non-Cited Violation
ODCM	Offsite Dose Calculation Manual
PNS	Probability of Non-Suppression
PRA	Probabilistic Risk Assessment
Radwaste	Radioactive Waste
SF	Severity Factor
SSC	Structures, Systems, and Components
UFSAR	Updated Final Safety Analysis Report
Vac	Volt Alternating Current
Vdc	Volt Direct Current